NUGENIA Global Vision

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A great deal of additional information on NUGENIA is available on the Internet.
It can be accessed through www.nugenia.org.

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1 Introduction

The document *NUGENIA Roadmap 2013 - NUGENIA Roadmap* challenges and priorities[1] - issued in October 2013 and presented at the FISA-conference-2013 as well as the 4th SNETP General Assembly in Vilnius was intended "to provide a single reference document for NUGENIA members and the wider nuclear generation II and III R&D community in Europe and in the World" on R&D challenges and priorities for light water fission reactors of current and future generations.

The NUGENIA Roadmap 2013 is a reference state of the art document to be periodically revised. It describes and shortly analyses the main R&D challenges in the field of fission GEN II - GEN III reactors and provides the readers with a preliminary and tentative list of practical proposals for prioritization. According to its very nature[1] , the document neither gives detailed information on the context of the R&D topics and status of the art, nor accurate description of challenges and tasks for action[2] .

Accordingly, it is to be complemented by a more comprehensive and detailed matter-of-fact oriented document aiming at supporting R&D actors in the investigation and the elaboration of suitable proposals for public – private collaborative R&D work within NUGENIA community and outside. That is the main scope of the present document, titled NUGENIA Global Vision Document.

Actually, when NUGENIA was originally established on November 14 2011, its R&D content was split into 7 TAs - Technical Areas - one out of them (TA7) being basically a cross-cutting one. The list of these original TAs is provided here below:

1. Safety and Risk of NPPs, now changed to Plant Safety and Risk Assessment,
2. Severe accidents
3. Improved Reactor Operation
4. Integrity Assessment of Systems, Structures and Components
5. Fuel Development, Waste and Spent fuel Management and Decommissioning
6. Innovative Light Water Reactor Design, now changed to Innovative LWR Design and Technology
7. Harmonisation[3]

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[1] The document is intented to address decision makers of NUGENIA's Partners, the EC and the internatinal community.

[2] It is widely aknowledged within the NUGENIA's community that the elaboration of project proposals, as well as the further investigation to precise the tasks and create new knowledge gather forces to progress.

[3] The possibility to drop TA7 out - for sake of convenience - and redistribute its content among the other TAs has been proposed at the 2014 NUGENIA's GA, addressed during and after the 2014 NUGENIA's Forum in Madrid, as well as in the third NUGENIA's TALs meeting held in Paris, June 2014. The final decision on TA7 was finally taken by the NUGENIA's EXCOM, during its 27th meeting, held in Otaniemi, September 2014. It was stated to

- Redistributing the STA7.1 and STA7.2 content, mainly moving it towards TA1, TA4, but other TAs have also been updated to include harmonisation - and particularly pre-normative research related - issues.
- Keeping the remainder, i.e. STA7.3 and STA7.4 content, within TA7, while rescaling the STA to ST7.1 and STA7.2 respectively and complementing with addition of topics on education and training.

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Later on, the ENIQ community voted unanimously joining NUGENIA and an additional cross-cutting TA was identified, regrouping the topics in ENIQ objectives:

8. In-service Inspection and Non-Destructive Examination now changed to In Service Inspection, Inspection Qualification and NDE Evaluation.

In 2014, the original title of TA8 was modified on TAL's request to In Service Inspection, Inspection Qualification and NDE Evaluation, for precision sake.

The present NUGENIA Global Vision Document - a living one - has been elaborated and written by the NUGENIA's Technical Area Leaders (the TALs) under the coordination of the Technical Coordinator (TA), with the valuable contribution from the members of the Secretariat and the EXCOM, as well as the STALs (Sub-Technical Area leaders) and some volunteers participants in the TA activity.

The document is to be issued by March 2015, approved by the NUGENIA GA and divulgated at the NUGENIA's 2015 Forum. It is to be periodically revised and up-dated[4] , upon the EXCOM decision.

The NUGENIA Global Vision Document provides a quite detailed description of the technical and scientific content of the above-mentioned TAs, while addressing the main inherent R&D objectives, recalling their general scope and state of the art and outlining the main R&D challenges in the medium and long term. The main references, abbreviations and contributors to the task are also mentioned and interface problems addressed.

Moreover, it includes outcomes from the FP7-NUGENIA+-project, especially its workpackage (WP1) concerning the main orientation and challenges and prioritisation established by the EXCOM and addresses the high level NUGENIA R&D objectives identified in the SNETP Deployment strategy document, under elaboration.

[4] It has been agreed among NUGENIA's Partners to provide a revision of the document on three-year basis (that means that the REV 1 is expected March 2018. In the mean-time a revision of the NUGENIA Roadmap 2013 should also be prepared and issued.

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2 TECHNICAL AREA 1 – Plant Safety and Risk Assessment (TA 1)

Technical Area Leader: Pavel Kral (UJV)

2.1 Executive Summary

2.1.1 Scope

Assessment of nuclear power plant safety and risk is a vital task and even a necessary condition for plant licensing, start-up and its safe operation. Original approach with conservative deterministic analyses of spectrum of transients and accidents up to maximal design basis accident (DBA) documented in Safety Analysis Report (SAR) has been gradually extended by probabilistic risk assessment, human reliability analysis, assessment of external hazards, application of best-estimate methodology to safety analyses, analyses of extended design basis events etc.

The extension of plant safety and risk assessment is accompanied by progress and development of computational tools which are utilized for safety and risk assessment. Advanced computer codes utilized for DBA analyses are continuously developed. Shift from 1-dimensional to 3-D modelling, coupling of system thermal-hydraulic codes with core physics and/or computational fluid dynamics codes (CFD) are the task being solved at present. Method and programmes utilized for probabilistic risk assessment has developed to complex computational tools enabling quantification of plant risk in both nominal and shutdown conditions include human reliability analysis (HRA), external hazards, grid impact etc. Combining of deterministic and probabilistic method is also a very promising direction of plans safety assessment.

The advanced methods and tools for plant safety and risk assessment enable upgrading of reactor safety systems to handle new safety demands, effective replacement of obsolete components and support of LTO.

Scope of the effort in TA1 ranges from support of development of more advanced and complex computational tools and methodologies, deepening validation of the computational tools and identification of missing experimental and plant data needed for computer codes and methods validation over identification and reduction of uncertainties and more exact quantification of safety margins to designing of upgraded and new reactor safety system.

2.1.2 Objectives

The primary aim of the effort in Technical Area 1 is the identification of R&D topics connected with development, assessment and application of state-of-the-art methods and tools for NPP safety and risk assessment. Elaboration of these topics and pushing the methods and tools to higher level is a natural content of the work in TA1. Seven technical sub-areas of TA1 address the seven major objectives listed below:

- Advancements in NPP probabilistic assessment and human reliability analysis – improvements in data acquisition, methods and tools.
- Further development of computational tools for deterministic plant assessment including coupled codes – progress towards multi-level and multi-physics computational capabilities.

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- Advanced safety assessment methodologies (identification and reduction of all uncertainties plus increase of their predictability, optimization of safety margins etc.).
- Support development of methods and tools to better insure complementarity of probabilistic and deterministic assessment, including integration of such methods
- Extended validation of deterministic computational codes and benchmarking of probabilistic assessment methods (including determination of missing experimental data).
- Improved understanding and modelling of internal events including fire and external hazards including grid disturbances. Improved methods to handle events with low probability / high uncertainties and transfer this to design specifications and to SAMG-strategies.
- Develop and apply tools and methods for upgrading of reactor safety systems to handle new safety demands, effective replacement of obsolete components and support of LTO.

Each of the major objectives from the list above is in relevant TA1 subarea elaborated into number of more specific challenges and topics – see below.

It is also worth to mention that composition and structure of TA1 gives to the R&D community a unique opportunity of long-term cooperation between experts from deterministic and probabilistic branches and thus utilise the synergies of such a tight cooperation. It is another indirect but important objective of TA1.

2.1.3 State of the Art

The R&D activities in the field of safety and risk assessment have relied, for a long-time upon wide support from international organisations and their programmes, such as the European Framework Research and Development Programmes (FP), the programs of the OECD working-groups and several others.

The main issues that have been addressed in these int'l programmes are related to the probabilistic and deterministic methods, the impact of external loads and hazards on the safety functions, the consequences of external electrical disturbances on the safety function and the advanced methodologies for safety assessment and their harmonisation.

For example: the OECD Working Group-risk, addressing risk assessment methodologies, the European FP6 and FP7 seeking for harmonisation among practices on PSA2 and addressing man-machine interface and ergonomic aspects.

As for the deterministic assessment of plant transients, several international and European programs have been organized lastly or are currently running – addressing either transient and accident phenomenology, computational tools development, collecting experimental data for codes validation and development of assessment methodologies.

The impact of external loads and hazards on the safety functions has been the subject of several national and international programs, as well.

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An overview of major international and regional R&D projects related to objectives of TA1 has been prepared during meetings and communication within TA1. The overview is divided according to TA1 subareas:

i. For PSA level 1 and 2, the reference programmes and projects are OECD WG-risk, NPSAG, SAFIR, VGB-PSA working group, APSA-network, ASAMPSA-2, MMOTION, and plant specific programs.
ii. Projects focused on deterministic assessment of plant transients: OECD WGAMA, OECD LOFC, OECD PKL, OECD ROSA, OECD SETH, OECD PRISME and PRISME II, NORTHNET, SAFIR, VGB, NURESIM, NURISP, NURESAFE, EUROSAFE, AER, OECD ISPs.
iii. Projects focused on impact of external loads and hazards on the safety functions: NORTHNET, SAFIR, NOG, VGB, ASAMPSA2-E, and NPSAG.
iv. Projects focused on impact of external electrical disturbances on the safety functions: OECD DIDELSYS, OECD ROBELSYS, BWR-club, NPSAG, IAEA (NS-G-1.8, D-NG-T-3.8).
v. Projects dealing with advanced safety assessment methodologies: OECD WGAMA (PROSIR), NORTHNET, SAFIR, EU-RTD-program (NURBIM, NURISP), OECD UAM, OECD BEMUSE, OECD SM2A, OECD PREMIUM.

The number of programmes lastly finished or currently running could be impressive; however, the high fidelity assessment of safety and risks needs a continuous effort, preparation of further steps, harmonization of activities and approaches. The continuous effort which will lead to reduction of the uncertainties and gaps in the existing methodologies, minimize limits and shortcomings of current computational tools, and enable appropriate validation of these methods and tools. And also, an effort that will lead to harmonised safety culture that among other takes into account the latest R&D results.

2.1.4 Challenges

To achieve the above-mentioned objectives, the work in TA1 and in whole R&D community working in the field of NPP safety and risk assessment should be focused on the following challenges:

Challenges in the field of PSA methods and application

- Improving the Probabilistic Safety Assessment - PSA - methods (e.g. accurately addressing the problem of system reliability and validation in actual operating conditions) and extending their application.
- Contributing to further common understanding and use of risk assessment techniques based on PSA. Special care is to be devoted to the evaluation of different kinds of dependency and human performance effects, and associated reliability data.

Deterministic assessment of plant transients

- Improvements in the field of deterministic assessment of plant transients and accidents, covering whole area of DBA and the extended design basis conditions.
- Progress towards multi-scale and multi-physics computational capabilities.

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- Extended validation of deterministic computational codes (including identification of missing experimental data).

Impact of internal and external loads and hazards on the safety functions

- Evaluating the impact of internal and external loads and hazards on the safety functions.
- Accurately addressing and accounting for the outcomes of the post-Fukushima evaluations (e.g., the effect of both single and multiple external events on safety function degradation is to be investigated and methodologies to assess the impact of internal and external loads and hazards on barriers and on structures, systems and components have to be either improved or improved. Appreciation of time (and feasibility) for recovery and the influence of non-safety systems on barrier strength have is to be improved.

Impact of external electrical disturbances on the safety functions

- Accurately accounting for and investigating the impact of external electrical disturbances on the plant safety functions.
- Focussing on the impact of electrical disturbances from the grid on plant safety systems and safety functions: the objective is to secure safety system performance.

Effects of Human errors and reliability evaluation

- Comprehensively addressing the effects of Human errors and reliability evaluation.
- Assessing the impact that human performance on reactor safety. Failure data on human reliability for use in risk assessments are have to be included in the investigation.

Advanced safety assessment methodologies

- Investigating and validating the advanced safety assessment methodologies.
- Integrating the deterministic and probabilistic safety assessments in order to better evaluate safety margins with best estimate methods, in particular determining the data, methods and knowledge needed to assess safety margins in components such as pipe/vessel and system (strength and weaknesses of NPPs).

Design of reactor safety systems

- Improving design and maintenance of safety systems to handle obsolescence of components and lack of spare-parts and up-grading them to face new demands.
- Improve design of reactor safety system to fulfil demands on defence in depth.
- Improve plant safety by using passive system concepts.
- Improved reactor design to cope with rare event and event with extreme uncertainty by combination of installed safety functions and by mobile functions steered by a philosophy for severe accident management.

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- Improved measurements to secure identification of specific failure modes.

Relations to pre-normative research

NUGENIA is a privileged frame to implement pre-normative research activities, because the Association's partners enjoy a wide portfolio of ongoing and future projects which propose and develop new technologies, new materials, new approaches and innovative technological solutions.

Several topics suitable for harmonisation have already been clearly identified. They mainly concern system design and safety practices, design, fabrication, test and operation of system and components, methodologies and practices for testing and validation. The standardization should start with sharing the state of knowledge among NUGENIA areas.

The safety margins evaluation approach is the subject of a common challenge. The actual activity in the field will be carried-out also in TA1, in subareas dealing with advanced safety assessment methodologies and deterministic assessment.

Moreover, NUGENIA's partners are directly or indirectly owners of a large amount of experimental results which can be effectively and proficiently analysed for pre-normative purposes to seek and propose new and more effective limitation and thresholds to onset parameters in the design. The topic is addressed in TA1 (STA1.2).

2.2 Sub Technical Areas (STA)

2.2.1 Data, Methods and Tools for Risk Assessment (STA 1.1)

2.2.1.1 Scope

Contributing to further development of common understanding and usage of risk assessment techniques based on probabilistic safety assessment (PSA). In the field of input data, special attention will be devoted to the evaluation of different kinds of dependency and human performance effects, and associated reliability data.

A PSA needs a large amount of information to be realized, which is the reason for the large number of points discussed under this heading.

2.2.1.2 State of the art

The safety of nuclear power plants is achieved by numerous methods and techniques. Probabilistic safety assessment (PSA) is a structured approach looking at how complex systems work together to ensure safety. PSA makes it possible to quantify risk and identify what could have the most impact on safety. The methods for performing PSA have been developing since the mid 70-ties and a state of the art PSA today typically covers internal initiating events as well as internal and external hazards, and it is performed for all operating states. PSA is also performed to address so called outside core events.

As probabilistic approaches have developed risk insights are being applied in areas where traditionally only deterministic approaches have been used. An example is in the in service inspection programmes. PSA is also applied to support different applications, for example risk monitors.

2.2.1.3 Challenge

For quantitative aspects, the following gaps should be covered:

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- Development of methodologies to quantify initiating event frequencies for low probability events, including external events and common cause failure (CCF) events
- Combination of events (including internal and external events)
- Methods used for establishing component failure rates with focus on components with low failure rates and also failure rates due to specific loads (such as loads due to fire, severe accident conditions etc.)
- Data and methods used to assess CCF inside specific system and trans-systems interactions at component level and at subcomponent level
- Development of methods for assessing human reliability and establishing a database with reference cases for supporting common risk assessments of human performance
- Methods to handle time dependent assessments in PSA
- Methods and data for quantifying the effects of aging on PSA outputs
- Methods and data for assessing failure frequencies of digital components
- Methods and data for performing fire PSA

For the PSA dedicated to source term issues (level 2 PSA) recommendations on the best strategies to couple or integrate level 1 and level 2 PSA should be done, and methodologies to assess shutdown states or external events should be developed.

Even if quite advanced PSA tools are well developed and extensively used, it requires also applying them correctly and in the most useful manner in specific risk informed decision management (RIDM) applications. If good methodology for development of “basic” PSA is necessary, the need of a large practice of well-developed and harmonized RIDM methodology is even more necessary.

2.2.1.4 Quantitative aspects of PSA

2.2.1.4.1 Selection and grouping of initiating events

Basis for selection and grouping of initiating events is particularly useful when quantifying initiating event frequencies for low probability events, including external events and CCI-events. In this sub-area, methods to combine the external event with the other initiated events are covered to develop a balanced PSA-study. It may have key impact on PSA results for most of PSA studies. Any decreasing of uncertainty in estimation of IE frequencies for low value frequencies may be important both for PSA results and for RIDM based on them. One important aspect is to cover the uncertainties in this analysis task, and the use of screening criteria. Another important issue is the linking of the PSA with SAR (safety case) events.

A further comment is the need for training and benchmarking of existing methods. May be we have enough methods but they need to be applied more properly?

R&D topics

- ✓ **Reassessment of External Events in View of the Fukushima Accident (NOG); the purpose is to initiate assessments within the field of external events in the light of the EU Stress Test Specifications.**

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2.2.1.4.2 Assessment of common cause failures

Data and methods used to assess common cause failures (CCF) should consider causes inside specific systems but also trans-systems interactions at component level and at sub-component level.

CCF analysis methodology was developed to relatively high degree in past, but there are still important (new) areas not covered, which may have key impact on PSA results (inter-system CCF, CCF for large groups of components etc.).

It may be difficult to go much further with CCF quantification. However, maintaining a high safety level requires a continuous war against dependencies – keeping control of existing dependencies and avoiding the introduction of new ones. Uncertainties are again an important aspect, and also during quantification to possibly get some credit for a strong dependency defence. Another important aspect is to continue work on identification of and defences against potential CCF attributable to several systems, since this is in most cases not included in our PSAs.

Thus, the topic is at least of medium importance. However, a more detailed description of the actual research approach would be needed.

R&D topics

- ✓ **Analysis of critical component types from a CCF point of view to:**
- ✓ **to keep control of existing dependencies and avoiding the introduction of new ones**
- ✓ **to identify of and to defend against potential CCF attributable to several systems**

2.2.1.4.3 Human reliability data

In this item, it hangs only about data for PSA: development of methods for assessing human reliability and establish database with reference cases for supporting common risk assessments of human performance.

2.2.1.4.4 Harmonized metrics for accident consequences

The objective of this work is to develop common definitions for consequence metrics for deterministic as well as probabilistic assessments like “core damage” or “large early release” suitable for PSA level 1, level 2 as well as level 3 results.

Recent work performed in projects related to safety goals and to develop harmonized methodologies for PSA- level 2 have indicated that there is a wish to use safety goals and to compare outputs from PSA-studies. These projects have also indicated that the detailed specification used for the END-state differ between different users. To support the use of safety goals it is important to get a common set of END-state definitions and to include a common view of the accuracy of different code to assess the end-state.

To broaden the safety goal it will be possible to develop safety goals on other parameters which is proposed in the on -going projects. The development of new metrics for safety goals applications is part of this issue.

R&D topics

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- ✓ **To develop best practice guidelines for the performance of Level-2 PSA methodologies with a view to harmonization at EU level and allowing a meaningful and practical uncertainty evaluation in a Level-2 PSA.**

The nuclear accident in Japan resulted from the combination of two correlated extreme external events (earthquake and tsunami). The consequences (flooding in particular) went beyond what was considered in the initial NPP design. Such situations can be identified using PSA methodology that complements the deterministic approach for beyond design accidents. If the performance of a Level 1-Level 2 PSA concludes that such a low probability event can lead to extreme consequences, the industry (system suppliers and utilities) or the Safety Authorities may take appropriate decisions to reinforce the defence in depth of the plant. The present topic aims at providing best practice guidelines for the identification of such situations with the help of Level 1-Level 2 PSA and for the definition of appropriate criteria for decision making in the European context.

2.2.1.4.5 Time dependant PSA

Methods to handle time dependent assessments in PSA- (Scenarios in which system demands are changed within the mission time or interrupted by manual actions)

For a base case PSA aiming at showing that the risk is below some target value, the need for handling of time dependencies is more limited than cases (applications) where the requirements on the degree of realism may be much larger.

Since the PSA methods that are most commonly used today are static in their nature it is important to look at the importance of this when it comes to PSA realism and the ability to demonstrate safety margins using PSA.

To develop and assess method in which realism in the PSA- studies are increased without going to dynamic PSA methods are requested from the end-user to increase the applicability of PSA-assessments.

The proposed R&D topics are considered as important practical steps towards deployment of advanced safety analysis and justification methods which combine the use of deterministic and probabilistic methods in industrial practice. The primary goals of the project are:

R&D topics

- ✓ **to develop further IDPSA methods and for joint application with PSA and DSA in practice of safety analysis;**
- ✓ **to assess advantages and present limitations of joint application of IDPSA with state-of-the-art PSA and DSA methods based on experience from a set of pilot realistic applications.**

The main outcomes of the R&D are to be:

New and improved IDPSA methods;
Recommendations and guidelines for joint applications of IDPSA, PSA and DSA;
Summaries of experience of addressing pilot realistic applications with IDPSA, PSA and DSA;

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That is to increase awareness of researchers, utilities and regulator communities about advantages and current limitations of the new approaches to safety analysis which tightly combine deterministic and probabilistic methods.

2.2.1.4.6 Deriving instantaneous risk from long term approaches

Risk monitoring needs to develop methods to use all the issues connected with the approaches used for transfer of “long term” (year) average risk in PSA model to instantaneous risk, where we work with mix of two categories of objects.

The acceptance of risk monitors and increased use of PSAs for assessing risks connected to single events asks for new methods for comparing instantaneous risk to the yearly risk level.

R&D topics

- ✓ **To develop risk assessment methodologies that enable a transfer of long term risk to instantaneous risk.**

2.2.1.4.7 Living PSA

This point describes the needs of methods and data for quantifying living PSA including the effects of aging on PSA-outputs- (connection to the APSA-network and its work).

PSA rarely explicitly models ageing effects on equipment reliability. In addition, PSAs traditionally ignore some components (e.g. cables, structures) as having low failure probabilities, but they may have increasing contribution due to ageing effects. All of these factors tend to limit PSAs validity in time.

Aging is also important to consider for component reliability/availability and with regard to dependencies – a set of components having the same age being exposed to similar stresses during the same time period will increase the risk if no action is taken. PSA results can be used to focus control/surveillance/maintenance activities on those most important components where aging effects will increase risk most.

PSA results are only good for a limited amount of time, thus the drive for a living PSA. Part of the living PSA approach is a regular update of reliability data. Ageing will manifest in increased failure rates

Each plant has strategies for replacement of component coming close to end- of life.

Old plants have a mixed of old components and new components. Replacements and modifications in a plant also introduce new failures.

Failure data do not distinguish among new and old components.

R&D topics

- ✓ **Development of better realism in living PSA-studies taking to the account:**
- ✓ **Identification of ageing signs on the performance of the systems, structures and components (SSC),**
- ✓ **Addressing ageing and maintenance effects in component failure models,**

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- ✓ **Approach for selection of SSC which are ageing sensitive and important from safety point of view,**
- ✓ **Practical methods for analysing component and system reliability data, with focus on the identification and estimation of ageing reliability parameters,**
- ✓ **State of the art of existing NPP component reliability data collection systems which aimed to elaborate aged components reliability parameters,**
- ✓ **Methods and approaches for Advanced Time-dependent Reliability Data Analysis (including ageing trend analysis),**
- ✓ **Estimation of ageing impact on common cause failure (CCF) coupling factors and CCF preventive means,**
- ✓ **Assessment of structural integrity assessment and lifetime prediction tools,**
- ✓ **Approaches for the consideration of ageing effects within existing probabilistic safety assessment model.**
- ✓ **Improvement of reliability and maintenance data collection system to fit with specific requirements of age and maintenance dependent reliability models,**
- ✓ **Development of methods (data, parameters, models) to treat ageing common cause failures in plant reliability models,**
- ✓ **Approaches (including recommendations for data, model requirements) to treat aged passive components in system reliability models,**
- ✓ **PSA code improvement and development of appropriate interfaces to improve effectiveness of treatment for ageing reliability models,**
- ✓ **Improvement of risk-based methodologies in order to optimize inspection and maintenance of NPP systems,**
- ✓ **Prediction of plant safety level and development of optimized approaches to PSR, PLiM, LTO,**
- ✓ **Incorporation of Ageing Effect into PSA – an R&D project that has been proposed to SSM.**

2.2.1.4.8 Failure frequencies of digital components

Methods and data for assessing failure frequencies of digital components are necessary. Digital hardware and software are becoming more and more important as it is used in more and more extent for the reactor protection system (RPS). Failure rates of digital systems are a relevant issue. Unfortunately, the reliability is obviously limited by CCF in the digital software (in a well-designed system).

The software CCF issue has eluded all efforts for a solution and will continue to do so. Thus, there are low prospects for a successful completion of the work. Particularly for current I&C technologies, there are many issues with significant risk impact, which have not been and should be solved.

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Nevertheless, the IEC standard IEC 61508 (Functional safety of electrical/ electronic/ programmable electronic safety related systems) describes principles for analysis of programmable systems. One important concept is the definition of Safety Integrity Level, SIL. The objective with the R&D topics is to evaluate the feasibility of applying principles and methods according to IEC 61508, in probabilistic analysis of programmable safety instrumented systems used in NPPs.

2.2.1.4.9 Methodology and data for performing fire PSA

Fire PSAs are performed with varying objectives and thus varying degree of realism resulting in difficulties to compare results between studies, and also to get a good understanding of real contribution from the internal hazards compared to internal events.

What is needed though are common methods for determination of fire initiating event frequencies that are simple to use and also justifiable. Also, in fire PSAs, there are a lot of assumptions regarding what the consequences of fire, or exposure to heat, will have on components. These assumptions often lead to a conservative approach which is not comparable to internal events PSA. Therefore, common harmonized methods to assess the impact of fire on different components in different compartments, including fire spreading between compartments would be beneficial to the area of fire PSA.

R&D topics

- ✓ **To develop common methods for determination of fire initiating event frequencies that are simple to use and also justifiable Parameters that are of interest to be further investigated in this area are:**
- ✓ **What parameters are of importance when it comes to understand the effect of a fire on components in a compartment?**
- ✓ **What kind of simplifications and assumptions are justified to make in order to get a more realistic and comprehensive fire PSA without ending up in a large volume of deterministic fire simulations to be done?**
- ✓ **How should sensitivity and uncertainty analysis be performed in a fire PSA?**
- ✓ **How to examine and analyse potential cliff-edge effects in a fire PSA?**

2.2.1.4.10 Methodology and data for performing seismic PSA

Many subtopics can be identified below this heading (seismic hazard curves, plant response spectra, SSC fragility curves, etc.). We also would like to stress the importance of linking to the deterministic safety analysis.

Incorporation of seismic hazards differs from a regulatory perspective from country to country, in some countries there are currently no requirements on including seismic hazards in the PSA (e.g. Sweden) but in other countries that are equally seismic active (e.g. Finland) there are requirements to incorporate seismic hazards in the PSA. This inconsistency will most likely disappear in the future. For countries with a high seismic activity detailed methods must of course exist (and are most likely already in place to a large extent) but for countries/regions with less seismic activity some reasonable, and simplified, approach to incorporate seismic effects in the PSA should be developed in order to establish the risk contribution from seismic hazards for NPPs in these regions.

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There is strong connection of this topic with Fukushima events and European stress-tests.

R&D topics

- ✓ **Development of methodology for seismic PSA for deriving hazard and fragility curves and establishment of a good practice for what to include in a seismic PSA, including dependencies, types of components to model, etc.**

2.2.1.4.11 Modelling functional dependencies

Modelling techniques for functional dependencies in electrical and safety instrumented systems.

The functional dependencies in I&C-system designed with analogue or digital techniques or a combination of these can be modelled with different details both concerning physical dependencies but also to the used sub-components and its testing schemes. In some cases black boxes are accepted in other cases the detailed modelling are requested.

There is a need to develop better understanding of the effects of using different modelling strategies on these issues.

2.2.1.5 General aspects on PSA results

2.2.1.5.1 Coupling between level 1 and level 2 PSA

The methodology for coupling level 1 and level 2 PSA or to develop integrated level1/level2 PSA has an effect on the overall result. Assessing the effects of using different strategies between level 1 and level 2 studies is important and it should help deriving recommendations on how to perform coupling. This work can be started from the output from the ASAMPSA2 project.

The different strategies are related to the choice of tools. Some methods apply tools where the level 2 part is separated in its own database. PSA level 1 results then have to be calculated first and be the input to the level 2 part. Different challenges exist with each approach. So far we have concluded (e.g. within ASAMPSA2) that they can be solved with both approaches. This is very important point in development of really integrated plant PSA model. Since many organisations are in the process of gradual development of integrated model, the topic is very interesting. International cooperation and know-how exchange in this area would be very useful. There is still a need for determine the negative aspect of using the different methods. Effects on LERF /LRF and in different applications by using the different methods shall be specified.

2.2.1.5.2 Considering shut down states and aggressions in L2 PSAs

Basic methods for shut down state and external event in PSA-level-2 are the same as for full-power. There is a lack of performed PSA covering these scenarios. The Fukushima event points out the need for assessing those scenarios in PSA-lvel-2.

There are some issues that are different from full power assessments, e.g. definition of end states, and that phenomenon will differ from those during full power but in general the procedure is the same as in a PSA for any plant – with the same ingredients – Initiating events, system, scenarios, data, human factors, quantification, result presentation and interpretation.

PSA level 2 for external events may be good to have as separate topic but as mentioned above the overall methodology is the same as for level 1 PSA. Understanding and mastering the uncertainties

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related to external events is prime importance as well it is the impact they have on the level 2 assessment.

R&D topics

- ✓ **Development of methodology for assessment of shutdown states, and external events in PSA level 2 and risks related to spent fuel pools.**

2.2.1.5.3 Setting up safety goals

Developing harmonized ways to specify safety goals for European reactors is important to define terms of reference for the PSAs, for instance specify important end-state to be used in setting safety goal for different purposes that are useful in the communication with deterministic regulations and with the public.

The usefulness of PSA and the acceptance of PSA in safety assessment will be increased if the coupling with other ways to assess safety and safety margins can be demonstrated. Different methods to develop this are discussed in projects related to Safety goals. This topic includes projects that assess new ways to establish such coupling. The idea of cross-connection of PSA end states and INES scale levels is worth mentioning, but does not belong to the topic related to PSA development; rather it leads to a kind of PSA application.

R&D topics

- ✓ **Achieving a suitable definition of safety goals in agreement to the WENRA safety objectives.**

2.2.1.5.4 Methodologies for level 3 - PSA

Partly based on the nuclear accident at Fukushima, and partly on current international interest in PSA Level 3, it is reasonable to believe that development of utility specific Level 3 PSA's will be needed and/or required in the near future. Also, it is anticipated that looking more into the PSA level 3 issues will further increase the quality of level 1 and 2 PSA.

Risk criteria used in PSA are basically based on some kind of expectation that if we can show that the frequencies for different consequences are smaller than a specific target value the risk is acceptable. Acceptable risk here means that the possible off-site consequences can be argued to be acceptable.

Therefore, it is the off-site consequence criteria that are most closely related to the primary safety goal, related to off-site health effects or environmental effects. In terms of application, a PSA level 3 is required to address off-site consequence criteria [Safety Goal project].

There are many parameters involved in the definition an unacceptable release, the most important ones being the time, the amount and the composition of the release. The underlying reason for the complexity of the release definition is largely the fact that it constitutes the link between the PSA Level 2 results and an indirect attempt to assess health effects from the release. However, such consequence issues are basically addressed in PSA Level 3, and can only be fully covered in such an analysis [WG Risk Task (2006)-2].

Most of the risk criteria that we are dealing with in PSA Level 1 and Level 2 have a background in PSA Level 3, nevertheless very few PSA Level 3 analysis are being performed today and very few formal

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requirements exist on PSA Level 3. However, when it comes to possible new-builds it is not unlikely that some kind of PSA Level 3 requirements will be made.

Development of peer review standards ANS/ASME 58.24 (PSA level 2) and ANS/ASME 58.25 (PSA level 3) is anticipated to last 2-3 years until these standards will be published.

A major findingfromthisworkwastheinconsistencybetweentheinformationprovided and the methods employed for a Level 2 PSA towards performing a Level 3 PSA. Therefore, continuation of this work will first focus on best practices in Level 2 PSA procedures, calculations, and analyses outputs to facilitate down-stream Level 3 PSA analyses.

R&D topics

- ✓ **Methodology for Level 3 PSA development taking into account:**
- ✓ **National and International Guidelines and Standards for Level 3 PSA Methodology**
- ✓ **Level 3 PSA Applications**
- ✓ **National and International Regulatory Requirements for Level 3 PSA**
- ✓ **Regulatory Use of Level 3 PSA**
- ✓ **Level 3 PSA Methodology: Challenges in the Light of Lessons Learned from the Fukushima Daiichi Nuclear Accident**
- ✓ **Level 3 PSA and Safety Goals for Nuclear Installations**

2.2.1.5.5 Benchmarking among existing PSA studies

Benchmarking of existing PSA-studies is useful to support comparability of PSA studies and to support use of safety goals in plant management.

Harmonization of methodological approaches used in PSA project is key issue for increasing of credibility of PSA results. Benchmarking is important to learn more and is a good tool for understanding of differences. Not necessarily restricted to the above scope indicated by the topic title.

2.2.1.5.6 Safety goals in safety assessments

Develop guidance is necessary to use safety goals in reactor safety assessments; it should be tested on case studies and benchmarks. This involves usage of output from projects on validity of safety goals and the reference cases in INSAG 25.

Further effort on PSA-deduced indicators and PSA-based safety goals is certainly needed. Guidance, course material, examples will be developed.

2.2.1.5.7 Risk related to spent fuel pool

All risks should usually be assessed according to country specific or international requirements and guidance. When it comes to PSA for spent fuel pools (SFP) the normal criteria for core damage, accident sequences (including mission time) etc. may not apply and it is therefore of importance that common methods are developed to assess the SFP risk. Also, depending on the barriers available the SFP PSA may be more related to a level 3 PSA depending on the location of the SFP.

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2.2.1.6 Best practice for PSA application

Even if we use well developed and quite advanced PSA tools and are able to get credible and useful results on the base of our PSA model, it does not mean that we are able to apply them correctly and in the most useful manner in concrete RIDM (Risk informed decision making) applications. More developed and common used RIDM methodology will support a broader acceptance of risk assessments. We suppose, we will try to propose cooperation to the utilities and to get some resources for our projects from them. In our opinion, almost all issues defined for this sub-area up to now may seem to be a bit academicals for people responsible for NPP operation. PSA applications are much closer to “their” reality and problems and we should take it into consideration.

2.2.1.6.1 Safety classification for systems

Development of safety classification methodologies can be based on risk insights and practical use of the IAEA-guidance on safety classification.

The safety classification controls in many ways design and maintenance criteria for the SSCs. In many cases the safety classification is made on deterministic rules only. The PSA can be used as a complementary method to justify, and possibly change, the classification of SSCs. A PSA application could result, for instance, in a comparative study for one category of nuclear reactors in different places.

R&D topics

- ✓ **Development of methodology on how to incorporate PSA results into Safety Classification process.**

2.2.1.7 Priority ranking

The priorities ranking of subarea STA-1.1 organized and evaluated in 2013 resulted in the following order of top challenges/topics:

1) Failure frequencies of digital components
2) Risk related to spent fuel pool
3) Methodology and data for performing seismic PSA

2.2.2 Deterministic Assessment of Plant Transients (STA 1.2)

2.2.2.1 Scope

This sub-area includes action/projects in thermal hydraulic (TH) assessment of normal and abnormal transients used in the deterministic assessment that are used for the safety justification of the plant safety performance up to core damage.

Improving the deterministic assessment of plant transients with conservative assumptions and extended coverage of validation and extending consensus on methodologies for transients' evaluation.

2.2.2.2 State of the art

A computer codes for advanced NPP thermal hydraulic (TH), neutronics etc. prediction of NPP behaviour – at system or local level - are developed and assessed in frame of several major

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international and national programmes (EU projects NURESIM, NURISP, NURESAFE; OECD/NEA projects PKL, PKL2, PKL3; US NRC int'l project CAMP; US DOE project IRUG; US project CASL etc.) and the Nordic research within NORTHNET.

The state-of-the-art computer codes are developed both as "individual" tools and in coupling manner. Latest trends are the multi-physics and multi-scale computational tools. But also the development and assessment of individual computer codes are still important task.

An overview of about 40 major and relevant international, regional and national projects, both underway and completed, was presented at the Budapest Plenary Meeting 3/2013. The overview is now available also at the NUGENIA's web workspace.

2.2.2.3 Challenges

In the assessment of plant transients and accidents, new challenges are arising but recurrent studies still need large efforts to be performed with effective results. Indeed, plant behaviour is the result of complex multidimensional physical phenomena; moreover, phenomena are tightly coupled.

2.2.2.4 Improved TH evaluation for the existing plants

Improved capabilities of TH evaluation of existing plants bring benefits in several areas – safety of NPP operation, economics, life time prolongation (LTO). To fulfil the objective of improved TH evaluation for existing plants, the main challenges in thermal-hydraulics are:

- A better understanding and modelling of the multi-dimensional phenomena, in particular in vessels and pools,
- A better understanding and modelling of the multiphase (steam/water, non-condensable gases…) phenomena,
- The interaction with reactor physics – neutronics - (in particular in reactivity transients), mechanic (fluid-structure interaction in steam generator for example), and thermo-mechanic,
- The experimental validation,
- The uncertainties evaluation.
- Some specific topics are given hereafter:
- Stratification in pools and vessels, in particular for BWR or AP condensation pools when steam flow is low
- Mixing in pools and vessels, in particular at low flow rates, including vessel pressurized thermal shock and boron mixing in reactor vessels
- Better predicting of the margins of instability in BWR-cores, in particular coupling 3D thermal-hydraulics with neutronics codes
- 3D flows in the reactor pressure vessel. (BWR/PWR/VVER)
- Assessing effects of non-condensable gases in reactor coolant system for scenarios with gas intrusion

R&D topics

- ✓ **Progress in understanding and modelling capabilities in areas of multi-dimensional multi-phase and multi-component flow, heat transfer, interaction with neutronics are foreseen.**

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- ✓ **Capability for Improved prediction of uncertainties and their propagation, more exact quantification of core margins are requested.**

2.2.2.4.1 Stratification in pools and vessels

For stratification in pool and vessel it is important to understand the phenomena and to validate and improve correlations or models.

There is a lack of experiments in real facilities for these configurations (pool design and blow-down pipes pipe levels below water).

Conservative assumptions have so far been used in assessing safety margins of residual heat removal capacity when realistic modelling cannot be performed as there is a lack of validated data. In assessing new scenarios including power up-rate the existing margins are no longer enough.

Indeed, e.g. Lappeenranta University experiments are not enough to verify real conditions.

The NORTHNET project (financed by the Nordic utilities and vendor) are on-going but have not yet developed all needed data for understanding the phenomena related to stratification in condensation pool.

The correlations/models developed and validated can be later implemented in different codes, which can be a separated task or integrated in this topic. This shall include stratification and break up of stratification depending of the geometry of the spargers and the positions of the sources effecting the stratification. The work shall include validations of new codes as the GOTHIC code instead of COPTA and RELAP 5.

2.2.2.4.2 Mixing in vessel at low flow rates

For flow mixing in vessel at low flow rates, a need remains to understand the phenomena and to develop the correlations/models.

In scenarios where the circulation pumps stops, the flows in the reactor vessel is driven by the heat from the core – in a natural circulation mode without any momentum from pumps. Depending on water level in the vessel, the residual heat, temperature of water and on dilutives as boron different flow path will be developed in the core, under and over the core.

Existing codes are not capable to assess the flow patterns in these scenarios with high accuracy. One event in Oskarshamn 3 during 2003 with cold water coming from the control rod cooling system into the bottom of the vessel have indicated the problem of assessing these situations. In several cases when boron shall be used, similar conditions are possible (BWR/PWR/VVER). Within this topic also issues related to natural circulation can be covered.

R&D topics

- ✓ **The development of BDT simulation methodology adopting up-to-date and accurate CFD models is requested. The CFD models are benchmarked against high resolution experimental data.**

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2.2.2.4.3 Fuel rods dryout and DNB

This topic relates to understanding, modelling and validation of 2-phase flow codes for assessing dry-out effects on fuel rods. This also includes benchmarking against real events which been close to dry-out or superseded dry-out criteria. (Primary BWR but also BWR/PWR/VVER)

Safety and efficiency aspects of nuclear power plants with light water reactors are very much related to thermal-hydraulic processes in nuclear fuel assemblies. In particular coolant flow and enthalpy distribution in assembly sub channels have significant influence on margins to Critical Heat Flux (CHF). Void fraction and local power distribution are coupled to each other due to neutronics feedbacks.

Hydrodynamics stability of two-phase flow through fuel assemblies is a complex function of flow, void, and power distribution. Clearly, accurate prediction of flow, enthalpy and void fraction distribution in fuel assemblies is very important for safe and economical operation of LWRs in general and for prediction of occurrence of CHF in particular.

However, models able to predict the above mentioned parameters with satisfactorily high accuracy - and in particular the occurrence of CHF - still do not exist. Currently used models are based on one-dimensional or a sub-channel approach which heavily depends on correlation-type closure relationships. It is obvious that to meet the requirements of accuracy and reliability of new models, the mechanistic approach has to be adopted. In such approach, the models are based on the fundamental conservation principles and – that is important in the fuel assembly context – they are able to take into account multi-dimensional effects in complex geometry.

This topic relates to major steps that have to be undertaken to develop fully mechanistic models of two-phase flow and heat transfer in fuel assemblies. Such models will have the capabilities to predict heat transfer and phase distribution in fuel assemblies under normal and abnormal conditions.

R&D topics

- ✓ **Mechanistic modelling of two-phase flow and heat transfer development for improved CHF prediction will be supported. Experimental study on Micro-Hydrodynamics of Flow Boiling and Mechanism of CHF are important to support reduction of uncertainties.**

2.2.2.4.4 Scaling effects in validation of TH system codes

In many cases it is not possible or economical feasible to perform test in a full scale mock-up. To be able to transfer knowledge from down-scaled experiments to the real case there is a need for understanding who to correct use the data from the down-scaled experimental set-up. It is of importance that there is a common understanding of the scaling effects on the results. For the utilities it will be of importance to be able to perform more tests in down-scaled mock-ups to reduce cost for experiments. Recent assessments in Nordic research have indicated difficulties to transfer data from downscaled mock-ups to real cases concerning stratification issues.

This topic relates to developing of common understanding on when and how scaling can be performed.

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2.2.2.4.5 Predict the margins to instability in BWR cores

It is well known that BWRs can experience unstable conditions during start-up, i.e. at reduced core flow and relatively high power level, as well as during normal operating conditions in case of an abnormal event (equipment malfunction). Calculations can be performed via adequate coupled neutronics - thermal hydraulic codes to verify conditions under which the reactor becomes un-stable. If the margin to instability is not large enough a new core loading should be designed.

Despite existing knowledge and precautions many instability events have occurred in the past in BWR worldwide. The occurrence of these instability events might be attributed to two main deficiencies in the actual way to analyse and predict the stability.

The first one is simply related to the lack of proper understanding of some of the key phenomena. The second one is related to the inadequacy of the actual system codes to reproduce complicated instability patterns.

The current simulation platform has shortcomings for predicting BWR instabilities e.g. in terms of their ability to model transient two-phase flow and in terms of their numerical fidelity. These short comings require by themselves a significant coordinated effort aimed at accurately modelling transient two-phase flow.

Three types of instabilities might occur in BWRs: global or in-phase oscillations, regional or out-of-phase oscillations and finally a pure Density Wave Oscillation.

The aim with this topic is first to improve the knowledge and understanding of the key physical phenomena and the secondly to develop the corresponding high-fidelity numerical capabilities.

R&D topics

- ✓ **Development of a low-order model to investigate global/regional/local oscillations in BWRs and study of new stability indicators is suggested.**

2.2.2.4.6 Asymmetric flows in the reactor pressure vessel

Most scenarios coupled to reactor pressure vessel or containment is failures that introduce a disturbance in one minor sector of the vessel or the containment. By having disturbances in one sector of the large volume introduces asymmetric flows and loads in the volume. In traditionally safety assessment it is still assumed that the flows and loads are symmetric in the volume. In some cases this will give minor uncertainties in other it will introduce large failures in the assessment.

It is very important understanding and mastering the consequences of assumptions postulating symmetry conditions (of any kind and degree) of systems where it is not the case, and are introduced when an asymmetric behaviour can be likely.

By CFD it is possible to assess asymmetric behaviour in small volumes but not in large volumes like RPV or Containment.

This topic will cover issues related to better understand the effects of asymmetric flows and loads as well as understand the degree of failure that are introduced by using symmetric assumptions.

R&D topics

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- ✓ **Development of a BDT simulation methodology using state-of-the-art accurate CFD models is to be claimed. The CFD models are benchmarked against high resolution experimental data.**

2.2.2.4.7 Safety assessment through benchmarking and experiments

There is a need for increased validation of existing codes (for example against SET and ITF) and to benchmark them with comparison of methodologies for Safety Assessments

This topics concern increased benchmarking of used codes and the methodologies used by different organizations to perform thermo hydraulic assessments related to reactor safety parameters. To support a common understanding of safety and safety margin there is a need to perform benchmarking against real cases and against outputs from research program. This will also support understand differences in methodologies used in different countries.

There is a continual ongoing development of methodologies within different organizations and in different countries. To support a common view of safety and safety margin assessment there is a need for supporting benchmarking of the developed methodologies.

Benchmarking is interesting for certain areas. One example is methodology for analysis of oxygen and hydrogen levels in the containment. Recombination requirements are verified with a completely different method in Sweden compared to other countries.

2.2.2.4.8 Effects of gas intrusion or release in reactor coolant system

Assessing effects of non-condensable gases in piping for scenarios with risk for gas intrusion has to be improved.

With an improper system design with failed operational procedures or maintenance procedure, a system may not be able to perform its function in the specified range it is designed for.

Concerning system containing water or fluids the risk holds that system may get to much air into its piping for the pump to be able to work in an appropriate way.

Gas and air can enter piping system in specific scenarios as blow-down scenarios; it can enter at service of systems. Gas and air can enter when water level is too low and vortex is developed. In US large programs have been developed to specify operational demands to secure that inoperability due to gas and air in process system will not occur. Similar program have not been developed in Europe.

This topic includes all aspects related to knowledge and methods that reduce the risk for having to high concentration of gas and air in process system.

R&D topics

- ✓ **Progress in understanding of multi-phase and multi-component flow and effect of non-condensable gas on heat transfer and RCS behaviour are requested.**

2.2.2.4.9 Design and evaluation of passive safety systems

The major challenge to a generalized adoption of passive systems for safety purposes is the achievement of a convenient and exhaustive full scale demonstration of their reliability in transient and accident conditions.

R&D topics

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- ✓ **To provide evidence of the passive system reliability despite the approximations and assumptions in the validation experiments and to clear the way to extrapolation will reduce uncertainties. .Increased knowledge and data on passive systems reliability will reduce uncertainties in risk assessment too.**

2.2.2.4.10 Passive systems behaviour

Development and validation of computer codes adopted to design passive systems (BWR, PWR, VVER), support their assessment and the safety demonstration of NPPs making use of them still need improvement and consolidation. The topic includes better understanding of the function of system driven by gravity, weight and pressure differences. Passive systems are based on these momentum sources. There is a demand to introduce more of these in existing plants and into new build plants. Existing codes are not verified and validated for these conditions and there is a need for developing more data on different scenarios that are based on the passive concept.

It will be of importance to understand how flow rates in such systems are depending on different geometrical and material data.

2.2.2.4.11 Assessment of PAR (PWR, VVER)

VTT has an on-going project within PAR (passive autocatalytic recombiner) where they have done MELCOR analyses with unexpected results.

Systems are during the implementation phase and there is too less experience about failure modes and mechanisms.

This topic relates to developing an increased knowledge of the function of PAR under different conditions.

2.2.2.4.12 Reliability of accumulator injection termination

Isolation of ECCS accumulators after injection is ensured either by passive float valves and/or by motor driven isolation valves. If the isolation fails, massive nitrogen intrusion in reactor coolant system follows, effecting behaviour of RCS and operator actions (EOP).

Effectiveness and reliability of accumulators were investigated in several mostly national projects. Lastly the VTT initiated program on reliability of accumulators float valves.

2.2.2.4.13 Modelling of pressure suppression systems

Pressure suppression systems – either of bubble condenser type or ice-condenser type – are important part of many containments and confinements. Their performance in accidents with large leaks of mass and energy from reactor coolant system (LOCA, MSLB etc.) of PWR or BWR is crucial part of NPP safety.

Modelling of pressure suppression system is an important task, which was studied in several int’l and national projects, but is still a challenge for containment codes.

Further effort and progress in modelling capabilities would be very useful for safety analyses and correct predictions of respond of pressure suppressions system and the whole containment. These issues have connections with issue 1.2.1.1. Knowledge of the effects of using different kind of spargers or without such device is requested. Data covering a large spectrum of temperature ranges in the condensation pool are important to reduce uncertainties in safety assessments.

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2.2.2.5 Computer codes coupling

Couplings such as neutronics and thermal-hydraulics have to be developed, for example for re-criticality scenarios in which several control rods partially scram (BWR). Both multi-physics and multi-scale couplings are covered in the topics below.

R&D topics

- ✓ **To provide computer tools for multi-scale and multi-physics coupling resulting in improved and more complex modelling of reactor system phenomena.**

2.2.2.5.1 Coupling between reactor coolant system and containment

The possibility to have combined effects in reactor vessel and in containment exists. Severe accident codes do combined analysis but through simplified methods. Many system T/H codes can model and simulate both vessel and containment at the same time but in most cases by using simplified methods.

However, new system logic means that all barriers meet a larger challenge than before. One example is that the conditions in the containment no longer are limited by LOCA cases as they were historically.

The higher requirements imposed on the vessel side when using codes like POLCA-T lead to also higher requirements on the containment analysis. One example is ATWS cases where it is uncertain when II isolation will occur, and this timing is very important for the scenario.

Several scenarios related to phenomena in reactor vessel are depending on the parameters related to the condition in the containment e.g. temperatures and pressures that initiates safety logics, effects on level measurements that effect the function of level measurement system. Water level in containment that affects depressurisation of the vessel

In other cases the condition in the reactor vessel effect the containment scenarios as sub cooling of the water, water levels that affects start of different safety logics.

R&D topics

- ✓ **With the aim to reduce uncertainties in safety assessment and to develop better procedures there is a need to develop codes that integrate the phenomena in the reactor vessel with the phenomena in containment.**

2.2.2.5.2 Coupling between system TH codes and CFD

In general due to the multi-dimensional nature of the case, the complexity of the reactor geometry and the different phenomena encountered, different modelling tools have to be used. Each of these tools has a specific range of application and domain of validity. For instance, coupling a CFD tool for a detailed description of a specific component of the reactor pressure vessel to a system code for the remaining primary system would allow capturing the detailed information lacking in the system codes. Coupling different generic tools with each other offers the possibilities of using each of these cods strictly in its domain of validity. Even though this coupling might appear appealing many obstacles render it problematic. It needs to be emphasized that a universal algorithm for code coupling is an ill-posed objective. Instead we shall expect a set of recommendations for which pairs of codes and embedded schemes can be used in analysis of selected types of scenarios.

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This topic shall address development of methodologies to be used for code coupling.

R&D topics

- ✓ **Investigation of efficient and reliable numerical algorithms for coupled reactor calculations**

2.2.2.5.3 Coupling between TH and neutronics codes

One of the "traditional" coupling tasks – connection of 1D/2D/3D thermal hydraulic model with 3D neutron kinetics model – is required for number of safety analyses of initiating events with substantial change of core neutron flux and power shape and with strong reactivity feedbacks from TH to neutronics.

Multi-physics character of this coupling together with different type of basic equations (TH codes are based among other on simplified Navier-Stokes equation which has got hyperbolic character, whereas 3D diffusion and transport neutronics codes originates in Boltzmann transport equation which has got elliptical character) bring difficulties also in numerical solution of the problem.

This topic shall address development of methodologies to be used for code coupling.

R&D topics

- ✓ **Investigation of efficient and reliable numerical algorithms for coupled reactor calculations**

2.2.2.5.4 Re-criticality scenarios in BWR

Develop and validate methods to understand re-criticality scenarios in which several control rods fails resulting in an event which is partly scrammed can be of use in the BWRs.

To address the consequences of incidental scenarios with control failing rod drop evenly or unevenly distributed in the core it is worth performing 3D –calculations to evaluate the power switch, the core, residual heat, the void, steam, water level in different zone of the core from zero to full power. It is important to address such scenarios to define system requirements for PSA and when preparing Emergency Operating Procedures (EOP).

It is also important investigating the likelihood of cliff-edge effects in presence of specific failure combinations.

By work performed within NPSAG-group such assessments have been performed using POLCA –T and a combination of S3K and RELAP.

To develop a common view on these scenarios and the uncertainties in the results there is a need for having broader benchmarking of similar assessments and also on code developments to cover such scenarios.

2.2.2.6 Containment behaviour

Particular phenomena need to be better modelled:

- ➢ Non condensable gas flows in the containment with and without spraying
- ➢ Heat transfer in the gas phase of the containment including the interaction with walls and pipes

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- Leak rates through containment up to containment break
- Clogging phenomena in strainers and fuel and associated assessment methodology
- Fire and gas explosion simulation methods and applications to reactor safety

R&D topics

- ✓ **To improve understanding and provide relevant computational capabilities for prediction of containment multi-phase and multi-component flow, heat transfer and other relevant physical phenomena.**

2.2.2.7 Containment thermal hydraulics

Understanding, modelling and validation of codes is needed for assessing phenomena on heat transfer in the gas phase of the containment including phenomena related to oxygen and hydrogen gas flows in the containment with and without spraying. Effects of interactions with containment walls and piping shall be better understood (BWR/PWR/VVER).

There is a need for better prediction of the phenomena in the gas phase of containment to understand the composition of gases in the containment during different phases of an accident. This will be of importance to steer containment venting and to specify how to measure and assess measured data from containment.

2.2.2.7.1 Containment failure under high pressure and temperature

Develop and validate codes that can assess leak rates through containment at different pressure and temperature up to major containment break is a useful topic.

To assess leakage to the environment in deterministic assessment as well as in PSA- level 2 it is important to understand which factors affect the leak rate through the containment walls and through penetrations in the wall (as door, hatches).

Factors that will affect the leak rate are temperature, pressure in containment and the chemistry of the water.

The industry needs better qualified data on the leak rate dependency of these factors.

Such data are important to assess scenarios in which these parameters reach extreme values and for developing of strategy for containment venting

This topic covers all issues that can reduce the uncertainties in leak-rate extreme temperature, pressure and water chemistry in containment.

2.2.2.7.2 Clogging on strainers and fuels

Understanding of clogging phenomena on strainers and fuel is necessary for safety purposes. Developing of a European standard on methods and data to use in assessing clogging effects on strainers and fuel are therefore useful for all reactor kinds.

Since the clogging of strainers in Barsebäck in the early nineties a lot of research and development have been done in Europe, US, Japan. These actions involve issues such as:

- Effects of insulation (aged and new) material on strainer,
- Strainer design,

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- Insulation materials,
- Fastening methods for insulation,
- Effects of paint, concrete and other dirt's on strainer,
- Effect on fuel from insulation materials.

There is no standard method developed for performing safety assessment or a common accepted table of which effects insulation have on strainer and when a strainer is overloaded.

Such assessment will involve data covering all steps from releases of insulation, transport of insulation, sedimentation, design and flows through the strainers and for the fuel aspect also transport within the piping to the reactor vessel.

R&D topics

- ✓ **To develop common data and methodologies for assessing risk of clogging of strainers and fuel.**

2.2.2.8 Fluid structure interactions

Special models need progresses:

- Turbulent flows and its effects on component aging
- Fluid-structure interaction in steam-generator
- Water hammer assessment
- Coupling between CFD and system codes
- Heat transfer along piping and vessel walls during turbulent flows

R&D topics

- ✓ **To provide better understanding and computational tools for prediction of fluid-structure interactions (FSI) resulting in more effective support of plants operation and components reliability.**

2.2.2.8.1 Structure wear due to fluid induced vibrations

Power plants are upgrading to higher power and also doing modernisation of plants with new components. In these cases the flow rates are changed which can introduce vibrations and also introduce flow accelerated corrosion.

Plant designer and Vendor designer have to understand in which range flow rates can change without introducing negative effect in the plant. Improper design will result in costly stop and repairs.

Existing knowledge are mostly based on "proven design" and not on models and calculations. There is a need to develop better knowledge that can be transferred to common accepted numerical models that almost have to be developed from scratch and above that experimental validation data should be foreseen

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2.2.2.8.2 Water hammer assessment

There exists still no common accepted methodology/code to assess water hammer loads. There is a need for further develop and validate CFD-codes for different kind of geometries to reduce the uncertainties in water hammer assessment.

This topic relates to different activities that support development of common accepted methodologies for water hammer assessment

2.2.2.8.3 Better understanding of heat transfer into piping and vessel walls during turbulent flows. Ways to predict near wall effects of the flows in BWR/PWR/VVER

Better understanding of heat transfer into piping and vessel walls during turbulent flows is needed for all reactor types. Several problems related to heat transfer into piping and vessel walls have resulted in crack development and costly stop of power plants in Sweden the last years.

Related to these specific problem research and assessment have been performed to better understand the basis for crack developments.

To develop better methods for design and for tools for assessment there is a need to develop common understanding of the heat-transfer into pipe and vessel walls in turbulent flows.

This topic is very important for the Long Term Operation (LTO) of all plants.

2.2.2.9 Fire risk

Different issues require specific development:

- Comprehensive characterization of different fire loads including increased understanding of passive means for reducing fire spreading (Self distinguishing cables, distance between cable trays, distances between cabinets, reducing oxygen contents)
- Fire suppression models and suppression technologies
- Methods and criteria to assess malfunction of electrical equipment considering combined effects of soot and thermal stress.

R&D topics

- ✓ **It is worth seeking for ways allowing improving modelling capabilities (including transfer of knowledge from non-nuclear field) and elaborating methodology for fire modelling and assessment.**

2.2.2.9.1 Fire and gas explosion simulation for safety

The physical phenomena of fire and explosion in nuclear power stations are often not well evaluated, due to lack of simulation methods that are validated specifically for nuclear applications and accepted by regulators. Such methods exist, for example in oil/gas industry and process industry. The purpose of this topic is to develop nuclear specific fire and explosion simulation methods/methodologies that can be used in safety analysis and be accepted by the regulators.

Several fire tests have been performed in national and OECD-program as well as in US-programs. Fire data have been collected within the OECD-fire project and in the PRISME-project. The knowledge from these fire tests is not fully transferred to fire models.

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There is also a lack of common accepted methodologies for performing fire assessment in European power plants. There is a need for establish a common European methodology for fire assessments.

2.2.2.10 Priority ranking

In July 2014, an update of priorities among the STA-1.2 topics was organized and evaluated. Among current priorities, the new ranking reflects new structure of the Roadmap and latest formulation of the topics. This ranking resulted in the following order of the highest priority topics:

a) Coupling between thermal-hydraulics and neutronics codes
b) Effects of gas intrusion or release in reactor coolant system
c) Passive system behaviour
d) Containment thermal hydraulics
e) Safety assessment through benchmarking and experiments
f) Coupling of system TH and CFD codes
g) Mixing in pools and vessels at low flow rates
h) Stratification in pools and vessels
i) Water hammer assessment

2.2.3 Impact of External Loads and Hazards (STA 1.3)

2.2.3.1 Scope

The main aim is improving methodologies to assess the impact of external loads and hazards on barriers and on structures, systems and components. Following Fukushima lessons, the effect of both single and multiple external events on safety function degradation need to be considered. Time (and feasibility) for recovery and the influence of non-safety systems on barrier strength are important to position. Among external events, a special focus will be put on the impact of electrical disturbances from the grid on plant safety systems and safety functions: the objective is to secure safety system performance.

This sub-area includes needed actions to correctly assess the impact of external events; the assessment shall involve both deterministic and probabilistic methods.

2.2.3.2 State of the art

See the projects focused on impact of external loads and hazards on the safety functions in executive summary.

2.2.3.3 Challenges

- The external events have to be characterised by loads and frequency as well as the risk for co-incident occurrences.
- The effects on non-safety systems and their effects on safety system functions in short and long term recovery actions have to be understood.
- An issue that is of general interest in many of the topics proposed is the consideration of the potential for successful preventive and mitigating human actions.
- Methods for frequency/magnitude assessment for events with very long return periods (>500-1000 years) and a potential for severe impacts on plant safety need to be further developed in view of the major uncertainties involved.

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- Consensus estimates of effects from the on-going climate change indicate substantial impacts on the frequency and magnitude of certain natural external events already in the near future, which needs to be considered in the analysis of external events.

R&D topics

- ✓ **Short term topics**
- ✓ **Assessment of impact from climate change**
- ✓ **Seismic risks and assessment methods in low and high seismic areas**
- ✓ **Methods to assess historical data (near plant and far from plant) to specify low frequency natural events (extreme water levels, precipitation, wind, etc.)**
- ✓ **Methods and methodologies to identify single and multiple external events**
- ✓ **Development of methodologies to assess effects on a multi-unit plant**
- ✓ **Safety impact through effects of external events on non-safety systems"**
- ✓ **Methodologies to specify the design loads from external events**

The external events have to be characterized by loads and frequency as well as by the risk for co-incident occurrences and the effects on non-safety systems on the safety system. The potential for successful preventive and mitigating human actions has to be considered. Methods for frequency/magnitude assessment for events with short and long return periods (from 100 to more than 1000 years) need to be further developed in view of the major uncertainties involved. Estimates of the effects of climate change also indicate substantial impacts on the frequency and magnitude of certain natural external events even in the near future, which needs to be considered in the analysis of external events.

Methods and methodologies to identify single and multiple external events are also necessary to assess effects on a multi-unit plant, as well as how the effects of external events on non-safety systems could affect safety systems.

2.2.3.4 Assessment of impact from climate change

Existing data and assessment of impact from external event have been based on assessing historical data. Based on the existing knowledge of the global environment we are expecting that for some of data on external loads that historical data will no longer give appropriate assessment of the coming risks.

This topic will cover developing of knowledge, models and methodologies to be used for assessing loads from external events for design of existing and new reactors.

2.2.3.5 Seismic events

Seismic activities have different frequencies and spectra's around Europe. Some plants are designed with demands to withstand certain loads. Other plants are not design with specific demands on seismic loads. Some of these plants have performed seismic margin assessment (SMA) to assess safety margin to specific seismic spectra's valid for the site.

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Some plants are performing seismic PSA on demands from regulators. Others are hesitating to perform seismic PSA as SMA assessment is performed and the seismic activity at the site is low.

The knowledge about seismic activities has increased during the last decade and new frequencies and spectra are specified for areas within Europe.

Based on this there is several aspects to handle concerning seismic event:

- Understanding of seismic activities (spectra and frequencies)
- Methods to transfer loads from the ground to the buildings and to components
- Quality and uncertainty of data on component resistance to different seismic loads (Hazard curves)
- Choice of method to be used for assessing plants strength and weakness for seismic loads. (SMA or PSA)
- Methods to upgrade a plant to new more severe seismic loads.

2.2.3.6 Natural events

Methods to assess historical data (near plant and far from plant) in order to specify low frequency natural events (extreme water levels, precipitation, wind, etc.), including multiple external events, are again on the top priority list.

Plants have been designed towards specific data related to extreme weather. These data have been developed based on different methodologies for each plant and specific for each country. No common method to specify design values for extreme values are in place in Europe and neither globally.

A methodology needs to be developed for historical data at or near the plant and on historical data far from the plant. Uncertainties in these data have to be established to specify design values for the plant. Design values shall be specified on frequent occurrence’s and on infrequent occurrence’s and perhaps also on hazard level.

It is also important to specify the validity of the developed data and when reassessment of data shall be performed.

Data on high (or low) sea level, rain and storm, lightening, solar storms, high and low temperatures, freezing river or coast tsunami, wave effects, and combined effects with high wind and assessment of flooding of rivers are included in this topics.

To assess the safety margins of plants new design loads of the natural and man induced external events that comply with the safety and security tracks suggested by the ENSREG should be specified.

Methodologies and models in order to evaluate the safety of plants subjected to beyond design basis external loads shall be developed.

Data are needed for both deterministic deign and assessment as well as for probabilistic assessment

Assessments required for PSA are necessarily associated with major uncertainties. Much work is on-going or has already been done related to methods for frequency/magnitude estimation (extreme value analysis etc.), and probably the tools are to a large extent already there. Additional approaches are needed to gain more robustness in screening methodologies and assessment of effects from

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uncertainties. Other important items include the identification and consideration of additional data types and data sources, e.g., historical data of extreme events (outside of the continuous measurement series) and qualitative description of the characteristics of external events.

The real problem will always be the quantification of the event frequencies, especially for low frequency/high consequence events. The crucial issue would be how to combine these methods with other insights from the relevant areas of science (geology, meteorology, and climatology) as well as simulation models to improve the predictive capabilities.

However, for some external events the specification of the design load (type of load and magnitude) is more difficult (e.g., clogging of main heat sink). This topic should initially try to identify the aspects that constitute a challenge. This may also involve assessment of the potential for successful preventive and mitigating human actions.

R&D topics

- ✓ **To develop methodologies for risk assessment of external events and combined events**

2.2.3.7 Impact of single or multiple external events

Methods and methodologies to identify single and multiple external events to assess their impact on reactor safety are needed as well as the development of countermeasures to reduce the effects of external events on NPPs.

Regarding the first sub-topic, the focus should be on multiple events – this is a crucial and surprisingly undeveloped area.

Regarding the second sub-topic, approaches have been suggested by, e.g., the Nordic Owners group (NOG), using the concept of “enveloping events”, i.e., identification of a set of external events with a similar impact on the NPP (structural load, impact on ultimate heat sink, etc.).

This issue relates to the topic “Methodologies to specify the design loads from external events”.

2.2.3.8 Effects of external events on a multi-unit plant

Safety nuclear plants are based on assessing each single plant separately. There exist some design demands on identify systems that are share among sister plants and to secure that the system in the sister plant are available when needed for the actual plant.

Design demands are not developed to secure demands when hazard events occur at several units at the same site. Risk assessment have historically not been developed assuming that several plant at the same site needs the capacity of safety systems and plant personal.

There are also connections to other “multi-plant issues”, such as the definition of a source term for a site, the definition of PSA L2/3 safety goals for a site, etc. The focus should probably be on identifying aspects of plant safety and of accident sequence characteristics with “multi-plant characteristics”, i.e., where it can be reasonably expected that the situation may be worse (or more favourable) on a multi-unit site than for a single unit site.

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2.2.3.9 Non-safety systems impact on plant safety

The strength to withstand certain extreme external events is depending on the operability of non-safety classed systems. Plant heating systems are needed to handle cold weather scenarios, Room cooling systems are needed for handling extreme warm weather situations. To handle scenarios in with the main heat sink fails by clogging in one or another way non-safety system are needed for cooling functions of systems and the core.

Depending on plant design there exist various scenarios in which non-safety system are important to handle the extreme weather scenarios.

Methodologies has to be developed and data have to be collected to develop a common methodologies perform PSA-assessments which this dependences included.

Concerning deterministic assessment it is important to specify demands on these system including demands on operability.

There are also other scenarios in which non-safety systems, operational system or diversified systems are important for the plant safety and the risk assessment. A common view of methods to include them in deterministic and probabilistic assessment has to be developed.

2.2.3.10 Priority ranking

The 2013 priorities ranking in subarea 1.3 resulted in the following order:

a) Impact of external events on reactor safety
b) Seismic events
c) Methods to assess historical data to specify low frequency natural events

2.2.4 Effect of Electrical Grid Disturbances (STA 1.4)

2.2.4.1 Scope

This sub-area focuses on external effects on the plant propagated from the external grid and its connection to the plant internal electrical buses and electrical components important to safety. It includes assessment of effects from lightning that can have negative impact on plant safety functions. It addresses also the impact of new Grid-Codes, provided by the Ensue, implying wider voltage and frequency ranges at the first substation near the power plants; this may lead to higher stresses on the existing equipment of the power plant.

It also includes the effects on modern electronics or digital equipment that is more sensitive to magnetic fields from motors or generators than earlier used components.

It is important to develop methods to assess the maximum effect that these disturbances can have on the plant electrical and I&C equipment, diesel generator for emergency power, etc. The plant control and protection systems have to be designed to withstand these effects or the sources for these effects have to be controlled.

2.2.4.2 State of the art

This sub-area has a connection with actions taken in the OECD DIDELSYS Task group as well as Owners group specific programme that have also been handled in the NPSAG (Nordic PSA-group).

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2.2.4.3 Challenges

Among the external hazards, particular attention has to be paid to grid disturbance effects on the plant through the internal electrical buses and other electrical components important to safety. It includes assessment of the effects originating either from lightning or from motor magnetic fields on modern electronics and/or digital equipment which are far more sensitive to magnetic fields than components used in the past. Other equipment like plant electrical and I&C equipment, or diesel generators for emergency power may be also affected. The design of plant control and protection systems has to be based on an increased understanding of these effects and the sources of these effects have to be investigated.

It includes the impact of the New-Codes on the equipment of the NPPs, on the generator side, but also on the local distribution network.

2.2.4.4 Electrical loads from the lightning.

Nuclear plants are supposed to withstand effects from lightning without interruption of the operation. For each plant a specific maximum loads (voltage surge) is specified for design of the plant protection.

If the lightning protection is too weak, the operation of both operation and safety I&C -systems for could be negatively affected. It is difficult to assess the negative effective in cases the lightning protection is failing or the loads are higher than those the protection is designed for. Based on this fact it is important to specify the maximum loads (including extreme cases) that the lightning protection shall be designed for.

The stress test give insights in those different loads are used for the lightning protection for different plants. It is not clear if these differences are related to difference in historical data for the specific site.

R&D topics

- ✓ **To develop a common methodology that includes a suitable safety margin for specifying the maximum loads from lightning.**

2.2.4.5 Impact of new Grid Codes on the NPPs

Methods to specify voltage surge and frequency changes in the external power grid including effects of renewable production connected to the Grid (IAEA D-NG-T-3.8) and dynamic effects of voltage surges and frequency changes in the external grid to the plant electrical systems appears to be rather complex.

The issues are affecting design of nuclear power plants. It also affects methods for performing deterministic safety assessment and risk assessments. The basic philosophy is that disturbances or new operating conditions on the external grid shall not affect the function of safety system. Protection devices shall initiate needed actions to secure that all safety functions will be operable under and after the disturbances. New operating conditions will be addressed with more renewable energy sources connected on the grid which may lead to transients phenomena or higher voltage steady-state values for certain durations which may stress the equipment; in particular, this will affect the frequency of loss of offsite power.

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Different kinds of failures resulting in loss of offsite power are probably connected with different probability on return time for power is back on the grid. To develop realistic risk assessments different failures shall be considered with most probable time for re-powering the grid or having power on the secondary external power supply line... Data related to these dependencies are lacking.

Events during the last 10 years have indicated that the design has not in all cases been successful to initiate the needed protective actions. The issue requires dynamic simulation of plant power supply and the grid as a boundary condition.

Lessons learned is that the understanding of possible cases that can give voltage surges affecting the internal grid have to be increased as well as data on the dependency between failure type and time for recharging the power lines.

R&D topics

- ✓ **To develop dynamic simulation of plant power supply and grid as boundary condition taking into account:**
- ✓ **Detailed analysis of disturbance – types and frequencies**
- ✓ **Refine the Modelling of the electrical systems**
- ✓ **Determine plant response with regard to ”new” disturbances**

2.2.4.6 Electrical grid “stabilization” with nuclear plants

The normal operation for Nuclear power plants is base load operation with full power; operating at lower power is primarily related to the power plant itself. Such full power operation do not stress power changes or running the power plant at lower or much lower power frequently.

However there is a demand from ENTSO-E- to operate in such a way to “stabilize” the grid with the NPPs, asking for more flexibility. There is a risk that a way of operation will result in new failure modes or infrequent failure modes will be frequent failure modes

Such changes in availability will be costly for the plant owners. Increased knowledge on the risk of not using the power plant as a base load plant is the focus of this topic.

2.2.4.7 Design basis for electrical components

Development of design basis for electrical components according to IAEA NS-G- 1.8 includes assessment of safety margins for establishing values for component protection devices.

To secure the operability of components during different kind of electric disturbances it is important to develop demands for each type of component on resistance to variations in different parameters as: Frequency, Voltage variations, Current variations, Magnetic fields. Such data shall be developed based on the specified levels for initiating protection devices and the knowledge of the time constant for such devices. It shall also be based on the knowledge of safety margins valid for each component based on the manufacturer’s qualifications.

This topic is supposed to support development of methodologies to perform such assessments to be able to find the optimum demands for each component.

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2.2.4.8 Safety margins for emergency diesel power

Assessment of the efficiency of the diesel generator for emergency power needs to evaluate their safety margins as safety systems of the plant. Diesel generators can be loaded to a certain level before protection devices cut off the loads.

The load from pumps motors is depending on their needed current during start-up and during base load operation. These data are depending on several parameters as: Load on the pumps, Temperature and other environmental data, Oil temperature and viscosity, Cooling water temperature.

Using the Diesels and their attached loads under normal condition or even under specified design base condition can be secured by testing and assessing.

What is unclear is if it is possible to specify safety margins (to a level that will result in failure of the diesels to operate) for different scenarios when some parameters exceed the specified values.

It is also unclear whether effects of ageing can be expressed in terms of safety margins.

This topic includes research that increase utilities understanding of the existing safety margins on their diesels for different scenarios.

2.2.4.9 Production of disturbances by electrical power equipment

Methods and design rules for avoiding disturbances from electrical motors and generators and the diesel generator on modern electronic equipment (Value of EMC-directive).

This topic is related to develop design methodologies or test procedures that secure that new modern equipment will work in the existing environment in the plant.

It includes methods to characterise the electrical environment in the plant and methods to reduce the disturbance level of certain parameters to a level that the new modern equipment will have high availability.

2.2.4.10 Design of electrical equipment against disturbances

Methods and design rules can help avoiding propagation of electrical disturbances from the grid to the plant electrical systems. This topic relates to design rules for all kinds of power sources to the internal grid and especially the grid that support functions for safety system and for mitigation system.

There exist a lot of international rules developed by 10 CFR 50, IEEE, IEC, KTA, etc...

These rules give a lot of specific demands but some of these demands are very similar between the different guides. On the other hand there are also a lot of differences among these guides and the power plants do not fulfil several of these demands.

There is therefore a need to develop a common set of rules and perhaps a peer review program to secure that suitable quality for high availability in hazard situation are secured.

This topic could include benchmarking of plants or guides.

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2.2.4.11 Priority ranking

The priorities ranking of subarea STA-1.4 organized and evaluated in 2013 resulted in the following order of top challenges/topics:

a) Design of electrical equipment against disturbances
b) Impact of new Grid Codes on the NPPs (earlier "Electrical current supply from a smart grid")
c) Design basis for electrical components

2.2.5 Effects of Human Errors and Reliability Evaluation (STA 1.5)

2.2.5.1 Scope

This sub-area includes topics needed for assessing the impact that human performance might have on the reactor safety. Data on human reliability for use in risk assessments are therefore included in this sub-area from the point of view of their effect on the plant safety.

Organisational structure, man machine interface and safety culture are covered in road map for AREA 3 and sub-area 3.1 and 3.2.

2.2.5.2 State of the art

A programme has so far been developed in the frame of the OECD-WG-risk, NPSAG (Nordic research in PSA), SAFIR (Finnish research program) and MMOTION.

2.2.5.3 Challenges

Typically, deterministic assessment relies on automatic functions in short term (30 minutes). Human is supposed to handle actions where more time is available for decisions. Human interactions (correct and incorrect) in short term will anyhow have impact on scenario developments.

Operators and maintenance actions that are not covered by procedures are assessed, neither in deterministic, nor in most of the risk assessments performed so far. Actions are taken to include such effects in the risk assessments.

Issues related to quantification of risk related to human failures in PSA assessment are the main subject for this sub-area.

2.2.5.4 Effect of human errors on the risk, critical interactions

During the PSAM11 conference in Helsinki there was a panel session with the objective to discuss where we are in terms of commission errors and the conclusion from that session was that there is no consensus as of today. It is nevertheless important to develop some common basis and practise to identify in what kind of scenarios that errors of commission should be taken into account and how that can be practicably achieved. It is foreseen that more research/pilot works will be done to extend the EOC analysis from the full power to the scenarios where EOC is more possible and important, e.g. shutdown situation, fire analysis.

It appears, there is a need for substantial further efforts to establish a sound and credible basis for the methodology and formal identification processes

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2.2.5.5 Human reliability data evaluation

Numerous methods are developed and applied for assessing human reliability for supporting common risk assessments of human performance. There seem to be missing consensus on the quantification while other steps in the analysis are well covered.

Need for training in methods, procedures and in real analysis work and further benchmarking may be important to arrive at results from different analysts that agree better. Past R&D projects have had difficulties in defining if any method provides better quantitative results than another method. It is therefore difficult to say that one method is preferred in front of another method. However, the application of any method often reveals inconsistencies in documentation including justification of choice of method. It is therefore important to develop guidelines on how to implement HRA in our PSAs.

There is also a need to develop detailed HRA guidance for specific applications: e.g. in fire analysis (NUREG-1921), external event analysis including seismic, tsunami, etc.

R&D topics

- ✓ **To produce a guideline for a state of the art HRA for PSA purposes, based on performed assessments, ensuring that plant specific properties are properly taken into account in the HRA.**
- ✓ **To define a suitable operating mode for all the different acknowledge operator's actions.**

The Working Group on Risk Assessment has completed a task to assess the feasibility of a joint effort on HRA data. The main objective of the task was to develop a framework for collecting, analyzing and exchanging HRA data. The result was published in NEA/CSNI/R (2008)9 Report.

2.2.5.6 Databases for supporting risk assessment

There is a need to set up a good human performance/error data collection system and to collect the more human performance data from plant near-missing reports, incident/accident reports and simulator exercises. Detailed human performance data would help understanding the important PSFs and sufficient human performance data would greatly help in the quantification of the human failure events in PSA.

2.2.5.7 Advanced modelling of human interactions

Develop PSA data and modelling according to the following ideas: One might explicitly model the procedures in some detail, and use these procedures models together with detailed plant state information to model human actions within each history. Within a simulated time history, everything is known (or can be known) that a modeller of human performance could wish for: history-specific event timing, history-specific equipment failures, history-specific instrumentation status, initial conditions (including core history), and so on. Taking full advantage of this modelling potential would be a lot of work (one would have to translate all the procedures into a model, and link to instrumentation and so on), but as one of the other comments implied, work has been going on for decades within the traditional framework, with less than complete success. This first paragraph boils down to HRA within a dynamic PSA. This is time consuming and a current field of research by GRS.

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In an early phase, one might first emphasize modelling the procedures and severe accident management guidelines correctly, just to understand how optimal the current procedures are, and only later introduce "errors" or off-procedural human actions. Modelling of post-accident actions in particular would be strongly influenced by the behaviour of instrumentation and possible lack of detailed knowledge in some conditions even when instrumentation is working (e.g., what is the water level in the drywell?).

The above comments include modelling of important effects such as recovery of failed equipment. If there is interest in really modelling severe accident management guidelines, this would receive high priority. Hard to solve are: which modelling approach is adequate, how can the level of detail/accuracy in modelling be balanced against the (financial) effort needed to produce results in time.

2.2.5.8 Priority ranking

The priorities ranking of subarea STA-1.5 organized and evaluated in 2013 resulted in the following order of top challenges/topics:

a) Assessment of the impact that commission errors may have in the risk assessments. Ways to identify critical human interaction in plant scenarios
b) Establishment of human performance and reliability database with reference cases for supporting risk assessments
c) Advanced modelling of human interactions

2.2.6 Advanced Safety Assessment Methodologies (STA 1.6)

2.2.6.1 Scope

This sub-area includes:

- ✓ Understanding of safety margins and best estimate methods, Integration of deterministic and probabilistic safety assessments, in order to better evaluate safety margins with best estimate methods
- ✓ Determining the data, methods and knowledge needed to understand safety margins (strength and weaknesses of NPPs) including safety margins in components such as pipe/vessel and systems

2.2.6.2 State of the art

Increased use of multi-scale mathematical modelling will lead to an increased understanding of plant safety margins by using best estimate methods for assessment of transients and of material properties in combination with new technologies, such as dynamic PSA and Monte-Carlo methods for identification of critical scenarios for nuclear power plants.

2.2.6.3 Challenges

This sub-area covers programmes that aim at increasing the knowledge about existing safety margins of the plant by assessing and comparing the quantitative data on margins with critical values associated with safety barriers and radiological release levels.

Assessments should be based on best estimate methods and material properties should be characterised by non-conservative data.

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This sub-area includes future development of methodologies in which new technologies (as Dynamic PSA and Monte-Carlo assessments or a combination) are used in parallel with existing methods for probabilistic and deterministic assessments, incl. for the assessment of the efficiency and functions of the safety systems. These methods may better model / predict dynamic behaviour and also time dependent scenarios. The assessment of these methodologies has to include uncertainties and it is critical to quantify the uncertainties in a correct way.

R&D topics

The general challenge in this sub-area is to increase knowledge about the existing safety margins of a plant. Several methodologies can be further developed in order to improve the accuracy of evaluations:

- ✓ **Risk informed methodologies, usually developed by operators, based on decision-making theory and/or economic models can still be improved in order to optimize the risk accuracy evaluation,**
- ✓ **Understanding the safety margins and best estimate methods, i.e. integrating the deterministic and probabilistic safety assessments, is another possible route to improving accuracy. This includes development of methodologies (such as methods developed in the frame of OECD SM2A project, Dynamic PSA and Monte-Carlo assessments or a combination) which are used in parallel with existing methods for probabilistic and deterministic assessments.**

A particular challenge for these methods is to better model/predict dynamic behaviour and also time dependent scenarios while not affecting the assessment of uncertainties.

Application of these developments should address better evaluation of safety margins for Reactor Pressure Vessels (RPVs), main containment, passive system and pipes in case of LOCA or PTS events, but also for beyond design situations or natural circulation conditions.

2.2.6.4 RPV safety margins

This topic relates to RPV and combines knowledge from thermal-hydraulic assessments with material data. Probabilistic methods knowledge from the different fields can be combined to develop a better understanding of the safety margins for each specific plant.

Loads from pressure and temperatures are calculated by CFD or integral codes and these data are used to assess the material toughness.

The output shall be used for supporting a common European bases for safety factors for RPVs.

A detailed description of the possible approach is given in the project proposal for PROSAFE.

2.2.6.5 Safety margins in piping system

This topic relates to piping and combines knowledge from thermal-hydraulic assessments with material data. By probabilistic methods knowledge from the different fields can be combined to develop a better understanding of the safety margins for each specific plant.

Loads from pressure and temperatures are calculated by CFD or integral codes and these data are used to assess the material toughness.

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A detailed description of the possible approach is given in the project proposal for PROSAFE.

2.2.6.6 Containment safety margins

This topic relates to methodologies in which the strength of the containment and its openings are assessed by methods using dynamic assessment tools. It will include large scale and scaling tests and related test data that could be used for validation of models.

The output shall be used for supporting a common European bases for safety factors for containments.

There is a need to understand when the containment is no longer in the elastic zone of damage and it is of great importance to understand leak rates.

Dependency of pressure and temperature to be able to develop emergency procedures are part of this topic. These data are also essential for PSA-level 2 assessments.

2.2.6.7 Best estimate methodology

This topic will include proposals that are related to develop methodologies for performing best estimate assessment and the associated uncertainty assessments.

The long term goal will be an increased acceptance from all stake holders to accept best estimate assessments in all areas related to nuclear safety.

Proposals can cover any aspect related best estimate assessments.

2.2.6.8 Safety margin during LOCA events

This topic is specific cases that are partly covered within the topic 6.4 but with specific focus on the effects on the core and fuel properties during LOCA scenarios. It can also include assessment of loss of feed water scenarios. The aim is to develop acceptable best estimate methods and uncertainty assessment to understand the margins to fuel failures.

2.2.6.9 Opening time for large break LOCA

In order to be able to assess the consequences of large breaks LOCAs inside connecting pipes and into the RPV and connected vessels, it is of large importance to be able to predict the break opening time. Break opening times for different break mechanisms shall be established at least for these that can result in very fast opening times.

2.2.6.10 Pressurized Thermal Shock (PTS)

This issue is important for specifying the life time of PWR-reactors. The development of advanced methods for PTS assessments appears useful, including for instance the coupling of system TH with CFD-codes.

This issue is Important even for single phase PTS, proper experimental validation data for numerical methods is not available, let alone for multi-phase PTS

2.2.6.11 Methodologies for beyond design assessments

Safety assessments for design bases scenarios are steered by specific rules mainly developed from US NRC and their guidance. Different variants exists in different countries and used in the licensing reports.

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Assessing beyond design event effects relies on very little guidance on accepted methodologies. Most international guidance specifies that best estimate assessment shall be performed, which asks for realistic assessments.

To specify realistic assessment methods needs support on issues like single failure, environmental data, use of non-safety classed system, plant status, etc. The realistic assessments have to be complemented by sensitivity assessments. The content and coverage of sensitivity is not fully clear yet.

2.2.6.12 Algorithm assisted assessment in nuclear safety

Applied mathematics can help optimizing the use of computer codes through optimisation of input data handling to identify, understand, and thereby control the safety response of the system corresponding to the given set of equations solved in the computer code. The main results that can be gained from these methodologies are the following:

- Uncertainties propagations in simulations
- Learning of safety response surface
- Design of computer experiments

Of special interest for the safety is to identify the cliff edge effects that would dramatically change the behaviour of the system. For example, this approach could help finding criticality conditions in a reactor core as a combination of many control parameters, but it could also be applied to other safety items such as: mechanics, thermal hydraulic, fire, release of radio nuclides, tools for dynamic PSA assessments etc.

2.2.6.13 Tools for dynamic PSA-assessment

Assessing specific scenarios in PSA in which a predefined path is not available to specify the existing used methods for PSA level 1 and 2 are not convenient. This also involves events in which human actions are involved and change the order of sequences for the assessed scenarios.

Tools have been under development and are partly in use for performing dynamic PSA assessments. The existing methods are in most cases time consuming and costly to use. There is a need to develop these methods further and also develop methodologies on when and how they shall be used.

The topic also includes needs for validation of these new tools.

2.2.6.13.1 Assessment of EOP by Monte-Carlo calculations

A sub-topic related to this issue is assessment of Emergency Operating Procedures (EOP). EOP include several operator’s actions that may be done at different times and thus in different reactor conditions which influence the scenario and may lead to different consequences of the initial event(s).

The operator actions may be then simulated at different time using simulator and or codes coupled with simulation of the different actions at different using Monte Carlo. Analysis of the results is helpful for the optimization and validation of the EOP.

Developing tools and methods to EOP is an R & D challenging issue.

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2.2.6.14 IDPSA project and network

This topic relates to the activities specified within the IDPSA network.

The network and its program focus on identifying new ways to find critical scenarios by assessing the outcome of different combinations of scenarios with fast running computers.Proposal for EU FP7 project: IDPSA - Integrated Deterministic – Probabilistic Safety Analysis:

(See also topic 1.1.5.). The proposed project is considered as an important practical step towards deployment of advanced safety analysis and justification methods which combine the use of deterministic and probabilistic methods into industrial practice

The primary goals of the project are:

- To develop further IDPSA methods and for joint application with PSA and DSA in practice of safety analysis;
- To assess advantages and present limitations of joint application of IDPSA with state-of-the-art PSA and DSA methods based on experience from a set of pilot realistic applications.

The main outcomes of the project will be:

- New and improved IDPSA methods;
- Recommendations and guidelines for joint applications of IDPSA, PSA and DSA;
- Summaries of experience of addressing pilot realistic applications with IDPSA, PSA and DSA.

This will give increased awareness of the research, utility and regulator communities about advantages and current limitations of the new approaches to safety analysis which tightly combine deterministic and probabilistic methods.

2.2.6.14.1 Other fast running codes

With today's computer capacity the development/evolution of engineering simulators which represents the whole plant and/or the whole site is possible. The use of these engineering simulators is very broad i.e. to be used in evaluation of event trees, development for emergency scenarios as well as to study the interaction between the combined external loads and the site specific critical functions.

The topic also includes needs for validation of these new tools.

Developing tools and methods of fast running codes is an R & D challenging issue.

2.2.6.15 CFD in multiphase applications

In many cases the assessment of transients and hazards are performed using correlations for specific phenomena. These correlations are developed with specific boundary conditions and have a limited accuracy outside the specified boundary conditions.

With CFD technique it will be possible to model the phenomena based on more detailed understanding of the phenomena and its interaction with surrounding media. This can be done with

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2D and 3D-solutions. This technique is under development and needs increased knowledge about phenomena and on calculation techniques.

In topic 2.7, 2.11 and 2.18 certain applications are specified related to this issue. This topic 6.12 will include all kinds of applications related to development and validation of CFD for use in multiphase applications.

Boiling prediction in NPPs, both PWR and BWR, is a very important subject as it directly impacts safety and performance of the plants and it is still not well understood.

2.2.6.16 Modelling of CCF by shock models

For somewhat different reasons, an opportunity might also be created to model common cause failure in more interesting ways, based for example on shock models. Here, the path forward is presently less clear than for human action modelling, partly because existing procedures for normal operations, emergency response, and action management already provide a basis for modelling of human actions, while PSA common cause failure models are much more abstract. However, progress has recently been made in coupling component reliability characteristics to scenario physics, so that environmentally-induced component failures (e.g., severe-accident-induced failures of instrumentation) can now be modelled more explicitly. The experimental basis does not exist to model these effects with high predictive accuracy, but interesting work could arguably be done in vulnerability search mode.

2.2.6.17 Priority ranking

The priorities ranking of subarea STA-1.6 organized and evaluated in 2013 resulted in the following order of top challenges/topics:

a) Methodologies for beyond design assessments
b) CFD in multiphase applications
c) IDPSA- project and network

2.2.7 Design of Reactor Safety Systems (STA 1.7)

2.2.7.1 Scope

Designing of reactor safety systems to handle obsolete components and to up-grade them to handle new safety demands.

2.2.7.2 State of the art

See the projects dealing with advanced safety assessment methodologies in executive summary.

2.2.7.3 Challenges

Existing plant and new plants have to be up-graded to handle new safety demands, new experiences, and to handle obsolete components to get permission to run the plants. The topic beneath are developed to support needs for this.

R&D topics

- ✓ **Among the challenges for the design of safety systems the most important are:**
- ✓ **design of digital system with integration into existing plants**

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- ✓ **increased diversification and robustness of safety systems**
- ✓ **use of passive system for safety function**
- ✓ **methods for reactivity measurements under accident conditions**
- ✓ **design of level measuring systems to withstand high temperatures**

2.2.7.4 Integration of digital system into existing plants

Existing plants have to be up-graded with new components and systems to handle aging issues and to handle plant up-grade programs. As the manufacturing industry no longer produces analogue components and systems there is a need for developing acceptable methods for integration new digital technology in analogue systems. Safety demonstration has to be developed for different cases.

2.2.7.5 Simultaneous loss of offsite power and ultimate heat sink

The Fukushima event and the outputs from the ENSREG stress test has indicated that there is a need for strengthen plant design to handle loss of offsite power (LOOP) and loss of ultimate heat sink (LUHS). These events are outside the design bases events and can be classified as beyond design events. There exist no common accepted international guidance to specify demands and criteria's for these events. There is a need to specify boundary conditions, system and components demands to support design of such functions and systems.

2.2.7.6 Use of passive systems for safety function

This topics focus on developing design strategies for using passive system for safety functions in existing and new built plants. The topic will specify arguments for the benefits of having passive systems instead of active system.

The topic will provide arguments for important design features on passive systems for them to be implemented into plant design and support or replace active safety functions.

Topic 2.3 is related to assessment methods for passive systems. To be able to specify the benefits of passive system/functions it is important that methods according to topic 2.3 are developed.

2.2.7.7 Reactivity measurements in severe accident conditions

Existing measurement methods can only give indirect assessment of any reactivity changes in the reactor core when the ordinary measurements in core has been destroyed or get out of specified operating range. Local reactivity changes in the core or in the molten corium are difficult to measure even with some of the ordinary reactivity measuring system operable.

This topic relates to developing of measuring system or simulators that can support the operators with information on any on-going reactivity change during severe accident progression.

2.2.7.8 Water level measuring system under high temperature

Existing systems for water level measurement within the reactor core work with high accuracy under normal operation and most of the design basis accidents.

In severe accidents with very high temperature in the core or/and in the containment the accuracy is low or the measurement are not valid.

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This topic relates to any development of methods to assess reactor water level in scenarios with high temperature e in the core or in the containment.

2.2.7.9 Strategies against external hazards

Counter measures against external hazards include Pre-warnings and other countermeasures for reducing effects of external events.

Plant operators have to operate their plant with specified boundary conditions to be within the range valid for the license. In case the boundary conditions are exceeded the plant has to be operated differently, and this has to be prepared. This topic relates to developing of strategies to handle scenarios in which there is an increased risk for passing these boundary conditions and to any kind of measuring system that can support such strategies.

This topic has connections with the topic 3.4.

2.2.7.10 Over-sized safety systems

Benchmarking of benefits and weaknesses of diversification of safety system with too high capacity is desirable.

To developed well balanced design of reactors including their safety system it is of importance to understand minimum capacity of safety functions to be able to fulfil specified acceptance criteria's for different failure modes.

The safety systems do not only have positive effects on the plant safety. There are several aspects that have to be assessed to find the optimum design of a safety system:

- To high capacity when used without single failure,
- To high capacity when it is used in appropriate ways in scenario's that it is not designed for. Activated manually or based on a failure in the safety logic,
- Interactions with diversified "safety" systems or non-safety systems.

With increased demands on more "safety" or diversified systems there is a need for developing methodologies for safety assessment that secure that safety system do not introduce critical failure modes into the plant..

One example is flow regulation for ECC-system and RHR-system where to large flow results non negligible in NPSH problems. This is to a certain extent an unexplored area today. This kind of problems is also discussed within the proposed IDPSA project.

2.2.7.11 Priority ranking

The priorities ranking of subarea STA-1.7 organized and evaluated in 2013 resulted in the following order of top challenges/topics:

a) Integration of digital system into existing plants (average ranking 8.0)
b) Use of Passive system for safety function (average ranking 7.73)
c) Strategies against external hazards (average ranking 7.57)

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2.2.8 Pre-normative Research (STA 1.8)

A new subarea of "pre-normative research" has been integrated into the Technical Area TA1 "Plant Safety and Risk Assessment" during 2014. Its scope, challenges and topics are under development. Major objectives and results of discussion in TA1 are summarized below:

- Several safety functions are to be maintained to guarantee the safe and secure operation of the nuclear installations in any circumstance and at any time in their life from design to decommissioning. They can roughly be summarized as follows:
- Maintaining the convenient cooling conditions in general and locally whenever needed , through the control of the thermal power, produced either by the chain reaction or the radioactive decay, and avoiding any exothermal reaction which could endanger the confinement capacity of the fuel cladding (when existing), the primary and secondary circuit and the containment,
- Keeping the cooling of all the heated parts of the plants, mainly the fuel, but also the fuel pound and the containment. A special item in the cooling capacity refers to the degraded core after a severe accident: the geometry of the core has to be such that the cooling remains possible, i.e. keeping sufficient space between fuel rods to have the coolant going through after the accident, and not to keep a coolable geometry despite the quenching of the core that could result in a big non coolable debris bed,
- Keeping the structural integrity of the different barriers between the FPs (Fission Products) and the environment. In this domain, one has to consider the risk of hydrogen explosion, but also, the resistance of the reactor in case of external aggression in case of earthquake and flooding, but also the consequences of different malevolent acts. The safety criteria are aimed at keeping the safety functions of the nuclear installations despite failures in materials and components and/or in human behaviour.

Safety criteria have been defined at the beginnings of the exploitation of civil nuclear power to guarantee respect of the above mentioned safety functions. They were in compliance with an established list of transients linked to "initiating events" and belonging to a safety set-case, quoted as the Design Basis Accidents (DBA). Their definition was provided in several documents, among which the appendix K of the 10 CFR 50 for the Pressurised Water Reactors (PWR).

R&D topics

- ✓ **To develop methodology identifying the safety research needs that should allow confirming the appropriateness of some of the criteria but also identifying new transient types or new physics to be considered, with the necessity to propose criteria for the designer to develop his conception and the safety assessment to evaluate the NPP safety characteristics.**

- ✓ **This work has to consider all the experienced gained as well as the experimental data and computational developments that were or are still under development.**

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2.2.8.1 Pre-normative research related to safety criteria

The existing safety criteria have been adapted to account for new fuel, improved burn-up, extended operation, but they have never been significantly reset since ancient times to guarantee respect of the mentioned safety functions, despite changes in the safety objectives reactor design and so on.

Nowadays, the widespread diffusion of nuclear energy and the permanently increasing safety objectives require enhanced safety levels for each individual nuclear installation, so that the whole safety approach is to be re-thought and fully revised, including criteria.

Accordingly, it is worth relying upon the outcomes from operating experience and the R&D to determine the domains in which the safety criteria could be up-dated and improved through pre-normative research to better match the objectives of a secure and safe operation of the nuclear installations, as well as those domains which could demand a specific R&D effort. This ambitious objective could be achieved through a suitable and comprehensive review of all possible accidents, including, as reasonably possible, those belonging to the Beyond Design Basis Accident (BDBA).

2.2.8.2 Specific safety criteria for BDBA

As said, pre-normative research on safety criteria is presently to be carried-out mainly in the BDBA domain, accounting for the lessons learned from the Fukushima accident targeting the Gen II and III plants in operation (e.g. in the work of the Design Extension Conditions, DEC), under commissioning and / or design.

Whereas the US regulatory framework do not foresee deterministic safety assessments for events that are less probable than the DBA (events with lower probabilities than 10-6 per year shall be assessed only by probabilistic methods), the European guidance do ask for deterministic assessment of events classified in the BDBA domain, too. The WENRA Safety Reference Levels for existing reactors (2008, update 2014) and WENRA Safety Objectives (2010) for new reactors are the major existing references in this area.

There is a lack of guidance on which acceptance criteria's shall be used for BDBA events. Some countries use the same criteria's specified for DBA-event (the US based criteria's) others use criteria's that are less conservative. Based on these differences there is a need for developing:

- Philosophy for acceptance criteria's for beyond design events
- Establish guidance on how to specify criteria's for different important safety barriers and other important parameters in beyond design events.

Also probabilistic assessments needs supportive guidance in specifying acceptance criteria's to be used for realistic best estimate assessment of system demands. So, a work related to harmonization of acceptance criteria's for beyond design events shall include also guidance for criteria's to be used in PSA assessments.

2.2.8.3 Pre-normative research for SMR

Current widely used licensing processes and regulatory requirements for safety demonstration have been developed mainly for very large light water reactors (1000 MWe+). In recent years there has been a trend to develop novel, small to medium size reactors, rated from 14 to 300 MWe, with creative siting options (even floating – the Russian barge power plant, or submersible – the Flexblue

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of DCNS) [see STA 6.2 Innovative LWR concepts including: High Conversion LWRs, Small Modular Reactors, etc…and STA 6.3 Innovative LWR specific safety issues, for details]

In order to become technically and economically feasible, such reactors require thorough reconsideration of both political and regulatory decision making. Many safety functions that in large reactors require complex interacting process, I&C and power supply systems can be implemented in SMRs using simple structures and inherent natural phenomena like conduction and convection. Clearly there is need for research as to:

- Which elements of the traditional defence-in-depth would need to be retained.
- Which elements might need to be implemented in new ways.
- What physical phenomena could be relied on for which safety functions.
- What design requirements and safety margins should be established for such "inherent safety" functions.

In this context, the DiD (Defence-in-Depth) approach is to be declined into two items:

1. the structural defence (release barriers) –
2. the functional one (prevention, limitation, mitigation).

The feasibility of SMRs depends on whether they could be deployed in a serial way, while the current licensing processes assume only few very large projects at any given time, necessitating every time unique and time-consuming review and approval processes and elaborate dedicated oversight of construction.

A fleet of 10 reactors each rated at 160 MWe and located on the same site (in the land or in the sea) is to be licensed, overseen and operated very differently from (and hopefully more efficiently than) a unique 1600 MWe reactor. Suitable pre-normative research programmes are needed to support and contribute to the deployment of SMRs in Europe.

2.2.8.4 Development of Failure Tolerance Analyses (FTA)

Further topic related to pre-normative research could be development of methodology and identification of the research needs that should allow for Failure Tolerance Analyses (FTA). According to moderns requirements (e.g. STUK YVL), FTA shall be carried out to demonstrate that all systems performing safety functions and their auxiliary systems satisfy the failure criteria, in order to ensure the reliability of a system or component, and consequently the reliability of the safety systems.

The FTA generally has to address:

- Redundancy (Single failure proof),
- Defence in depth,
- Diversity (CCF proof),
- Separation and
- Human errors.

The licensing case definition can be based on the same presentation principles as the PRA model. The baseline for this is the initiating events and their categorization into event classes in SAR. The PRA

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model needs to be modified for FTA. This requires an assessment of the PRA to decide whether the PRA can be made correct and complete for the deterministic FTA applications.

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3 TECHNICAL AREA 2 – Severe Accidents (TA2)

Technical Area Leader: Jean Pierre van Dorsselaere (IRSN)

3.1 Executive Summary

3.1.1 Scope

The main public safety goal for nuclear power is to allow society to benefit from low-cost, clean energy without being faced to unacceptable health and/or economic risks. With appropriate site risk evaluations, plant designs and management, current Gen.II and Gen.III nuclear power plants (NPP) show high levels of robustness and low probabilities for severe accidents (SA). But, despite the highly efficient accident prevention measures adopted for the current Gen.II and the still more demanding ones for the Gen.III plants, some accident scenarios may, with a low probability, result in SA, as recently emphasized with the Fukushima-Daiichi accidents in Japan. This SA can result in core melting, plant damage and dispersal of radioactive materials out of the plant containment, thus threatening the public health and the environment.

This risk can be substantially decreased when state-of-the-art devices currently available for prevention and mitigation of severe accidents are implemented. Lessons from the Fukushima accidents and consequences related to Accident Management provisions from the recently completed ENSREG stress tests and other national activities will lead to further enhancement of the NPPs safety.

Within this technical area, general objectives are defined and followed by specification of research and innovation challenges to further reinforce the NPP safety provisions.

3.1.2 Objectives

Some predominant phenomena require a better understanding in particular to improve the Severe Accident Management Guidelines (SAMGs) and to design new prevention devices or systems for mitigation of SA consequences (or even terminate a SA).

Seven technical sub-areas address the 6 objectives listed below, the three first ones being directly linked to mitigation processes:

1. Increased efficiency of cooling a degraded core
2. Preservation of the containment integrity
3. Reducing the source term
4. Reducing the uncertainty on environmental impact assessments
5. Understanding the evolution of complete severe accident scenarios
6. Improving the emergency preparedness and response

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Objective	Sub-area 1	Sub-area 2	Sub-area 3	Sub-area 4	Sub-area 5	Sub-area 6
Increased efficiency of reactor cooling	■					
Preservation of the containment integrity		■	■			
Reducing the source term				■		
Reducing the uncertainty on environmental impact assessments					■	
Understanding the evolution of complete severe accident scenarios						■
Improving the emergency preparedness and response					■	

3.1.2.1 Objective: Increased efficiency of cooling a reactor core

The major safety objective is to cool a reactor core by water addition as a means of limiting or terminating the SA progression. Substantial knowledge now exists concerning cooling of a large intact and rod-like geometry. The main R&D objective is to address the remaining uncertainties or possibly close issues concerning the efficiency of the degraded core cooling.

The highest priority R&D issues are: debris bed formation and cooling; corium pool coolability in the Reactor Pressure Vessel (RPV) lower head, especially for BWRs with presence of control rod and instrumentation guide tubes; RPV external cooling conditions with evaluation of the critical heat flux.

3.1.2.2 Objective: Preservation of the containment integrity

3.1.2.2.1 Ex-vessel corium interactions and coolability

The major safety objective is to preserve containment integrity, against both rapid failure (steam explosions, Direct Containment Heating or DCH) and slower basemat melt-through and/or containment over-pressurization.

The highest priority R&D issues are: fuel-water premixing and debris formation, complementary research on Molten-Core-Concrete-Interaction (MCCI) (oxide-metal layer interaction, reactor concrete compositions, top flooding) and finally analytical work to transpose MCCI experiments to reactor scale.

3.1.2.2.2 Containment behaviour, including hydrogen explosion risk

The containment represents the ultimate barrier to prevent or limit the release of fission products (FP) to the environment. If local concentrations of combustible gases (hydrogen and carbon

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monoxide) occur, gas combustion might occur and cause a pressure increase that could eventually cause containment failure.

The highest priority R&D issues concern the containment atmosphere mixing and gas combustion (including BWR containments with nitrogen atmosphere): gas distribution, accounting for mitigation systems, regimes of deflagration and of deflagration to detonation transition. Scaling (qualitative and quantitative) of phenomena from experimental facilities to actual containment should also been addressed in priority.

Most efforts in the short and mid-term should focus on extensive simulations using both Lumped-Parameter and CFD (Computational Fluid Dynamics) codes in order to interpret a whole set of different experiments with consistent models. Reliable models of deflagration and deflagration-to-detonation transition should be developed in order to improve the present modelling mainly based on empirical correlations.

3.1.2.3 Objective: Reducing the source term

The source term to the environment refers to the amount, chemical speciation and isotopic speciation of all radio-elements that can be released to the environment. At present, the increased safety requirements in both existing and new NPPs aim at reducing the source term by proper measures for limitation of uncontrolled leaks of the containment and for improvement of filtering efficiency of containment venting systems. In particular the Fukushima accident underlined the need for studying the impact on the source term of the filtered containment venting systems that are important radionuclide removal processes.

The highest priority for R&D concerns: impact of filtered containment venting systems on source term and development of improved devices; oxidizing environment impact on FP release from fuel, in particular for ruthenium (e.g. air ingress for high burn-up and mixed-oxide or MOX fuels); high temperature chemistry impact on FP behaviour in the RCS; and containment chemistry impact on source term, mainly for reducing the uncertainty on iodine source term.

3.1.2.4 Objective: Reducing the uncertainty on environmental impact assessments

The impact of severe accidents on the environment in the near-field[5] around the NPP must be assessed as part of the NPP Environmental Impact Assessment (EIA) in accordance with the European and national legislations and because near-field contamination can directly affect the definition or the implementation of the accident management strategies.

The objective is to reduce uncertainties on all phenomena (and on models and space/time discretization), either in-reactor or in near-field, leading to the atmospheric, on-ground and underground (liquid release) contamination of land from inside the plant to the near-field, including the impact of mitigation measures. Such progress will increase the importance of considering environmental measurements in the assessment of the in-reactor accidental situation diagnosis. Determining the "acceptable uncertainty" of a source term composition for these EIA studies might also impact the research activities the other subareas.

[5] What is called “near-field” here is defined as the area that is for most of the countries under the responsibility of the NPP operator in case of an accident involving monitoring in environmental fluids (air, water), soils and surfaces located in the vicinity of the NPP and that would be first impacted. This area is thus the distance from the facility below which measures to mitigate environmental releases can be implemented or below which radioactive releases can affect implementation of SAM operations

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The main highest R&D priorities concern: account for chemistry of radio-elements in atmospheric, on-ground and under-ground dispersion models and on/under-ground water dispersion models; inverse modelling from on-site environmental measurements in order to reconstruct a plausible source term and in-reactor scenarios.

3.1.2.5 Objective: Understanding the evolution of complete severe accident scenarios

The integral codes (or system codes) are essential to simulate the SA complete scenarios up to the evaluation of the source term in the environment, as well as to evaluate SAM measures and the efficiency of mitigation systems. The high priority is to continue to capitalize existing knowledge in these codes, in priority the ASTEC code (IRSN-GRS), and to ensure a rapid feedback of the Fukushima accidents interpretation in the next years. Attention should be paid in particular to models of BWR core degradation and to their validation.

The Fukushima accidents have also underlined the importance of the need of new instrumentation for SA diagnosis and management and of the behaviour of spent fuel pools in case of loss of cooling. The applicability of integral codes, in priority ASTEC, to SFPs should be improved. Further R&D in support to SFPs must focus on large-scale flow convection, situations of partial dewatering of fuel assemblies, and clad mechanical behaviour in an air-steam atmosphere. Another challenge is to investigate the re-criticality risks in case of spent fuel pool dry-out or of damaged NPP core.

3.1.2.6 Objective: Improving the emergency preparedness and response

In the emergency phase, which comprises a threat phase and a release phase, it is mandatory to implement recovery actions to protect the population around the plant. Capabilities to overcome the accident require the availability of reliable information from the NPP and of efficient and reliable tools and methods. An identified objective concerns the wider use of PSA level 3 which are still poorly addressed in Europe.

The main highest priorities for R&D concern: improvements of fast methods and tools to assess the actual situation of the accident course and to predict source terms; improvements of methods and tools for better appraisal of uncertainty analyses in support of probabilistic safety assessments.

3.1.3 State of the art

During the last 30 years considerable amount of knowledge has been accumulated on SA phenomenology through applied R&D carried out mostly out-of-pile, with the corollary of a few in-pile programmes like Phébus FP (Fission Products) [TA2-4], and theoretical simulations, as the accidents at TMI2 in 1979 and at Chernobyl in 1986 were the only major NPP reference cases until the Fukushima-Daiichi event.

From 2004 to 2013, the state of the art has been periodically updated in the frame of the SARNET network (Severe Accident Research NETwork of Excellence) [TA2-1, TA2-2], coordinated by IRSN and co-funded by the European Commission in the 6th and 7th Research Framework Programmes (see www.sar-net.eu). It is summarized in the Sections of sub-areas 2.1 to 2.4 and 2.6. The ranking of R&D priorities was first built in the EURSAFE FP5 project [TA2-5] using a PIRT approach (Phenomena Identification and Ranking Table) with the support of more than 100 severe accident experts from 20 organizations (see Annex of TA2 section). This ranking has been reviewed in 2012-2013 in SARNET to take into account the first feedback of the Fukushima-Daiichi accident [TA2-3] (and such update will continue in the next years according to the progress of the understanding of this accident). The identified challenges account also for the results of all past and on-going international programs

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(Euratom-FP7, OECD/NEA, ISTP…). A non-exhaustive list of projects, facilities and codes relevant to this area is shown in Tables 1 and 2 in Annex of TA2 section.

For the sub-area 2.5, significant efforts have been made in the last decades to improve models of in-reactor airborne source term and a clear roadmap exists to predict this airborne source term evolution from the short to the long terms (see sub-area 2.4). The predictability of the evolution of the liquid source term did not reach a similar level since R&D efforts remain to be performed to predict adequately the evolution of the chemical conditions and source term of the in-reactor liquid phase.

Research efforts in atmospheric dispersion modelling in the last decades have produced models and some of them are even used in preoperational or operational framework in case of a radiological emergency. The major platforms devoted to studies and operational preparedness and response (DSS: Decision Support System) can perform various tasks at various degrees of integration, such as atmospheric and water dispersion at different spatial scales, manage criteria for the post-accident phase, and improve parameterization through data assimilation and urban, natural and agricultural situations down to assessing impacts via dose estimates. The Fukushima-Daiichi accidents have prompted a debate on extending activities on PSA Level 3.

The current approach for emergency preparedness and response largely results from lessons learnt after Chernobyl and it is currently applied consistently to the radiological impact of the Fukushima events. EC launched the NERIS platform for evaluation and benchmarking of emergency preparedness and response across the EU. It has resulted in proposals for the improvement of the cooperative international system of emergency preparedness and response, focusing on adopting best practices, revealing weak points and bridging the gaps.

3.1.4 Challenges

The main challenges were identified under the first 4 sub-areas: in-vessel corium/debris coolability, ex-vessel corium interactions and coolability, containment behaviour and hydrogen explosion risk, source term. Experimental efforts will be needed for all these issues, accompanied by modelling development and validation.

The 5th sub-area "severe accident linkage to environmental impact and emergency management" is specific since aiming at interfacing the nuclear safety community with the radioprotection one (mainly composed of NERIS and ALLIANCE platforms) and addresses cross-cutting R&D issues where promising synergies can be envisaged.

The 6th sub-area "severe accident scenarios" is more transverse: it addresses gaps and identifies actions essential to capitalize knowledge, such as integral simulation codes, databases of experimental data and instrumentation for SA diagnosis, and to spread knowledge and expertise.

The listed challenges will account for the results of all past, on-going and future international programs (OECD/NEA, ISTP…) and for the impact of the further analysis Fukushima-Daiichi accidents. Furthermore, all R&D outcomes will be useful for the plants decommissioning in the future.

In conclusion, the new R&D projects to launch in the upcoming years should clearly focus on the efficiency of mitigation systems (such as filtering systems, venting systems or recombiners in the containment) and on the engineering features in terms of improvement/optimization/innovation. The knowledge gained and the modelling improvements will allow the SAM optimization.

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The Table 3 in Annex of TA2 section presents the ranking of priorities (high, medium, and low) for the short term (0-2 years).

3.1.5 Interactions with other Areas

There are interfaces with TA1 where the PSA methods and tools are developed, TA3 about criticality, TA4 about the dynamic and static behaviour of containment, and TA5 about SFPs.

3.2 Sub Technical Areas (STA)

3.2.1 In-vessel corium/debris coolability (STA 2.1)

3.2.1.1 Scope

If the core remains for some time without water, there is an increase in local fuel rod temperatures which may eventually provoke significant and irreversible core degradation. The mechanisms producing such degradation may be chemical or mechanical. Depending on the temperature attained locally, the consequences may be more or less severe: hydrogen production, fission product release, formation of molten corium and propagation towards the vessel lower head. This sub-area addresses the efficiency of cooling reactor core structures and materials during SA, either in the core or in the vessel lower head, so as to terminate or limit the progression of the accident.

3.2.1.2 State of the art

The efficiency of cooling reactor core structures and materials during severe accidents either in the core region or in the vessel lower head is a key issue to limit (or even stop) the progression of the accident. This can be achieved either by ensuring corium retention within the reactor pressure vessel (RPV) or at least by limiting the corium progression and, consequently the rate of corium release into the reactor cavity. These issues were covered within the scope of accident management for existing reactors and within the scope of design and safety evaluation of future reactors. The specific objectives addressed the creation of a database on coolability of degraded cores, debris formation, debris coolability and corium behaviour in the lower head, development and validation of models and computer codes for simulation of in-vessel debris bed and melt pool behaviour, performance of reactor scale analysis for in-vessel corium coolability and assessment of the influence of SAM measures on in-vessel coolability.

Available data on degraded core reflooding are summarized in the reflood map developed at KIT, which is being constantly updated to include findings from previous and current R&D on the coolability of the reactor core, taking into account available coolant mass flow rates and core damage state. Substantial knowledge and understanding of the phenomena governing the coolability of intact rod-like reactor core geometry was obtained in numerous national and international projects. However, major gaps in knowledge still exist in the areas of debris and molten pool behaviour.

There is a significant progress on simulation of in-vessel core coolability with 2D/3D simulation codes, focusing on the enhanced coolability of debris beds by lateral and/or bottom water inflow. These phenomena are being investigated in a number of experimental programs including study of multi-dimensional effects on debris bed coolability (see Table 1 in TA2 Annex), either in SARNET frame or in the FP7 experimental platforms, LACOMECO (KIT) until 2013 and now SAFEST and ALISA, both starting in 2014 for 4 years (the latter involving Chinese partners with KIT and CEA). They are accompanied by numerous efforts on modelling in simulation codes (see Table 2 in TA2 Annex)

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within SARNET and benchmarking activities among these codes (e.g. OECD BE-TMI2 benchmark). The final SARNET/FP7 synthesis reports that were delivered mid-2013 have summarized the state of the art on this subject.

Regarding the applications of different mechanistic and integral codes to different reactor designs, the work is mainly focused on core degradation, melt relocation to the lower plenum, quenching of corium by residual water, re-melting of debris beds and molten pool formation in the lower head during severe accident sequences for different LWR designs. Significant attention was paid to in-vessel coolability during different accident stages and specifically to stabilization and localization of a volumetrically heated molten pool in the RPV lower head, with application to external reactor vessel cooling. The last point is considered as an ultimate goal of SAM. Achieving the in-vessel corium retention, i.e. a stabilized state when the decay heat from the molten pool is removed through the RPV wall, is a promising SA strategy for reactors of relatively small size where low thermal loads on reactor wall are expected. Different levels of knowledge and different modelling tools exist currently to analyse different stages of SA progression. Analysis of a typical severe accident sequence starting from front-end thermal-hydraulic phase and ending with a developed molten pool in the RPV lower head represents a challenging task from the modelling point of view. Several SA computer codes like ASTEC or MELCOR are being applied and validated in reactor applications providing an important feedback to code developers.

Recognizing the impact of the Fukushima-Daiichi accident, Collaborative Research Projects (CRP) are planned to be launched by IAEA within the Technical Working Group on Light Water Reactors (TWG-LWR) including non-European participants like Japan and Russian Federation.

3.2.1.3 Challenges

The main R&D objective will be to reduce or possibly solve the remaining uncertainties on the efficiency of cooling reactor core structures and materials during severe accidents, either in the core or in the vessel lower head, so as to terminate or limit the progression of the accident. This can be achieved either by ensuring corium retention within the RPV or at least by slowing down or limiting the corium progression into the containment. These issues are related to operational objectives: for existing reactors, SAM optimization, safety evaluation and design improvement; for Gen III reactors, SAM definition, and evaluation of design and safety.

Experimental and modelling efforts will concentrate on the formation and cooling of debris beds and molten corium pools in order to demonstrate effective cooling modes and rates and coolability limits. Modelling efforts will aim at assessing and validating the models in integral (or system-level) and detailed codes for core degradation, oxidation, debris and molten pool behaviour.

The specific R&D objectives are to create and enhance the database on debris formation, debris coolability and corium pool behaviour in the lower head, to develop and validate the models and computer codes for simulation of in-vessel debris bed and melt pool behaviour, to perform reactor scale analysis for in-vessel corium coolability and to assess the influence of SAM measures on in-vessel coolability.

The work should comprise experimental and modelling activities with strong coupling between the tasks. Moreover, since most of previous SA research topics were focused on PWR design and majority of SA codes were developed for PWRs, there are clear needs of improvement and

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optimization of BWR specific models to permit accurate Fukushima-Daiichi accident analysis and validation against existing or future experiments.

3.2.1.4 In-vessel hydrogen generation

At temperatures above 1300 K, Zr in the fuel cladding is exothermically oxidised by steam, the released energy is comparable to that of decay heat and the reaction rate varies exponentially with temperature. The hydrogen produced during the Zr oxidation may escape from the reactor coolant system and mix with containment atmosphere, which may lead to explosion and endanger the containment integrity. During the core reflooding a risk remains that further degradation occurs before quench is achieved, possibly due to rapid steam formation blocking water entry or thermal/mechanical stresses on the damaged components, to the extent that quench is impeded. This can result in temporarily higher temperatures, accelerated degradation and hydrogen generation due to increased oxidation. The ability to evaluate hydrogen production (instantaneous and cumulative) is a key issue in safety analysis.

A significant effect of nitrogen on the clad oxidation and hydrogen release kinetics has been recently discovered in KIT experiments. This effect is closely related to possible future R&D on the SFP behaviour (see section 2.6.3).

3.2.1.5 Hydrogen generation during reflooding

Cladding rupture is still poorly understood and experimental efforts are required. It is a critical phenomenon because it can result in increased oxidation of the newly exposed inner Zircaloy cladding surfaces, but it can also curtail the local oxidation by relocating liquid Zircaloy mixtures to lower and colder elevations. However, the experiments required to improve the level of knowledge are both difficult and expensive (use of real materials is necessary).

Corium moving down through the core contains Zircaloy which has not been completely oxidised and which will rapidly oxidise upon contact with steam. There is a general understanding of the phenomenon, but there are insufficient available data for oxidation kinetics of molten U-Zr-O mixtures. However, past integral tests in 90’s such as CORA (KIT) have demonstrated substantial hydrogen production over a very brief period for scenarios involving reflooding or a local increase in steam flow rate.

The main project under way in this field is the QUENCH experimental programme in KIT.

R&D topics

No needs of further R&D at short term

3.2.1.6 Hydrogen generation during melt relocation and flooding

For H2 generation during melt relocation into water like with corium jets etc. the only data available come from past experiments at JRC/Ispra (FARO) and they must be reinvestigated analytically. The issue has high risk significance because hydrogen detonation could endanger the containment integrity. Since the fundamental knowledge is available, research is needed for confirmation and validation of models.

No new experiments are planned in the short term but, for mid-term, experiments could be proposed within DEFOR (KTH) and QUENCH facilities.

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R&D topics

Short term

- ✓ **Hydrogen generation during melt relocation into water.**

Mid term

- ✓ **New experiments on this subject could be envisaged at mid-term.**

3.2.1.7 In-vessel core cooling / RPV cooling

The fuel claddings lose most of their mechanical strength after a significant oxidation. Therefore, the fuel rods are likely to collapse and form a “debris bed” of fuel pellet fragments and cladding remnants, in particular at the time of reflooding when the mechanical thermally-induced strains occur on the rods. Compared to an assembly with intact fuel rods, such debris bed is more difficult to cool down because of its higher resistance to the progression of water inside. It is very important to determine the criteria (particle size and composition, connected porosity) for coolability of such debris bed and estimate the efficiency of various water injection methods.

If the temperature is sufficiently elevated to melt fuel, molten pools may form in the core. As the molten mass increases, the pool grows axially and radially in the core until it reaches either the baffle or the lower core support plate, and then relocates towards the lower head. Predictions relating to mass, composition and temperature of the material relocated to the lower head, as well as relocation times, are critical to the study of subsequent accident phases. Models have been developed in most codes but the levels of validation and detail are not yet satisfactory relative to the experimental data available. More representative data should be obtained to enable characterisation of pool changes over time using multidimensional rod assemblies.

It is generally assumed that the lower head is filled with water when sub-oxidised corium enters this area from the core zone. When hot corium comes into contact with residual water, steam is produced and will cause a pressure spike, or even an in-vessel steam explosion. As another risk, upon contact with corium, the vessel will undergo a local heat flux, which may be of considerable magnitude, potentially resulting in vessel rupture. There is thus the need to predict the changes corium will undergo, from its relocation towards the lower head until its cooling or transfer out of the reactor vessel. The main phenomena which can govern these changes are: corium jet fragmentation and debris formation; direct contact of corium jets with the vessel wall; steam explosion, debris bed dry-out and reflooding possibilities, formation of molten pool, natural convection in the molten pool, metal/oxide stratification in the molten pool and possible focussing effects, corium oxidation including H2 production, and dissolution of reactor vessel steel at sub-melting point temperatures.

3.2.1.8 Core coolability during reflooding and thermal-hydraulics within particulate debris

Two types of debris bed may form in the course of a core degradation sequence. The first type results from the collapse of highly oxidized fuel rods. It may affect a large part of the core, as it was observed in the TMI2 accident. The second type results from the fragmentation of melt relocating into the lower plenum, if water is present. This bed is either very compact, if there is little interaction with water, or highly porous. Both types of debris beds are formed of particles of comparable average size (between 1 and 5 mm) but the particles shape and the bed porosity may be quite

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different. In both cases, non-homogeneities of temperature and porosity induce multidimensional effects on the flows of water and steam, which make modelling more complex.

Therefore, debris coolability and more generally the prediction of water and steam flow within a debris bed are of crucial importance to predict the efficiency of water injection in the core, and the consequences on hydrogen production (but steam availability is the limiting process at high temperature). A great deal of uncertainty remains on this issue (that occurs also in ex-vessel situations).

The modelling of the process is satisfactory for situations with one-dimensional flow (corresponding to existing experimental data) but it is not assessed for large debris beds with non-homogeneities of temperature and porosity or with multidimensional effects. Therefore, experiments in larger scale on coolability in representative conditions will be necessary, increasing the necessary knowledge significantly and leading to improvement of models in the detailed and system-level codes.

The main experimental projects underway are: PRELUDE and PEARL (IRSN), QUENCH-Debris (KIT), DEBRIS (Univ. of Stuttgart), and for ex-vessel situations COOLOCE (VTT). There is a close link with other experiments that are indicated in the sub-area 2.2 (see Section 2.2.2) and devoted to debris bed formation. Corresponding activities on modelling and implementation in mechanistic codes or in integral codes, ASTEC in priority, are also done in continuity of the SARNET FP6 and FP7 projects.

R&D topics

Short term

- ✓ **Reflooding of debris beds where a new project continuing the current experimental and modelling efforts should be planned in continuity of SARNET FP7 work.**

3.2.1.9 Core coolability in lower head and integrity of RPV due to external vessel cooling

Should molten corium form a pool in the lower head, heat exchange between the pool and the vessel may provoke localised overheating (or partial melting), thereby resulting in vessel local thickness decrease and possible failure. The vessel failure time as well as the failure location and size of the breach are considered as key elements because of their role in ex-vessel accident progression.

The in-vessel melt retention aspects and the improvement of the predictability of the thermal loading and mechanical resistance of the vessel structures are a matter of high interest. For BWRs, thermo-mechanical and thermo-chemical interactions of corium with control rod guide tubes, instrumentation guide tubes and nozzles of the pumps are especially important since there may be up to several hundreds of penetrations in the lower head. Cooling of the control rod guide tubes from inside can influence significantly the behaviour of melt in the lower head, in particular for reactors with low power density and in scenarios with a small amount of relocated melt.

Two main topics are of interest for scenarios with a large pool of molten corium: heat flux to metal layer in layered melt configuration, and 3-layer configuration. One further point is the thickness of the metallic layer. This is of high priority because it is a question for the in-vessel retention concepts in new designs. The simulation of the oxide pool is quite well understood. However, the simulation of corium behaviour inside BWR-type lower heads is still open, with characteristics such as corium relocation into a deep lower head filled with water, significant thermo-mechanical loads on the vessel structures in the process of debris re-heating and re-melting, and high fraction of Zr in the core.

The database for critical heat flux and external cooling conditions needs to be improved in order to evaluate and design SAM measures for external vessel cooling with the aim of in-vessel melt retention. Especially the influence of BWR lower head penetrations on melt cooling and its influence on the external convection should be examined.

Several experiments were performed in SARNET FP7 frame, LIVE (KIT) and RESCUE (CEA), both along with modelling activities. New LIVE and RESCUE experiments are planned in the SAFEST FP7 project. The CORDEB project has been launched in 2013 in collaboration between NITI (Russia) and four French organizations (IRSN, CEA, EDF, AREVA NP). The APRI-MSWI project is currently running at KTH (Sweden).

R&D topics

Short term

- ✓ **Corium coolability in lower head and integrity of RPV due to external vessel cooling where new projects continuing the current experimental and modelling efforts should be planned in continuity of SARNET FP7 work.**

3.2.2 2.2 Ex-vessel corium interactions and coolability (STA 2.2)

3.2.2.1 Scope

Corium will eventually flow and be dispersed out of therapy, if the reactor vessel fails. At this stage, the safety objective is to ensure that the containment integrity is maintained and is not challenged by corium – both in the short term (prevention of large early releases of radioisotopes) and in the long term. A different phenomenology must be considered depending whether the reactor pit is dry or filled by water (either in case of postulated failed in-vessel retention or as a SAM measure[6]).

Corium can interact with water (Fuel Coolant Interaction or FCI), with concrete (Molten Core Concrete Interaction or MCCI), with air (Direct Containment Heating or DCH) or with a dedicated core catcher. Due to the radioactive decay heat, corium must be cooled in order to arrest its progression and protect the containment, but this cooling must not be too rapid to avoid over pressurization of the containment (FCI, DCH…).

3.2.2.2 State of the art

Many experiments on this issue are being performed in international projects (see Table 1 in Annex); either in SARNET frame or in the FP7 experimental platforms PLINIUS (CEA) and ALISA. They are accompanied in SARNET by large efforts on modelling in simulation codes (see Table 2 in Annex) and benchmarking activities among these codes. Other R&D efforts are based on ANL (USA) CCI experiments. A state of the art report on MCCI is under way in OECD/NEA frame, with a significant contribution by the SARNET network. Concerning FCI, the OECD/NEA project SERENA2 (that finished

[6] e.g. in Nordic BWRs the drywell pedestal is partially flooded to ensure corium debris coolability.

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in 2013) has provided a large amount of experimental data on steam explosion, including advanced visualization of premixing (i.e. the explosion initial conditions).

The final SARNET/FP7 synthesis reports that were delivered mid-2013 have summarized the state of the art on the subject of ex-vessel corium interactions and coolability.

Globally, an important database exists on the phenomena occurring in non-mitigated ex-vessel phenomena (steam explosion, DCH, spreading molten core concrete interaction, debris bed formation and coolability) and for the support of commercially available core catchers. Nevertheless, there are some remaining lacks of knowledge that create uncertainties on the transposition from laboratory scale to reactor scale and must thus be toppled.

Recent focus on in-vessel retention by cavity flooding leads to new configurations (pressurized fuel directly entering coolant, MCCI starting under water) that have not previously led to thorough experimental and analytical R&D programmes. Another potential point of short term R&D needs is linked to the qualification of new core catcher designs. Finally, the lessons from Fukushima will become a significant part of ex-vessel corium R&D if vessel melt-through is to be confirmed. As a first issue, the effect of non-demineralised water for corium cooling has to be assessed.

3.2.2.3 Challenges

Corium R&D is based on a combination of modelling, analytical experiments with simulant materials and experiments with prototypic corium. The latter are necessary both for phenomenological analyses and for model validation. There is a need to federate the European corium facilities and to build a larger scale corium facility[7] .

A big challenge will be to improve the transposition of R&D from laboratory to reactor scale. In the medium to long term, progress is expected from the analysis of Fukushima corium, if calculations indicating vessel melt-through are confirmed.

The highest priority R&D topics are: fuel-water premixing and debris formation, complementary research on MCCI (oxide-metal layer interaction, reactor concrete compositions, top flooding) and finally analytical work to transpose MCCI experiments to reactor scale.

3.2.2.4 Melt release and FCI or DCH

The melt release and FCI or DCH are related to the transient phases following the vessel rupture. If the reactor pit is filled with water, there can be an interaction between corium and water, which can be explosive (steam explosion) or not. In any case, jet fragmentation forms particulate debris which will have to be cooled in the longer term. If the vessel is pressurized and not surrounded by water, the pressurized melt ejection may lead to direct containment heating (coupled to rapid hydrogen generation).

3.2.2.5 Corium release following vessel failure

The predictability of the RPV failure mode and the location of the corium release into the cavity/containment should be improved. Different modes of failure may exist for PWRs and BWRs. The information on the break time and location is regarded as sufficient for PWRs, but there is no

[7] European large scale corium facilities, FARO and COMAS, have been dismantled in the late 1990s. Currently facilities dealing with more than 100 kg of corium are only operated in the USA (CCI at ANL) and South Korea (VESTA at KAERI).

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adequate prediction of the break size. It is determined by the initial size and rate of the ablation, which depends on melt temperature, composition and velocity. The initial leak size is a significant starting condition for possible DCH or for ex-vessel FCI but, due to the leak ablation during the outflow, this lack of knowledge is regarded as a R&D subject of low priority.

In the short term, the SARNET work on BWR RPV failure location, timing and mode should continue (this work is closely linked to the necessary efforts to improve the modelling of corium behaviour in BWR lower heads: see section 2.1.2). In the longer term, there will be interest for the resolution of the issue of crack evolution during and between corium pours when progress in fracture mechanics gives sufficient probability of success.

R&D topics

No needs of further R&D at short term

3.2.2.6 Fuel Coolant Interaction including steam explosion and particulate formation

If the reactor pit is filled with water, the corium transfer in the pit will form particulate debris by jet fragmentation. It is important to characterize the jet fragmentation and the formed debris bed (particle size distribution, geometrical repartition, chemical state) in order to be able to evaluate its coolability. Compared to previous work, for PWRs focus must be put on lateral jets (due to a side ablation of the vessel by convective heat fluxes). For PWRs and BWRs, focus should be put on jets into a water pool (deep in BWRs).

When there is contact between two fluids, with one at a temperature higher than the boiling point of the other (the coolant), an explosive interaction may be triggered. The steam explosion global phenomenology has been relatively well understood since the 1970s. This phenomenon involves the unstable coupling of the two mechanisms, energy transfer from the hot fluid to the cold fluid through fragmentation, and pressurization and relative flow between the fluids, inducing fragmentation.

R&D topics

Short term

- ✓ **Continuation of R&D on FCI premixing phase, in particular on fragmentation of corium jets into a water pool (deep in BWRs), in order to provide reliable initial conditions for the steam explosion phase.**

Mid or long term

- ✓ **Improvement of the explosion models at mid or long term on the basis of improved knowledge on premixing.**

3.2.2.7 Direct Containment Heating (DCH)

A jet of corium and steam can reach the containment and over-pressurize it by heating its atmosphere, if the vessel is pressurized at failure. This issue is design-dependent and has been investigated thoroughly for several plant designs (EPR, French PWR and VVER-1000, Konvoi…) in the DISCO facility at KIT. Based on the scaled experiments the hazard of overpressurization was not more regarded as an issue for the abovementioned European designs. DCH phenomena in BWR geometry have to be addressed in the future studies.

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Using the DISCO data basis the development of DCH models suited for SA integral codes like ASTEC has made good progress but the available model approaches have not yet been applied to a reactor-scale scenario. The complex behaviour of hydrogen burning under DCH conditions (burning of hydrogen for the situation of a mixed hydrogen/steam/melt-jet entering a premixed atmosphere) is regarded as a weak point in the models. Further, melt entrainment as one of the key phenomena in the DCH models is validated only for the specific designs investigated.

In the mid-term, analytical and experimental efforts are still needed on the scaling effects and on hydrogen combustion during DCH.

R&D topics

Mid term

- ✓ **DCH scaling effects and hydrogen combustion during DCH, in particular for BWR.**

3.2.2.8 MCCI and corium catchers

The concrete basemat in the reactor pit is part of the third and last barrier for most Gen.II reactors. It can also contribute to temporary retention before transfer to an ex-vessel core catcher in several Gen.III reactors (e.g. EPR, VVER 1000 AES 2006, ESBWR, EU ABWR…) that aims at preventing the ablation of containment concrete and possible melt-through by a corium pool.

There are two major safety issues:

1. Will the concrete basemat ablation last long enough to prevent large early releases?
2. Will concrete ablation stop before basemat leak, or will the core-catcher be able to keep (indefinitely) the corium inside the containment?

Answering to these safety issues is reflected in the following 5 R&D topics.

3.2.2.9 MCCI molten pool configuration and concrete ablation

As a corium molten pool delivers a significant part of the decay heat to the bottom, there is little chance to arrest melt progression without cooling. It is nevertheless necessary to study concrete ablation and molten pool configuration, as means to verify that there is no risk of early basemat melt-through, and to provide the initial conditions for a cooling phase or to a transfer to a core catcher.

Current 2D oxidic pool MCCI experimental programmes have shown that limestone-rich concretes are almost isotropically ablated while for silica-rich concretes lateral ablation was much larger than vertical ablation. This reproducible behaviour is not yet understood so its transposition to reactor scale or to other concrete compositions[8] (e.g. basaltic) is subject to caution. Furthermore, recent experiments with oxidic and metallic pools[9] have shown phase repartitions which are different from simple-layers assumptions considered in MCCI codes (emulsion or gravity stratification).

In the recent years, international R&D was mostly performed in the frame of the SARNET/FP7 network, mainly on the basis of the VULCANO (CEA), MOCKA (KIT), HECLA (UJV) and SICOPS (AREVA

[8] For instance, the limit in terms of concrete composition between the isotropic and anisotropic ablation behaviour is still unknown.

[9] In which most of the power is prototypically injected into the oxidic phase.

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GmbH) experiments. New experiments are planned in most above facilities in the frame of the SAFEST FP7 project. An OECD/NEA/CSNI state-of-the report on MCCI is also in progress, including a contribution from SARNET/FP7 on MCCI in dry conditions.

R&D topics

Short term

- ✓ **Complements to the MCCI 2D experimental database with concrete compositions that have not been evaluated (e.g. basaltic concretes mainly used in Japan and in the USA, iron oxide concrete as used in EPR, concrete reinforcement) and analysis of the experiments in simulant materials investigating the 2D convective heat transfer distribution.**

Mid term

- ✓ **Analytical work using the whole experimental database (real material and simulant experiments) is necessary to understand the 2D ablation in an oxidic layer and also in the oxide-metal configurations. It will be also useful to complete the understanding of metal oxidation during MCCI. Additional data from MCCI experiments are also needed for the MCCI long term (i.e. time higher than 1 day of a reactor scenario) with elevated concrete fractions and reduced heat fluxes to the concrete interface.**

Long term

- ✓ **If a faster melt-through due to pool stratification with metal below cannot be discarded taking into account the analysis of the whole experimental database, it will be necessary to carry on large scale MCCI tests with metal and oxide pools as a necessary step to complete the understanding of the oxide/metal pool configurations and to validate the transposition from laboratory scale experiments to reactor scale for improving the modelling of pool configuration evolution and reducing its conservativeness.**

3.2.2.10 Ex-Vessel corium coolability and top flooding

The knowledge of cooling mechanisms by top flooding of the ex-vessel corium pool needs to be increased, because it influences significantly the calculated results, if the basemat erosion stops or will not decisively affect the containment integrity. The uncertainties are large and some experiments in the OECD MCCI2 project suggest that early[10] top flooding could be an efficient means of arresting concrete ablation.

R&D topics

Short term

- ✓ **Corium coolability by top flooding, including heat transfer between the upper crust and the water above through water ingression and melt eruption phenomena, which will require experimental work both with oxidic pool and mixed oxide-metal melts.**

Mid term

[10] i.e. before mixing of a significant fraction of silica with corium, dramatically decreasing its cracking ability

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- ✓ **Depending on the results of these short term activities, further research might be needed to check the efficiency of top flooding at a large scale. If the efficiency of top quenching deduced from future coolability experiments appears to be limited, further R&D should be done on corium bottom flooding.**

3.2.2.11 2.2.2.3 Ex-Vessel corium catcher: coolability and water bottom injection

The efficiencies of different corium catcher designs and the efficiency of water bottom injection were demonstrated for Gen.III applications, thanks to experiments in KIT (so-called COMET concepts), ANL, University of Wisconsin (USA), KTH and CEA and modelling mainly at IKE and University of Wisconsin. However the understanding of basic cooling mechanisms and the impact on the cooling efficiency and homogeneity of the injection site density, the sacrificial concrete layer, the corium height and properties and the detailed geometry of the design has to be still completed in order to permit the extrapolation to the reactor case with different concept variants. Moreover in the view of Gen.II plant back-fitting, the technical solutions might need to be modified for implementation in a highly radioactive environment.

R&D topics

Short term

- ✓ **Continuation of research but rather in the frame of specific bilateral projects (with possible extension in the mid-term). The focus should be put on the risks of coolant bypass and loss of efficiency due to cooling heterogeneity, the control of the maximal steam rates and the elimination of steam explosion risk. A few experiments will be necessary in support of this R&D but their definition will require the detailed analysis of the existing experimental database and the identification of shortcomings of past experiments.**

3.2.2.12 Ex-Vessel corium catcher: corium ceramics interaction and properties

The simultaneous interaction of corium with ceramic and concrete must be reassessed in view of the recent results of corium-concrete interaction. The issues of heat transfer from corium to the sacrificial material and to cooling channels below and of heat transfer from corium to water pool above are of high importance. More generally the robustness of the core-catcher functionalities (corium collection, fast pouring and spreading and finally long term cooling) should be carefully assessed for a large variety of core degradation scenarios with multiple (and delayed) corium pours.

Moreover, new designs (in terms of geometry and materials) are being proposed and should be assessed but this may lead to the problem of industrial confidentiality.

R&D topics

Short term

- ✓ **Validation of some core-catcher designs with corium ceramics interaction, but rather in the frame of specific bilateral projects.**

3.2.2.13 Ex-Vessel debris bed formation and coolability

In Nordic-type BWRs and some other PWR and BWR designs, corium melt is ejected from the vessel into a deep pool of water. It is expected that the melt will be fragmented during interaction with

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coolant and will form a debris bed coolable by natural circulation. However coolability of such bed is contingent on the properties of the debris bed such as particle size distribution, porosity, spatial configuration. There are inherent uncertainties in the debris bed formation process which include: melt jet break-up and hydrodynamic fragmentation of liquid droplets; agglomeration of partially liquid debris and formation of non-coolable cake; oxidation of metallic component of the melt; fracture of solid debris; spreading of the debris by large scale convective flows in the water pool; self-levelling of the debris bed due to boiling; coolability of settled debris bed.

Given the uncertainty in the scenarios of melt release, there is significant uncertainty in the resulting configuration of the debris bed and thus in its coolability. Further experimental and analytical investigations are necessary in order to build adequate models for prediction of the debris bed formation phenomena and clarify the uncertainties.

In the recent years, international R&D on debris bed formation was mostly performed in the frame of the SARNET network, on the basis of the DEFOR and POMECO (KTH) and KROTOS (CEA) experiments. New experiments in the above facilities are planned in the frame of the SAFEST FP7 project. There is a strong link with the section 2.1.2 on in-vessel debris bed reflooding.

R&D topics

Short term

- ✓ **Ex-vessel debris bed formation and characterization, in particular for BWR situations. A new project continuing the current experimental and modelling efforts should be planned in continuity of SARNET/FP7 work, in close collaboration with the debris bed coolability issue.**

3.2.2.14 Effect of impurities in water

Since this issue concerns also the sub-area 1 for in-vessel situations, it should be addressed in close collaboration between both sub-areas.

During the Fukushima SA, raw sea water has been used to cool the reactor cores, although this had never been considered in previous SA studies. Since it is not possible to exclude totally the use of untreated water (sea water but also hard -calcareous- water, or dirty water pumped somewhere in the NPP) during a postulated future SA, investigations must be conducted to define possible R&D needs. In particular, studies should include the effects on corium chemistry (low temperature eutectics or volatile species, e.g. effects of chloride) and the study of precipitate formation[11] . The effects of impurities on FCI[12] should be studied too.

R&D topics

Short term

- ✓ **Effects of "untreated water" on corium chemistry.**

Mid term

- ✓ **Effects of impurities on FCI.**

[11] Salt or limestone precipitate could block coolant flow.
[12] Which has been observed for concentrated brines in volcanology.

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3.2.2.15 Corium thermo-physical and thermodynamic properties

Corium modelling requires data on corium physical properties and on corium phase diagrams (which result from its chemical thermodynamic properties). Due to the high temperatures (1500-3300 K) of corium and the large range of corium compositions with a large number of elements, insufficient data exist on corium constituent properties at least for some particular compositions. For corium phase diagrams, great work has been performed and is available thanks to NUCLEA data base which still needs to be completed for several composition values. However, some properties such as viscosity or surface tension are not sufficiently known.

R&D topics

Mid term

- ✓ **R&D continuation to improve the existing experimental database in order to provide validated data for severe accident codes.**

3.2.3 Containment behaviour, including hydrogen explosion risk (STA 2.3)

3.2.3.1 Scope

The containment of a nuclear power plant represents the ultimate barrier that is expected to prevent (or at least limit) the release of fission products to the environment during a SA. It should be able to withstand the maximum pressure caused by possible combustion of gases such as hydrogen or carbon monoxide generated by oxidation of the reactor core materials and eventually later by MCCI. If, in some region of the containment, the respective concentrations of hydrogen, steam and oxygen are within certain limits (so-called flammability limits), hydrogen combustion might occur (in safety analyses it is always assumed that ignition would almost certainly occur due to the many possibilities of random sparks).

However, if containment failure in the form of cracks or controlled containment venting occurs, it will be necessary to determine the release of fission products to the environment.

3.2.3.2 State of the art

The basic phenomenology of gas distribution and combustion is well understood from results of experiments that were performed under simple influences or in simple geometries. However, the knowledge on mechanisms of gas distribution under complex influence (such as the simultaneous action of multiple mitigation systems) and of gas combustion in complex geometries (such as a network of compartments) is lacking. Theoretical simulations with computer codes on different length scales (i.e. both Computational Fluid Dynamics or CFD codes and lumped-parameter or LP codes) have shown that it is indeed possible to simulate gas distribution and combustion in the containment. However, the results of simulations, performed for a system for which no experimental data are available, still cannot be trusted enough and are mostly considered as speculations. That is especially true for actual containments, as the current knowledge is based (directly or indirectly) on experiments in facilities that are some orders of magnitude smaller.

Concerning the material transport aspect of the containment static and dynamic behaviour, research has been performed on the flow rate of gases, liquids and aerosols through cracks. However, further research on these issues is still necessary.

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Many experiments have been performed in the containment domain in the frame of SARNET FP7 (see Table 1 in TA2 Annex). They were accompanied by large efforts on modelling in simulation codes (see Table 2 in TA2 Annex) and benchmarking activities among these codes. Other R&D efforts were done in the ERCOSAM FP7 project in collaboration between European and Russian organizations (that finished in 2014).

A more detailed state of the art on specific topics is provided in the relevant sections below. The final SARNET/FP7 syntheses that were delivered mid-2013 have described in details the state of the art on the containment topic.

3.2.3.3 Challenges

The highest priority R&D issues are related to containment atmosphere mixing (including BWR containments with nitrogen atmosphere) and gas combustion, which implies the following phenomena: gas distribution in the containment, including influence of mitigation systems, pressure increase during hydrogen combustion, and deflagration to detonation transition. Scaling (qualitative and quantitative) of phenomena from experimental facilities to actual containments should also be addressed in priority.

Experiments focus today on interaction between SAM systems (including hydrogen mitigation systems) and hydrogen distribution in the containment atmosphere. Most efforts in the short and mid-term should focus on extensive simulations using both LP and CFD codes in order to interpret a whole set of different experiments with consistent models. Reliable models of deflagration and deflagration-to-detonation transition should be developed in order to replace ad-hoc criteria, based on experimental results.

A lower priority R&D issue is, in case of moderate pressure peak, the prediction of the mass flow rate of non-condensable gases, steam and aerosols through the cracks, as it will allow determining the mass release of fission products to the environment

3.2.3.4 Gas distribution inside containment

In order to assess the risk of gas combustion, the gas distribution in the containment must be determined, in particular accounting for the influence of mitigation systems such as sprays or Passive Autocatalytic Recombiners (PAR).

For hydrogen combustion to occur, local hydrogen concentration in some region of the containment should exceed a certain lower limiting value (and so should oxygen concentration; any possible steam concentration should be below some other limiting value). Thus, the hydrogen distribution within the containment may influence the occurrence of combustion. The distribution may be influenced by the following phenomena: steam plumes or jets from the primary system; circulation in the containment, induced by steam plumes or jets or other causes; convection induced by steam condensation; pressure mitigation devices, such as containment sprays, coolers or fan coolers; hydrogen mitigation devices, such as PARs; inertisation of containment, e.g. by nitrogen in BWRs.

In principle, the use of hydrogen mitigation systems is supposed to help prevention of hydrogen combustion. Using inert gas in BWR containment is a widely used mitigation means against effects from hydrogen. However, the use of mitigation systems might also cause some adverse effects, which could promote hydrogen combustion. PARs promote natural circulation and may influence the

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distribution of hydrogen in the regions away from PARs either in a favourable or in an unwanted way. Spontaneous ignition may also be generated due the high temperature of the PARs catalytic plates.

Concerning experimental research, the separate influence of the above-mentioned causes has been extensively investigated within past OECD and EURATOM projects (OECD SETH, OECD SETH-2, OECD ISP-47, OECD THAI and THAI2, SARNET, ERCOSAM) in many experimental facilities: TOSQAN (at IRSN, France), MISTRA (at CEA, France), THAI (at Becker Technologies in Eschborn, Germany), PANDA (at PSI, Switzerland) and SPOT (at Afrikantov OKB Mechanical Engineering, Russia). Although more experiments would still be useful, we may consider that a basic data base has been established, from which the phenomenology of hydrogen distribution under various separate influences could be understood.

However, the combined influence of above-mentioned causes has not been extensively investigated yet. Therefore SA relevant experiments with a combination of causes acting simultaneously should be performed. Currently, the OECD HYMERES project (2013-2016) deals with the issue of hydrogen distribution under the combined influence of SAM systems and other phenomena (such as convection or heating). Experiments on hydrogen distribution are usually performed using helium as a substitute because of safety reasons (THAI experiments showed the similarity of results with both gases). In the absence of a better alternative at present, this method should still be used. However in the long term, this assumption will need to be checked.

There is no significant priority concerning new experimental activities on gas distribution without mitigation. For recombiner operation under extreme conditions (oxygen starvation, carbon monoxide recombination, optimal PARs location…), experiments have already been performed or on-going (BMC in the past, REKO on CO effect, OECD THAI2 on starvation effect) but new experimental activities are still needed in the short and medium term.

Concerning theoretical research, extensive simulations have been performed to date, using both Lumped-Parameter (LP) codes (which consider the containment as a network of control volumes, in which conditions are modelled as homogeneous, and which are connected with flow paths) and CFD codes (which solve the basic equations of fluid mechanics on a local instantaneous scale). However, at present, the work consisted mostly in fitting simulation results to experimental data. The dispersion of results within "blind" phases of various benchmark exercises, organised within OECD or SARNET, demonstrated that, at present, simulation results of an arbitrary scenario cannot be considered yet as totally reliable. Thus these activities should be continued until a satisfactory agreement between experiments and simulations is obtained using consistent models. The available experimental results from the above-mentioned facilities should be used.

R&D topics

Short term

- ✓ **Simulation of hydrogen distribution under the influence of steam or air plumes or jets, and simulation of PARs operation under extreme conditions.**

Mid term

- ✓ **Simulation of hydrogen distribution under the separate influence of mitigation and other devices.**

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Long term

- ✓ **Simulation of hydrogen distribution under the combined influence of plumes, jets, and mitigation and other devices.**

3.2.3.5 Gas combustion inside containment

The R&D topics are related to both regimes of gas deflagration regime and deflagration to detonation transition.

3.2.3.5.1 Pressure increase during hydrogen combustion and deflagration regime

The main threat of hydrogen combustion is pressure increase as it could cause containment failure. If hydrogen combustion occurs in the form of deflagration, where the flame propagation velocity is lower than the velocity of sound, the maximum pressure might be (in principle) lower, similar or higher than the containment design pressure. Thus, the possibility to predict relatively accurately the maximum pressure during hydrogen deflagration at different conditions would contribute to lower the uncertainty of safety analyses. At present, the majority of the computer codes in use predict the maximum pressure, observed during experiments involving hydrogen deflagration, with an uncertainty of up to 0.5 bar. Although this may be considered satisfactory, the issue still cannot be considered as closed. Even if most codes predict the maximum pressure, the pressure increase rate is still not well predicted. To enhance the predictability of codes, efforts are still needed to investigate flame acceleration, deceleration and quenching.

In the case of hydrogen detonation, where the flame propagation velocity is higher than the velocity of sound, the maximum pressure would be much higher that the containment design pressure in any case. As the containment is not supposed to be able to withstand such high pressures anyway, research to determine the exact magnitude of the pressure increase is not of such relevance from the point of view of nuclear safety.

For the experimental results on hydrogen deflagration to be applicable as much as possible (see section 3.1.4) to actual containments, experiments should be performed in as large experimental vessels as possible and with a high level of instrumentation. At present, some experiments have already been performed in some relatively large vessels such as: in the past BMC (Germany), HDR (Germany), and NUPEC (Japan), and in recent years THAI (Becker Technologies, Germany) and HYKA (KIT). However, the range of conditions at which experiments have been performed is limited.

Experiments on hydrogen combustion have been performed at KIT within the EC FP7 project LACOMECO (2010-2013), and at Becker Technologies (Germany) within the OECD projects THAI (2007-2009) and THAI2 (2011-2014).

The research activities (mostly mid and long term) on pressure increase during hydrogen combustion should continue in the following directions:

- ➢ Experiments should be performed in large-scale facilities within a wide range of initial conditions (pressure, temperature, gas mixture composition, combustion under spray, combined hydrogen and other gases such as CO, CH4, combustion...). These facilities should be highly instrumented to allow measurements of turbulence level, temperature, pressure, gas composition …;

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- Existing theoretical models (incorporated in both LP and CFD codes) should be assessed using experimental results (that is, essentially the calculated pressure should be compared to the measured value) and eventually improved. Effect of turbulence and instabilities (including the impact of heat absorbing and turbulence generating structures like walking grids on combustion) should be included in new models to accurately predict flame acceleration, deceleration and quenching.

R&D topics

Mid and long term

- ✓ **Hydrogen combustion and deflagration, with needs of combined experimental and theoretical work.**

3.2.3.5.2 Deflagration to detonation transition

In principle, hydrogen combustion should be avoided (although, in some cases, hydrogen may be ignited deliberately as part of SAM, so that hydrogen combustion occurs while its concentration is still low and its mass is reduced with limited negative consequences). However, if hydrogen deflagration occurs, the maximum pressure is expected to be lower or of the same order as the containment design pressure, so that the containment integrity would not necessarily be threatened. However, if deflagration to detonation transition would occur, the maximum pressure and the corresponding impulse could be almost certainly such that the containment integrity would be threatened. Thus deflagration to detonation transition should definitely be avoided.

Concerning experimental research, some experiments on deflagration to detonation transition in relatively large facilities have been performed (as in the RUT facility at the Kurchatov Institute in Moscow, Russia). Any additional experiments would be useful. However, given the cost of such experiments, systematic experimental investigations of deflagration to detonation transition in large-scale facilities cannot be expected in the near future.

Concerning theoretical research, there is no reliable model at present, based on first principles, to predict in which conditions deflagration to detonation transition (DDT) would occur. For the time being, ad-hoc criteria, based on experimental results, are used. Thus, reliable DDT models should be developed in the short term to predict, with limited uncertainty, whether hydrogen deflagration under some conditions could lead to detonation. Later on, the models should allow designing adequate mitigation measures to avoid DDT, and finally designing adequate containments (or adapt existing ones) with reduced DDT likelihood.

R&D topics

Short term

- ✓ **Reliable models of deflagration to detonation transition.**

3.2.3.6 Scaling from experimental facilities to actual containments

Experimental results, from which the phenomenology of hydrogen distribution and hydrogen combustion is inferred, are mostly (apart from HDR facility) obtained in experimental containments with a volume of up to a few hundred m3, whereas the containments of actual nuclear power plants have volumes two orders of magnitude higher. In addition, computer codes that are used for safety

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analyses (both LP and CFD codes) are validated on experimental results. Thus the following issues are still open:

- Could a similar qualitative behaviour of hydrogen, be it distribution or combustion, be expected in an actual containment as was observed in an experimental containment?
- How can quantitative data, obtained in experimental facilities (e.g. maximum pressure during hydrogen combustion) be extrapolated to actual containments?
- Can computer codes that were validated on tests performed in experimental containments be trusted to provide reliable results for actual nuclear power plants?

The current consensus is that the scaling is addressed through simulations with computer codes. The codes are first validated on experiments, performed in large-scale (in the experimental sense) facilities, and then used to predict hydrogen behaviour (distribution and combustion) in an actual containment. This assumes that relevant basic phenomena, for which models are validated on experiments, are not qualitatively different on the scales of actual containments. For instance, natural circulation loops may be larger in an actual containment than in an experimental vessel. However, the basic phenomena (diffusion, laminar or turbulent convection) are supposed to be independent from the overall scale of the containment.

As experiments on the scale of actual containments will (probably) never be performed, the scaling from experimental facilities to actual containments will always remain open to speculation. The ERCOSAM FP7 project (involving both European and Russian facilities) has contributed towards solving the issue by addressing this scaling aspect by considering facilities of different scales.

R&D topics

- ✓ **Given the complexity of the task of scaling from experimental facilities to actual containments, it is difficult to propose at present short, medium and long term R&D priorities.**

3.2.3.7 Dynamic and static behaviour of containment

If a high pressure peak, exceeding many times the containment design pressure, and which may be caused by hydrogen detonation, occurs, eventual containment failure is supposed to occur as a large breach. In that case, the release of fission products to the environment could no longer be limited.

If a moderate pressure peak, higher than the containment design pressure and presumably caused by hydrogen deflagration, occurs, eventual containment failure is supposed to occur with the appearance of cracks. Even if the release of fission products to the environment is quite limited in this case, the central issue is the prediction of the mass flow rate of non-condensable gases, steam and aerosols through the cracks, as it will allow determining the mass release of fission products to the environment. The following aspects should be investigated but with a low priority:

- Dimensions (i.e. width) of cracks, depending on pressure and temperature in the containment, and on the material of the containment structure,
- Flow of non-condensable gases through narrow cracks,
- Two-phase (gas-liquid) flow of steam (possibly mixed with non-condensable gases) through cracks since steam might condense,

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- Aerosol transport through cracks, with the effect of crack tortuosity and of plugging by accumulation of particles.

R&D topics

- **No needs of further R&D at short term.**

3.2.4 Source term (STA 2.4)

3.2.4.1 Scope (including specific objectives)

The source term to the environment refers to the amount, chemical speciation and isotopic speciation of all radio-elements that can be released to the environment. Those radio-elements produced by the fission of the nuclear fuel can be either noble gases or reactive species. Reactive radio-elements can be either under vapour/gas or solid/liquid aerosol forms during their transport from fuel to the environment through the reactor cooling system and containment.

Noble gases cannot be easily retained or treated neither by controlled or uncontrolled leaks of the containments but innovative filtering devices are under investigation for xenon and krypton. They do not represent any risk of contamination but their contribution to the dose rate inside and in the near field of the plant will have influence on the SAMs.

In the source term, reactive radio-nuclides are classified depending on their contribution to radiological consequences. Efforts should specifically be made on iodine and ruthenium as iodine, through its 131I isotope and its organic volatile forms, and ruthenium, through its 103Ru and 106Ru isotopes and RuO4 volatile forms, will be the main contributors to the radiological consequences at short term. For the post-accident management, fission products (FP) impacting the source term in the long term such as caesium are important too.

3.2.4.2 State of the art

In the early stage of SA studies, the source term was postulated as associated with low-probability events and it was calculated using rough conservative assumptions. Since then the situation has drastically changed. On one side, unfortunate events, like Chernobyl and Fukushima, have shown that, in spite of being highly unlikely, severe accident scenarios can give rise to unacceptable source terms. On the other side, a much deeper knowledge of source term behaviour has resulted from more than 30-year research. This becomes even more significant nowadays in which safety requirements in both existing plants and new designs are much more demanding. As a consequence, the source term has become a main target of nuclear safety and additional measures to the already existing ones have been and are being investigated for a sharp reduction of uncontrolled leaks. Essentially all the source term measures could be grouped in those dedicated specifically to source term reduction and those focused on keeping containment integrity (i.e. filtered containment venting).

Significant knowledge has been obtained in the past on most contributions to the source term: release of FP from fuel (except for MOX and high burn-up fuel) and of core material aerosols (control rods, structural materials, actinides etc.), FP/aerosols transport in primary and secondary circuits (noting that chemical kinetic effects, such as those involving iodine, still need further quantitative understanding, as they can lead to increased gaseous iodine being injected into the containment), and FP/aerosol behaviour in the containment, mainly FP chemistry in containment sumps and

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aerosol depletion. Nevertheless, there are remaining issues that still need to be investigated, like the iodine-paint interactions or the FP re-entrainment for example. In other words, attention should be paid to processes capable of producing compounds hard to be filtered (as organic iodides) or those producing unexpected FP release in a longer run (i.e. vaporization of saturated pools).

The five integral IRSN experiments of the Phébus FP international programme have provided in the last 20 years a very large and important database on FP behaviour but this facility is now being dismantled. A final seminar in June 2012 summarised the main conclusions of this programme. In the current International Source Term Programme (ISTP) that finishes around mid-2015, analytical experiments are being performed to complete the knowledge. In addition many experiments have been performed in this domain in the SARNET/FP7 frame (see Table 1 in Annex). They were accompanied in SARNET by major efforts on modelling in simulation codes (see Table 2 in Annex) and benchmarking activities among these codes. These have included the completed OECD/CSNI/ISP-46 standard problem based on the Phébus FPT1 experiment, and a similar ongoing exercise being conducted in SARNET, based on Phébus FPT3.

Other R&D efforts are conducted in the frame of three OECD/NEA/CSNI projects under way: ThAI-2 (led by GRS and Becker Technologies in Germany) on iodine behaviour in a multi-compartment containment, BIP2 (led by AECL) on iodine and ruthenium behaviour, and STEM (led by IRSN) on gaseous ruthenium transport and delayed releases from iodine deposits in circuits and on long-term behaviour of iodine aerosols and interactions between iodine and paints.

The final SARNET/FP7 synthesis reports that were delivered mid-2013 might be seen as a sort of summary of the state-of-the-art on the subject of the source term.

3.2.4.3 Challenges

The main challenge concerns the development of models able to predict the behaviour of all FPs relevant to each type of accidental sequence whereas their reactivity (especially of iodine and ruthenium) and the accident conditions make representative experiments difficult to perform. Combined theoretical and experimental approaches have to be used to face this difficulty. Source term is linked to many coupled complex physical and chemical phenomena and another challenge is the determination of the issues that the R&D efforts should be focused on. Evaluations of source terms for each SA category of any nuclear installation should thus be made including analysis of uncertainties coming both from models and boundary conditions.

The highest priority for R&D concerns: oxidizing environment impact on FP release from fuel, in particular for ruthenium (e.g. air ingress for high burn-up and MOX fuels); high temperature chemistry impact on FP behaviour in the RCS; containment chemistry impact on source term, mainly for reducing the uncertainty on iodine source term. In addition, the Fukushima accident underlined the need of reducing the uncertainties about pool scrubbing that is an important radionuclide removal process.

3.2.4.4 In-vessel FP release

The in-vessel phase of the accident is characterized by release of noble gases and solid FPs from overheated fuel. FPs can be classified depending on their volatility (high for Xe, Kr, I, Cs…; low for Ba, Ru, Sr…, and significantly much lower for the others). The main uncertainties come from:

- Behaviour of Ru, Ba, Sr, and Mo,

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- Impact of mixed air/steam or very oxidizing conditions,
- Impact of high fuel burn-up on the fuel micro-structure (RIM zone),
- FP release for accidental sequences where the fuel temperatures remain quite low (lower than 1500°C as it can be the case for example for spent fuel pool dewatering or for sequences for which core cooling is recovered before the fuel melt is reached).

The impact of control rod material (Ag-In-Cd, B4C, etc.) on iodine speciation is considered to be closed in the near future (experimental data are available).

3.2.4.4.1 Core reflooding impact on FP release

As regards FP releases from fuel, limitation of fuel heat-up will lead to a limitation of FP release. Injection of water onto overheated cladding that has not been fully oxidized before can lead to a break-away and thus a temperature rise. Here classical models for FP release should apply in this situation but their applicability should be verified.

Fragmentation of the fuel microstructure and consecutive formation of debris due to the thermal shock have not been observed in the only dedicated experiment carried out in RIAR (Russia) in the frame of the ISTC-1648 project (International Science Technology Centre). This issue should be considered carefully for MOX fuels due to the specific evolution of their microstructure. In that case, an assumption of release of the FP inter-granular inventory should have to be assessed.

Possible lixiviation of FP by the coolant should also be analysed.

R&D topics

No needs of further R&D at short term

3.2.4.4.2 Oxidizing environment impact on FP release

The issue of oxidizing environment impact on FP release is of highest priority due to the impact of ruthenium on source term and radiological consequences, and also due to the insufficiently predictive current models. Models are insufficiently predictive under oxidizing conditions and peculiarly when the fuel is submitted to mixed air/steam conditions. The release kinetics of some FPs including ruthenium strongly depends on the evolution of the oxidation potential inside the fuel. Contribution of ruthenium to radiological consequences at short term to midterm would be, in case of high oxidizing potential, comparable to iodine one, what raises the priority of this issue for both power plants and spent fuel pools. The main phenomenon to consider is the oxidizing impact of the atmosphere on the fuel and then the evolution of the oxidizing potential inside the fuel that must also account to the fuel characteristics (impact of higher fission rate of high burn-up or MOX fuel). Three experiments were performed in the VERDON facility (CEA) in the ISTP frame and a last one is planned end of 2014. To develop predictive models, mechanistic modelling approaches should also be considered: atomistic and molecular dynamics approaches to compute diffusion coefficients of ruthenium inside the fuel validated with dedicated analytic experiments; coupled effects of fuel micro-structural evolution under irradiation and thermo-mechanical behaviour at the whole rod scale on FP release. Discussions are ongoing on possible new projects based on the VERDON facility.

R&D topics

Short term

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- ✓ **Oxidizing impact of the atmosphere on the fuel and on the evolution of the oxidizing potential inside the fuel that must also account to the fuel characteristics.**

3.2.4.5 Source Term in Reactor Cooling System and containment

Chemistry of iodine and ruthenium in the RCS remain of highest priority as the source term to the environment is in the early stages almost proportional to the fraction of iodine and ruthenium that reach the containment in gaseous form. The behaviour of ruthenium in the containment is considered as sufficiently well known. Remaining issues on iodine in the containment concern the production of organic iodine and the (closely linked) behaviour of iodine oxides.

The interrelation of iodine and other FP with thermal-hydraulics in the containment leads to reduce efficiently the source term but this is well modelled in codes and is considered as a low priority for R&D.

3.2.4.5.1 RCS high temperature chemistry impact on source term

Predictability of iodine species exiting the RCS has to be improved to provide the best estimate of the source into the containment. Up to now the variable iodine speciation at break measured in the Phébus.FP experiments cannot be reproduced even qualitatively by models. Analysis of interactions of many other different transported species (FP, structural materials, carrier gas) should be made with representative concentrations. Iodine speciation can be considered as a result of the chemistry of the {I, Cs, Mo, B, In, Cd, Ag, O, H} system including its possible kinetics limitations. The dedicated CHIP (IRSN) experiments, done in ISTP frame, can provide valuable information on the main mechanisms involved in iodine speciation. Combined theoretical and experimental approaches have been developed, which appears to be fruitful especially in the recent interpretations of the Phébus.FP results. This should continue to reach predictive models in the future. Experiments of iodine interactions with silver, indium and cadmium represent the main experimental needs. Results have been recently published on the theoretical determination of the kinetics of the reactions of {Cs, I, O, H} system in the gas phase. Interaction of caesium with molybdenum and boron are under investigation. Interaction of iodine with the RCS substrate is another important issue especially for delayed releases. Indeed those releases that can occur some days after the accident initial event have been the subject of little research up to now whereas they have been observed experimentally to be important.

Other FPs of high source term impact also need to be checked for volatilisation or revaporisation from surfaces in the event of rapid oxygen potential changes in the RCS expected in typical SA scenarios.

Regarding ruthenium, the experimental basis has been analysed in the SARNET FP7 framework (EXSI in VTT, CHIP in IRSN). Similar combined theoretical and experimental approach that is used for iodine should be developed for ruthenium but the reactions are limited to interaction with carrier gas and RCS substrate. The BIP2 and STEM OECD projects continue to address this issue.

R&D topics

Short term

- ✓ **Gaseous ruthenium transport and delayed releases from iodine deposits in circuits. But these topics being addressed in the OECD/NEA STEM project (2011-2015) with small-scale parametric experiments, no new project seems necessary at short term.**

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3.2.4.5.2 Containment chemistry impact on source term

Iodine behaviour in the containment is linked to a very complex heterogeneous chemistry. It has been the subject of many research efforts in the last decades. The following thermal-hydraulic phenomena have an impact on source term but they are considered now as low priority issues since much progress has been made in the recent years: multi-compartment transport, wash down of deposited iodine and aerosols into the sump (e.g. Ag particles), sump/ gas iodine mass transfer. The main open issues concern:

- Production of organic iodide by interaction of iodine with paints. This issue is of highest priority due to the contribution of volatile organic iodide to radiological consequences. The two main programs ISTP (EPICUR experiments in IRSN) and OECD/BIP (experiments in AECL in Canada) demonstrate different tendencies that cannot be fully explained by models. More predictive models based on an identification of the interaction of iodine with paint chemical groups have to be developed to explain these differences and thus to give estimates in reactor cases.
- Characterization and behaviour of iodine oxides produced by interaction of volatile iodine with air radiolysis products and assumed to nucleate. SA simulations show that their concentration is higher than molecular iodine or organic iodide ones. Data are available from THAI experiments but they are not sufficient to produce predictive models. New dedicated experiments in containment accidental conditions should be performed to characterize the products of the destruction of molecular iodine and organic iodide by interaction with O° and HO° radicals.
- Stability of metallic iodide aerosols under irradiation. High rates of dissociation of CsI or iodine oxides aerosols have been evidenced experimentally and this should be confirmed to develop models of this additional contribution to delayed releases. More generally interaction between gaseous iodine and aerosols needs to be further investigated.

Two OECD/NEA/CSNI projects are under way and will finish in 2015: BIP2 (led by AECL) on iodine and ruthenium behaviour, and STEM that addresses, in the containment, the long term behaviour of iodine aerosols and the long term interactions between iodine and paints. THAI2 (led by GRS and Becker Technologies in Germany) on iodine behaviour in a multi-compartment containment has finished in 2014.

R&D topics

Short term

- ✓ **Production of organic iodide by interaction of iodine with paints, behaviour of iodine oxides and stability of metallic iodide aerosols under irradiation. But it seems necessary to wait for the outcomes of the OECD projects THAI2, BIP2 and STEM in 2015 before defining new R&D projects for the mid-term.**

3.2.4.5.3 Aerosol behaviour impact on source term

Existing open issues concern the following areas:

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- The conditions leading to primary pressure boundary failure which leads to containment by-pass,
- The quantification of aerosol retention in the secondary side of steam generator,
- The leakage through cracks and damaged penetration seals in the containment wall.
- The re-evaporation of volatile FP compounds from aerosols in hot zones (RCS and self-heating deposits in the containment), as well as particle dry resuspension caused by hydrogen explosion/steam blast, need to be better quantified.

R&D topics

No needs of further R&D at short term

3.2.4.6 Pool scrubbing

The issue of pool scrubbing was typically associated to BWRs of the Generation II (or III, e.g. ABWR). More recent designs, such as Westinghouse AP600/1000, introduced the phenomenology of pool scrubbing also in PWRs as an important aerosol removal mechanism. TMI2 accident scenario, as well as other scenarios like PWR seal LOCAs, can be responsible for gas bubbling through water sumps within the primary circuit. Steam generator tube rupture accident sequences should be also taken into account.

The accident at the Fukushima BWRs underlined the need of reducing the existing uncertainties about this important radionuclide removal process. The present pool scrubbing models for retention of aerosols and gas-phase radio-nuclides (I2, organic iodides, HOI, HI) have been developed for bare pool conditions, where an ideal jet of gas is discharged from a nozzle into a water pool.

The main open issues about the estimation of scrubbing efficiency are represented by the rise velocity of the bubbles (which are in the form of a swarm and are affected by water recirculation inside the pool), effect of high-flow injection from RCS (out of design conditions), presence of obstacles, influence of saturated (or boiling) conditions inside the pool, presence of water impurities.

R&D topics

Short term

- ✓ **Pool scrubbing but it is being addressed in the PASSAM FP7 ongoing project. New experiments on this subject could also be performed in a possible new THAI German project in the next years.**

3.2.4.7 Filters behaviour in severe accident conditions

The application of filtration systems to both existing and future containments could significantly mitigate the radionuclide releases in case of SA (with containment depressurization operator-activated or through valves and/or rupture disks). This issue has become an interesting option after the accident of Fukushima, although most plants in Europe have adopted this technology today (but with various solutions (sand bed, pool scrubbing…).

Different filtration systems are used (pool scrubbing, sand filters plus metallic pre-filters) or could be potentially used (agglomerators to be mounted upstream a filtration system, electrostatic precipitators, spray agglomeration systems, electric filtration systems, improved zeolites, etc.).

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Research programs are under preparation to highlight both the existing knowledge and the lack of knowledge in this field. The PASSAM FP7 project addresses this issue of filtered containment venting. From that step, the real needs will be identified and they will allow a precise definition of the experiments to be performed to improve the knowledge. For each type of filtration system the following questions will be answered: What has been tested (aerosols, molecular iodine, organic iodine, other gaseous species…)? Under which conditions (more or less relevant as regards severe accident conditions)? What efficiency? What is the understanding of the trapping phenomena? Are there models and/or correlation to pre-estimate the filtration efficiency of a specific system during an accident? Is there any risk of fission products re-entrainment or revaporisation due to dose rate, evolution in the carrier gas composition, FP concentration…? More generally the influence of those conditions on filter media behaviour and efficiency should be addressed.

R&D topics

Short term

- ✓ **Filtered containment venting systems but it is being addressed in the PASSAM FP7 ongoing project.**

3.2.4.8 Effect of impurities in water on source term

The presence of impurities in water could have some effect on the iodine chemistry in sump in the containment. The impurities could also affect other isotopes important for the source term. Other effects include buffer chemicals or materials in reactor buildings (e.g. Cl from impure water or cabling) causing precipitates that block cooling system filters.

R&D topics

Short term

- ✓ **More investigations are necessary to define R&D programmes on the effect of water impurities on source term and no project can be yet proposed at short term.**

3.2.5 Severe accidents linkage to environmental impact and emergency management (STA 2.5)

This subarea is interfacing the nuclear safety and the radioprotection communities and addresses cross-cutting R&D issues where promising synergies can be envisaged. Such a strategy aims at yielding a continuum of knowledge, simulation capabilities and expertise, from the evolution of the situation inside the plant and associated consequences (source term) and the overall impact of this source term on the environment and people inhabiting it. The description of this subarea particularly accounts for the analyses made of the Fukushima situations and their management.

3.2.5.1 Scope (including specific objectives)

The impact of severe accidents on the environment in the near-field[13] around the NPP must be assessed as part of the NPP Environmental Impact Assessment (EIA) in accordance with the European

[13]What is called “near-field” here is defined as the area that is for most of the countries under the responsibility of the NPP operator in case of an accident involving monitoring in environmental fluids (air, water), soils and surfaces located in the vicinity of the NPP and that would be first impacted. This area is thus the distance from

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and national legislations and because near-field contamination can directly affect the definition or the implementation of the accident management strategies.

The scope includes all phenomena leading to the atmospheric, on-ground and underground contamination of land from inside the plant to the near-field, including the impact of mitigation measures. These phenomena can be categorized as follows:

i. Evolution of in-reactor liquid and airborne source term to account for the impact of chemical conditions on the environmental processes,
ii. Evolution of reactor containment uncontrolled leakage paths,
iii. Processes of dispersion of radionuclides in the near-field zone, close to the reactor or facility:
iv. Atmospheric dispersion,
v. On-ground contamination by atmospheric wash-down due to precipitation (rain, snow…) or possible future mitigation systems such as sprays,
vi. Underground diffusion up to the terrestrial aquatic systems (water table, river systems).

The two first categories are already addressed in other TA2 subareas but specific/extended characteristics are required for the purpose of this subarea. Models for these phenomena should be assessed or improved in order to provide the near-field situation, globally considered as an evolving boundary condition, to the environmental dispersion simulation tools and to the emergency and post-accident management tools that mostly focus on mid- to far-fields (including those supporting PSA level 3).

Gains on the prediction of processes that lead to near-field contamination will also in turn yield an increased importance of considering environmental measurements in the assessment of the in-reactor accidental situation diagnosis.

A cross-cutting general issue dominating this subarea consists in reducing uncertainties in models and space/time discretization associated to assessing the impact of severe accidents on the environment. Determining the "acceptable uncertainty" of a source term composition for these EIA studies might be useful for the research done in the other subareas.

Another cross-cutting issue is the chemistry of radionuclides where the chemical forms of the most reactive ones including iodine are impacting their behaviour inside the plant and the dispersion modes in the environment.

3.2.5.2 State of the art

Significant efforts have been made in the last decades to improve models of in-reactor airborne source term and a clear roadmap exists to predict this airborne source term evolution from the short to the long terms (see the Subarea 2.4 “Source term”). The predictability of the evolution of the liquid source term did not reach a similar level where the R&D efforts mainly focused in the containment sumps on the trapping of radionuclides that can possibly form volatile compounds. The prediction of the liquid source term can benefit from R&D on: the evolution of the chemical

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the facility below which measures to mitigate environmental releases can be implemented or below which radioactive releases can affect implementation of SAM operations

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composition of the sumps performed to investigate the efficient functioning of the water recirculation safety systems (sump clogging issue) during the long period of the accident management, on one side, and on degraded fuel lixiviation on the other side. Nevertheless significant efforts remain to be performed to predict adequately the evolution of the chemical conditions and source term of the in-reactor liquid phase.

Research efforts in atmospheric dispersion modelling in the last decades have produced models and some of them are even used in preoperational or operational framework in case of a radiological emergency. In the CFD code category, the MERCURE-SATURNE code developed by EDF is one of the most advanced projects able to address the issue of atmospheric dispersion at very small scale between buildings.

The major platforms devoted to studies and operational preparedness and response (DSS: Decision Support System) in Europe are:

- ARGOS developed by DEMA (Danish Emergency Management Agency) and RODOS, currently linked to the NERIS network via an users' group (European Platform on Preparedness for Nuclear and Radiological Emergency Response and Recovery: see www.eu-neris.net), both having been conceived and developed after the Chernobyl accident;
- C3X-paZ and SYMBIOSE more recently developed by IRSN and successfully used during the Fukushima-Daiichi accident, as well as ECOLEGO developed by FACILIA (in Scandinavia).

These platforms (or some of them) are able to perform various tasks at various degrees of integration, such as atmospheric and water dispersion at different spatial scales, manage criteria for the post-accident phase, and improve parameterization through data assimilation and urban, natural and agricultural situations down to assessing impacts via dose estimates.

The current approach for emergency preparedness and response largely results from lessons learnt after Chernobyl. Currently, it appears to be applied consistently to the radiological impact of the Fukushima-Daiichi events, and is therefore continuing its worldwide testing in order to improve technical, organizational and methodical issues suitable to better mastering emergency preparedness and post-accidental longer term response. The Fukushima-Daiichi accidents have prompted a debate on extending activities on PSA Level 3. Such studies are still scarce today in Europe, except in a few countries like Finland by STUK, differently from the USA. This issue should cover the preparation of harmonized legislation framework across EU member countries, improved guidelines for developers and new advanced and standardized code to replace the rather old-fashioned COSYMA code (especially its user interface).

EC also launched projects for evaluation and benchmarking of emergency preparedness and response across the EU (NERIS platform). These have resulted in proposals for the improvement of the cooperative international system of emergency preparedness and response, focusing on adopting best practices, revealing weak points and bridging the gaps. Such an improved system of emergency preparedness and response to be adopted across the EU countries should ensure effective data and information exchange during radiological accidents, cooperation in assessment of the situation and prediction of the consequences, effective early notification and warning system

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activation in the affected countries (cross-border issues) and finally effective sharing of resources (decontamination units, evacuation vehicles, logistics etc.) and good information on their availability in real time.

In the OECD frame, a project is under elaboration on the long-term consequences of emergency situations and could start in 2015.

3.2.5.3 Challenges

Although quite significant efforts are currently dedicated to handle impacts on the environment at large from accidental releases, particularly under radioprotection-related platforms such as NERIS and ALLIANCE (European Radioecology Alliance), a large array of uncertainties still affect current assessment capabilities. Reduction of such uncertainties critically depends on a better liaison to be developed with the nuclear safety community, especially on up-stream issues inside and closer to the NPP and its local environment where the source term formation takes place.

Indeed the source term composition and dispersion, which will next promote the contamination of the environment at large, is governed by a complex array of interactions that are not exclusively restricted to the conditions of failure within the NPP structural barriers, but also involve interactions with the local atmosphere, soils and waters on-site, which are prone to influence the subsequent capabilities for further dispersion towards the much wider environment.

3.2.5.4 Near-field dispersion of atmospheric releases

3.2.5.4.1 Near-field atmospheric dispersion models

On-site, the environmental surrounding is mainly characterized by an area with large buildings. This type of configuration leads to complex atmospheric flows and specific atmospheric dispersion mechanisms (plume wash-down, recirculation…). Some CFD codes have started to address these issues but they still need to be improved.

The development of models able to represent with sufficient accuracy the non-homogeneous turbulent flows between buildings and the recirculating flows for groups of obstacles is a key step to be achieved in order to have a better view of on-site contamination resulting from a severe accident, but also to prepare plans for workers intervention during the development of the accident. These models need to be validated by means of comparison with site experimentations involving micro-meteorology, tracer experiments and also wind tunnel campaigns. A key issue is to improve the representativeness of models for stable atmospheric situations.

Under this subarea, a significant input from TA2 towards NERIS would be the evaluation of a source term to the environment as reliable and realistic as possible. This would be done in close links with the STA 2.4 and 2.6 (see resp. Sections 2.4 and 2.6).

R&D topics:

- ✓ **Improvement of the near-field atmospheric dispersion models, including stable atmospheric conditions, especially on-site, and in close coordination with similar issues addressed in NERIS.**

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3.2.5.4.2 Inverse modelling

Evaluating source terms, i.e. the amount of radio-elements which may or which are released into the atmosphere from the affected reactors during a response to a developing radiological emergency, is always difficult. The key parameters of the reactors are often not available to understand the situation and the state of these reactors. Most of the emergency management organizations rely on source term databases to be adapted to the situation.

During the last decade, some inverse modelling techniques have been developed and applied to the atmospheric transfer problem in order to allow a reconstruction of the source composition. They should now be adapted to the specific issues of giving access, from environmental monitoring around the NPP performed under the responsibility of the operator, to indications on the amount of some key radio-elements which have been released from the reactor(s).

The key issues to be developed are: on one hand, to be able to mix monitoring data types (time integrated deposition measurement, gamma spectrometry, air activity…); on the other hand, to use the dose rate measurements that are most commonly available in the immediate vicinity (a few km) or within nuclear sites. Dose rate monitoring is easy to implement and give direct information about the external dose to which the workers or the local population are exposed. However, this overall dose rate integrates the contribution of all radio-elements present into the atmosphere, or already settled on the ground. The main challenge here is to be able to use this type of monitoring with adapted radioelement spectrum algorithms by means of inverse models in order to reconstruct knowledge of the source term.

The Fukushima-Daiichi accident showed some atmospheric releases of very different radioelement spectra with time: the use of inverse atmospheric dispersion models should have provided a quick diagnosis of the major atmospheric releases and allowed the evaluation of the radiological impact of the on-going accident in order to protect the population in due time.

Moreover, such reconstruction will help the process of determining what the most plausible/pertinent SA scenarios on-going within the NPP are. This work should be made in close collaboration with R&D done on integral codes such as ASTEC and on emergency preparedness and response methods and tools (cf. Sections 2.5.3 and 2.6.1).

R&D topics:

- ✓ **Improvement of inverse modelling methods to reconstruct knowledge of the source term on the basis of available measurements out of the nuclear sites; This work is to be carried out in close coordination with similar issues addressed in the NERIS frame.**

3.2.5.4.3 Uncertainties

The origins of uncertainties in the evaluation of environmental impact at large are numerous. However, the biggest uncertainty certainly is the appropriate knowledge of the source term composition which quite significantly alters the EIA performance because the impact varies linearly with the source term. This issue can be reduced by sustained efforts on SA and by using inverse modelling, as described above. Other major sources of uncertainty requiring further attention relate to the local meteorology conditions (especially precipitation and wind), the parameterization of appropriate near-field atmospheric processes and also the adequate incorporation of radionuclide speciation aspects within the atmospheric models.

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In the meteorology forecast area, the use of ensemble computations (i.e. variations of calculations for different initial weather data) to improve the reduction of uncertainties and the overall forecast process has proved to be efficient. The Fukushima-Daiichi accident showed that all major meteorology agencies were not able to forecast the meteorological conditions observed during March 15 and the night of March 16 where the major atmospheric released from the reactor 2 contaminated the land of Japan. The wind direction forecasts were not correct and the radioactive deposition on the ground was not computed with accuracy before some field measurements were made. Also, recent analyses have shown a significant discrepancy between dosimetric aero-surveys of the Fukushima-Daiichi contaminated zone and doses calculated from sampling contaminated surfaces, with probable involvement of processes related to wet/dry deposition.

Thus, in order to estimate and reduce uncertainties, the adaptation and the development of ensemble modelling for radio-nuclides atmospheric dispersion is needed. The ensemble modelling has also to be calibrated with observations, mostly for meteorology, air activity and deposition processes. One of the objectives is to give, during a response to an emergency, quality information to decision-makers through the evaluation of some uncertainties, especially those related to local up-stream processes close to the source term formation.

R&D topics:

- ✓ **Development (or improvement) of ensemble modelling for radio-nuclides atmospheric dispersion to reduce the remaining uncertainties. This work is being carried out in close coordination with similar issues addressed in the NERIS frame.**

3.2.5.4.4 Chemistry of highly reactive radionuclides

The impact of the evolution of the chemical forms formed by the radionuclides in the near-field atmosphere is only poorly accounted in the atmospheric dispersion codes whereas, as demonstrated during the Fukushima-Daiichi situation, this evolution can impact both deposit rates and filtering efficiency of measurement devices for highly reactive radionuclides such as iodine. Development of models of the evolution of the chemical forms taken by iodine in the near-field atmosphere can benefit from both in-reactor iodine models and iodine behaviour modelling developed for pollution and climate change issues.

If the impact of radioactive decay on the chemical form (e.g. iodine/xenon or tellurium/iodine filiation) taken by radionuclides is not expected to strongly impact the radiological consequences, accounting for this effect is useful to guaranty the mass balance of radionuclides computed from the environmental measurements.

Under this subarea, a significant input from TA2 towards NERIS will be the combination of the models developed for radionuclides chemical reactivity within the containment to the atmospheric reactivity models.

R&D topics:

- ✓ **Investigation of the impact of the radionuclides reactivity and of the radioactive decay on their near-field dispersion.**

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3.2.5.4.5 Wash-down and mitigation

Radioactive aerosols are formed during a severe accident and can carry contamination outside when they are released outside of the NPP. Understanding how they interact with falling water droplets is paramount both within the containment and in the outside environment. In the former case, the temperature rise within the reactor containment is counteracted by spraying water in order to reduce the pressure build-up using the CSS (Containment Spray Systems). In the second case, it is well-known that deposits of radioactivity on environmental surfaces are much enhanced in case of rain which washes down the aerosols. R&D is needed to more precisely identify the processes involved, and evaluate the feasibility of reducing widespread dispersion of radioactive aerosols by washing it down with artificial rain/water by spray systems. This R&D topic will benefit from the research already made to assess the CSS efficiency.

R&D topics:

- ✓ **Identification and modelling of radionuclides wash-down processes in environmental conditions.**

3.2.5.5 Liquid releases and impact

One important issue highlighted by the Fukushima-Daiichi accident is the problem of water contamination on-site, with the associated potential impact on the external environment and its living components via the movement of contaminated waters out through run-off, under-ground water, and finally water tables and river network receptors.

Proper methods and tools for the evaluation of the in-reactor liquid source term are also largely lacking. There are clear cross-cutting issues needing better exploitation, as in-reactor evaluations (source term, sump clogging issues, preparation of damaged facility post-accidental management and then dismantling), such as detailed chemistry of radionuclides, for example, can provide the necessary input information to support more accurate evaluations of external contamination impact in the environment.

Depending on the geological structure and content on which the NPP has been built, various kinds of geotechnical barriers have been envisaged. However, geotechnical barriers have been proved to act primarily on the delay before contamination may escape from the site (an important aspect for emergency management), but they are most often not able to ensure definitive containment of the contamination on-site in the longer term.

Facing the potential for leakages out from the site leads to the need of improved mitigation potential techniques on the management side. But it also leads to the need of improved knowledge on their contamination content (source term and speciation issues). For example the on-site decontamination techniques to be set up will require to be dimensioned with respect to the contamination content and activity, themselves governed by interactions between water, physico-chemical composition of the under-ground geological materials and radioactive materials enclosed in the corium or leaching out.

The link should be established with R&D in STA 2.2 on MCCI (see Section 2.2), a minima to give initial and limit conditions for evaluation of transport or radio-elements within the ground.

R&D topics:

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- ✓ **Development of methods and tools for the evaluation of the in-reactor liquid source term.**

3.2.5.6 Emergency preparedness and response

Despite all effort invested in improving the safety of NPPs and other nuclear installations, there is no way to reduce probability down to zero, nor the associated risk for a radiological accident with large detrimental consequences. It is therefore necessary to develop and improve diagnosis and prognosis methods and tools aiming at reducing or avoiding the impacts of accidental release of radioactive substances on the populations and to the environment, and to back use them in support of SAM. When combined, these methods and tools should provide all the necessary information for the decision-makers to first take the necessary decision to protect the population, and then, in a cyclic process, to review the diagnosis/prognosis for both population protection and SAM improvement until the facility returns to a safe state. Deterministic diagnosis tools will target the evaluation of the worst situation possible to define the population protection measures whereas probabilistic tools will help in improving the SAM measures to implement. These tools can be also used, in combination to SA system codes such as ASTEC, to improve the diagnosis of the facility situation and so prepare the accident long-term management.

Additionally to these tools, the R&D identified in Section 2.5.1 on methods of inverse atmospheric dispersion modelling will also contribute to the response to emergency situations by using the environmental measurements of main release episodes to narrow the diagnosis made of the on-going situation inside the facility.

R&D topics:

- ✓ **To improve the understanding of human and organizational factors during emergency (human behaviour under stress conditions, availability of reliable information, preparedness in terms of scenarios, good interpretation in order to discard spurious signals...) and their integration in associated methods and tools,**
- ✓ **to develop and improve predictive methods and tools to assess the actual situation of the accident course and to predict inadmissible release of radioactive substances (source terms) from nuclear facilities, and to evaluate the potential radiological consequences; information support based on fast predictive calculations,**
- ✓ **to develop and improve methods and well suitable tools allowing for better appraisal of uncertainty and sensitivity analysis in support of probabilistic safety assessments level 3.**

3.2.6 Severe accident scenarios (STA 2.6)

3.2.6.1 Scope

The integral codes (or system codes) are essential to simulate the SA complete scenarios up to the evaluation of the source term (i.e. radiological releases) into the environment, as well as to evaluate SAM measures. The research efforts along these last decades have produced a number of integral codes, mostly developed in the framework of national programs, that constitute the main tools for deterministic studies (in addition to mechanistic codes) and for PSA level 2 studies.

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The Fukushima-Daiichi accidents have also underlined the importance of the behaviour of SFPs in case of loss of cooling.

3.2.6.2 State of the art

The main integral codes used in Europe are: ASTEC (developed by IRSN and GRS), which is considered now as the reference European SA code since it capitalizes the whole European knowledge on severe accidents, MELCOR (developed by SNL for the USNRC) and MAAP (developed by FAI Inc., USA, for a consortium of vendors and utilities). In addition specialized codes, based on mechanistic approaches, are developed for studying in details some phenomena (circuit two-phase thermal-hydraulics, release of fission products from fuel, aerosol transport, etc…), including CFD codes.

The integral codes are considered now as the repository of knowledge and most of their models are considered today state of the art and extensively validated but all the ongoing R&D outcomes (see the sections 2.1 to 2.4) must continue to be integrated into these codes. The Fukushima accidents have also underlined the need to improve the modelling of mitigation systems and of BWR. The applicability of these codes to SFP must also be improved.

Large efforts on ASTEC improvements have been made in SARNET/FP7 frame with benchmarking activities with the other integral codes. Out of Europe, the International Standard Problems (ISP), periodically organized by OECD/NEA/CSNI, allow benchmarking activities among codes on given high-quality experiments. A similar benchmarking process has been launched in November 2012 by OECD/NEA/CSNI in the BSAF project on the Fukushima accidents, both in reactors and spent fuel pools.

3.2.6.3 Challenges

A first challenge is to continue to capitalize on knowledge in the integral codes, particularly the ASTEC code, and to feedback the interpretation of the Fukushima accidents in the coming years. Attention should be paid in particular to models of BWR core degradation and to their validation. All these efforts in integral codes will produce necessary conditions for preserving and disseminating knowledge to end-users (utilities, safety organisations) and to new nuclear countries.

Another challenge is to improve the capability of codes to simulate the SFP behaviour in case of loss of cooling. Specific R&D in support should address the following phenomena: large-scale flow convection, impact of partial dewatering of fuel assemblies on thermal runaway and fuel degradation, clad and fuel mechanical behaviour in an air-steam atmosphere (the latter being already addressed in the sub-area 2.4: see section 2.4.1).

Another challenge is to investigate new instrumentation to be used for SA diagnosis and management, as well as for early source term predictions and emergency preparedness outside the NPP site.

Finally it is essential to store in reliable databases the results of the huge amount of SA experiments that were performed during more than 30 years. They should remain available for any further analysis of SA phenomena for validation of simulation codes.

3.2.6.4 Development and validation of integral codes

The integral codes cover globally all needs for SA evaluation in present PWR, BWR and VVER, and in some Gen III NPPs like EPR. The preliminary analysis of the Fukushima-Daiichi accidents did not

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underline new phenomena that could not be modelled in SA integral codes. But it showed that additional efforts must be done to adequately model and simulate:

- the systems of mitigation of consequences: water cooling of degraded cores and of corium during MCCI, FP filtering, especially during containment venting,
- the behaviour of SFPs in case of loss of the cooling system or loss of water inventory.

Model improvements should come from the benchmark exercises amongst integral codes that will continue in the next years in an international context (OECD/NEA, SARNET…) on the Fukushima simulation, especially BWR models for core degradation. The trend is also to include progressively more mechanistic models, especially in ASTEC and MELCOR. Attention should be paid in permanence on the training on code use in order to get qualified users.

The main current project in Europe is the joint IRSN-GRS development of ASTEC code that played a central role in SARNET FP6 and FP7 projects by capitalizing the knowledge acquired by the network (and beyond, such as OECD/NEA projects) through new or improved physical models. Currently 30 software agreements have been signed with European partners (plus a few outside Europe). This project will continue intensively with code improvements planned in 2014 in the next major version V2.1 that will allow completion of BWR calculations. The CESAM project, coordinated by GRS and with a strong IRSN involvement, started in FP7 frame in April 2013 for 4 years with 18 partners: it aims at ASTEC improvements in the light of the Fukushima accidents, in particular on SAM modelling and on interfaces with codes for simulation of off-site atmospheric dispersal of radio-elements (as already done for MELCOR through the interface with the MACCS code). Such interface will allow using reverse uncertainty methods in order to evaluate the families of potential events occurring within the reactor buildings, based on measures of off-site deposits. This process will be done in close relation with the issues addressed in section 2.5.2.

Improvements are also continuing in the MELCOR and MAAP codes that are used by several European organizations and in other codes such as the ATHLET-CD/COCOSYS GRS system.

R&D topics

Short term

- ✓ **Continuation of the capitalization of knowledge in ASTEC, especially for BWR. Another topic is the creation of ASTEC interfaces with codes for off-site atmospheric dispersal of radio-elements. Both activities are being performed in the current CESAM FP7 project.**

Mid or long term

- ✓ **Investigations will be done on advanced software platforms for the use of diverse codes (such as CFD and integral codes…) in the same software environment and if necessary to couple them. This would also for instance to replace the presently used lumped–parameter approaches by CFD approaches where thermal-hydraulics and airborne transport of radio-elements will be described more realistically.**

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3.2.6.5 Simplified modelling for PSA2 or emergency situations

Two different approaches are usually being used for PSA2 studies: either intensive use of integral codes or coupling of simplified very fast-running models with probabilistic tools.

The continuous increase of computer performance allows the reduction of the calculation times of integral codes. Faster calculations will allow more intensive uncertainty and sensitivity analyses of critical parameters for SA scenarios since in general the integral codes are coupled with tools for propagation of uncertainties (such as SUNSET in IRSN used for ASTEC calculations, or SUSA in GRS). This should make their use for PSA2 studies easier.

For tools dedicated to emergency situations, this requirement becomes a logical need since the calculation time must be extremely short. Simplified models are used in general in such tools and benchmarked against the integral codes. The latter "validation" of simplified models should continue in the future. But in the future, the integral codes could be used more and more frequently by adopting rough nodalization of time and space.

All these above efforts done on integral codes, e.g. on ASTEC in European frame, should allow to build in the future graphic and interactive simulator tools in SA conditions, useful for training NPP operator or for education of young researchers or students.

R&D topics

Short term

- ✓ **Continuation of R&D on one side on the development of simplified modelling and on the other side on the reduction of computing time of the integral codes.**

Medium or long term

- ✓ **Creation of graphic and interactive simulator tools in SA conditions.**

3.2.6.6 Spent fuel pool scenarios

The Fukushima accident has called the attention to the behaviour of SFPs in case of loss of the cooling system. Heat exchange between water and air in the fuel building is insufficient to evacuate the residual power of the assemblies, which may lead to water boiling and gradual pool boil-off. There is also the risk of criticality in the case of boiling in the storage racks.

The fuel temperature rise will promote the highly exothermic oxidation reaction of zirconium cladding by steam and air present in the fuel building. Dewatering of the fuel assemblies will imply a potentially large hydrogen production, and the extensive failure of hot fuel cladding in air will be associated with a very significant release of radioactive materials into the environment and finally a severe oxidation and crumbling of fuel.

In order to estimate the safety margins, more knowledge is first necessary on the large-scale convection phenomena in SFPs in the different phases of the accident. Then the situation of partial dewatering of the fuel assemblies should be investigated as a complement to the OECD SFP project (2011-2013) that consisted in SNL experiments (USA) on thermal runaway due to the exothermic oxidation reactions of zicalloy cladding and on fuel degradation. Finally, in order to better estimate the margin before runaway reactions, more knowledge should also be acquired on the role of nitrogen in the acceleration mechanisms of degradation of the cladding and on the mechanical

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behaviour of a cladding having undergone oxidation/nitriding in an air-steam atmosphere. The combined effects of cladding attack and fuel degradation in air on the fuel fragments and cladding debris formation should also be investigated.

In addition the leaching of all FPs from the oxidised fuel and cladding debris in the remaining heated water should also be investigated (rapid initial and then longer term exposures) and their subsequent movement (e.g. dispersal in water or surface deposition).

R&D topics

Short and mid term

- ✓ **Reduced scale experiments on convection in prototypical SFPs, and assembly-scale experiments to study the thermal-hydraulic conditions representative of a loss of coolant accident, with and without dewatering. CFD codes could be used to interpret these experiments and be further validated. Phenomenological models should also be elaborated for integral codes such as ASTEC.**

3.2.6.7 Re-criticality in SA conditions

In relation with the first analyses of the Fukushima-Daiichi accidents, the issue of re-criticality in conditions of a degraded core or in a SFP must be considered.

This situation can concern for instance the reflooding by non-borated water of a partially degraded BWR core, with intact fuel but melted and relocated control rods. The length of the time window for such configuration is of interest for accident management. The question rises also on the stability of the abovementioned core configuration geometry (here strong link with the sub-area on in-vessel corium coolability).

R&D topics

Short term

No new project is planned on this subject at short term.

3.2.6.8 Instrumentation for SA diagnosis and management

SAM Guidelines (SAMG) implementation and effective use require accident progression to be detected and consequences of different SAM measures to be predicted. Currently the instrumentation available during SA is insufficient for effective SAM. Very little sources of information came from inside of the reactor buildings in the Fukushima accidents. Operation teams during both TMI2 and Fukushima-Daiichi accidents had difficulties to understand the current status of reactor and systems that are important for the plant safety.

Different kinds of novel instrumentation systems for direct and indirect measurements and observation (optical devices with the advantage of remote location outside of the reactor buildings or remotely operated devices at key points in the reactor buildings), could be potentially used for identification of several important parameters, like: fuel/clad temperature and fuel location inside RPV, hydrogen release and concentrations in most important locations of primarily circuit and containment, RPV lower head temperature and integrity, corium temperature and location in the reactor pit, basemat concrete ablation depth, FP releases and concentrations in gas and liquid phases of different locations in containment.

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R&D topics

Short term

- ✓ **Survey of existing SA instrumentation and then the identification of possible improvements.**

3.2.6.9 Experimental databases

A huge amount of experiments has been performed for more than 30 years on SA. It is not sure at all that some of these results can still be accessed today in a practical and easy way. It is essential to keep their results available today and in the future for any further analysis on the various phenomena and for validation of simulation codes.

An attempt to gather important results started in the SARNET frame in 2004 on the basis of the JRC STRESA tool. A database, named DATANET, has been created and is managed by JRC/IET: end of 2013, it included 265 experiments from 43 different facilities. All proprietary aspects are respected since the access is protected by a secure process and can only be given by the data's owner. JRC/IET is currently working to upgrade the STRESA tool and the storage in various European sites should continue in the next years.

R&D topics

Short term

- ✓ **Continuation of storage of SA experimental data in continuity of SARNET FP7, both for new experiments and for older ones still missing in the DATANET database. JRC/IET is currently upgrading the STRESA tool as basis for further storage.**

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3.4 Annex

Table 1: list of main experimental facilities

The experiments in italic fonts are performed by non-European organizations but are the object of bilateral agreements with some European organizations (or in the frame of a FP7 project).

Sub-area	Facilities (owner)	Objectives
2.1	QUENCH (KIT) CODEX (AEKI) DEFOR (KTH) PRELUDE/PEARL (IRSN) DEBRIS (IKE) COOLOCE (VTT) QUENCH-Debris (KIT) LIVE (KIT) RESCUE (CEA) JRC/ITU exp. *CORDEB (NITI)*	Core degradation " Debris formation and coolability Debris coolability " " Debris formation Corium behaviour in vessel lower head External vessel cooling Corium thermo chemistry *Corium behaviour in vessel lower head*
2.2	DISCO (KIT) VULCANO (CEA) HECLA, COMETA (UJV) SICOPS (AREVA GmbH) CLARA (CEA) MOCKA (KIT) POMECO and DEFOR (KTH) KROTOS (CEA) *CCI (ANL)*	DCH MCCI (real materials) " " MCCI (simulant materials) " Debris formation and coolability Debris formation *MCCI*
2.3	TOSQAN (IRSN) MISTRA (CEA) THAI (Becker Techn.) PANDA (PSI) CONAN (Univ. Pisa) HYKA (KIT) ENACEFF (IRSN-CNRS) REKO (Jülich) *Russian facilities in ERCOSAM*	Gas distribution " " " Condensation on containment walls H_2 combustion " PARs *Gas distribution*

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	project	
2.4	Phébus.FP (IRSN) VERDON (CEA) RUSET (AEKI) FIPRED (INR) CHIP (IRSN) EXSI (VTT) JRC/ITU exp. EPICUR (IRSN) THAI (Becker Techn.) PARIS (AREVA GmbH) Chalmers exp. (Sweden)	Integral experiments on source term FP release " " FP (Iodine, Ruthenium…) behaviour in circuits " " Iodine in containment " " Iodine and Ruthenium behaviour

Table 2: list of main simulation codes

Sub-area	**Integral codes (owner)**	**Mechanistic codes (owner)**
2.1		ATHLET-CD (GRS) + VECO (IKE): core degradation MC3D (IRSN): FCI and steam explosion CORIUM-2D (RSE): corium pool behaviour DECOSIM (KTH): debris formation and coolability
2.2		MC3D (IRSN): FCI and steam explosion JEMI, IDEMO (IKE): FCI and steam explosion TOLBIAC (CEA): corium pool behaviour
2.3	ASTEC (IRSN-GRS) MELCOR (USNRC) MAAP (EPRI)	COCOSYS (GRS): containment phenomena ECART (RSE): containment phenomena FUMO (Univ. Pisa): containment phenomena TONUS (IRSN): CFD code for gas distribution and explosion GASFLOW, COM3D (KIT): CFD code for gas distribution and explosion REKO-Direkt (Jülich): recombiner behaviour SPARK (IRSN): recombiner behaviour CFD commercial codes (ANSYS: CFX and Fluent)
2.4		ECART (RSE): behaviour of fission product and aerosols COCOSYS (GRS): behaviour of fission product and aerosols in containment INSPECT, IODAIR (NNL): iodine chemistry in containment

Table 3: list of priorities for all TA2 R&D Topics

The ranking of priorities is directly derived, for STA 2.1 to 2.4 and STA 2.6, from the exercise performed by the SARNET network [TA2-1, TA2-2]. A first ranking was elaborated in 2002 in the EURSAFE FP5 project [TA2-5] using a PIRT approach (Phenomena Identification and Ranking Table) with the support of more than 100 severe accident experts from 20 organizations of divers types (R&D, utilities, regulatory, industry, universities). The methodology consisted in 3 steps:

- ✓ Establishing lists of phenomena covering the whole spectrum of severe accident situations and events,
- ✓ Ranking these phenomena according to their relevance to reactor safety (safety-oriented groups),
- ✓ Ranking the phenomena according to their degree of knowledge (phenomena-oriented groups).

The final ranking took into account both safety and knowledge aspects and was the basis for the elaboration of R&D programmes.

During the two FP6 and FP7 SARNET projects, the SARP (Severe Accident Research Priorities) group of ad-hoc experts, led by GRS, has updated continuously from 2004 to 2013 this ranking, taking into account the R&D progress, the PSA2 studies and, after 2011, the impact of the Fukushima-Daiichi accidents [TA2-5].

The table below summarizes the ranking for the short-term period (0 – 2 years).

Sub-area	High priority R&D topics	Medium priority R&D topics	Low priority R&D topics
2.1	- Debris bed reflooding - Corium coolability in lower head and RPV integrity due to external vessel cooling	- Hydrogen generation during melt relocation into water	- Hydrogen generation during reflooding
2.2	- FCI premixing phase (in particular fragmentation of corium jets into a water pool) - Complements to MCCI 2D experimental database with other concrete compositions; analysis of the experiments in simulant materials investigating the 2D convective heat transfer distribution - Corium coolability by top flooding in the cavity	- DCH: scaling effects and hydrogen combustion	- Corium release following vessel failure - Core catchers of different designs: efficiency of water bottom injection, validation of designs with corium ceramics interaction

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	- Ex-vessel debris bed formation and characterization, in particular for BWR situations - Effects of “untreated water” on corium chemistry		
2.3	- Simulation of hydrogen distribution in containment under the influence of steam or air plumes or jets - Simulation of PARs operation under extreme conditions - Reliable models of deflagration to detonation transition		- Aerosol transport through containment cracks
2.4	- Oxidizing impact of the atmosphere on the FP release - Gaseous ruthenium transport in circuits and delayed releases from iodine deposits - Production of organic iodide in containment by interaction of iodine with paints; behaviour of iodine oxides and stability of metallic iodide aerosols under irradiation - Pool scrubbing - Filtered containment venting		- Core reflooding impact on FP release - Aerosol behaviour impact on source term
2.6	- Capitalization of knowledge in ASTEC integral code, especially for BWR - SFP accidents: thermal-hydraulic experiments, simulation with CFD and integral codes - Survey of existing SA instrumentation and then the identification of possible improvements - Continuation of sustainable		

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	storage of SA experimental data		

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4 TECHNICAL AREA 3 – Improved Reactor Operation (TA3)

Technical Area Leader: Ales Laciok (CES)

4.1 Executive Summary

4.1.1 Scope

The technical area TA-3 covers various aspects of reactor and core operation, and is not oriented to the design and nature of physical and chemical processes; rather it includes human and organizational factors, as well. In fact, safe and efficient operation of nuclear power plant is the result of a correct blend of human, organizational and technological aspects. Excluding more general sub-area “improvement of the operation economics and NPP flexibility”, five specific sub-areas have been identified to be taken care of in TA-3, directly impacting the reactor operation:

1. human and organizational factors
2. digital technologies in NPP operation
3. core management
4. water chemistry and LLW management
5. radiation protection.

The scope of the area TA-3 was defined with the intention to avoid overlapping with safety issues, particularly addressed in the area TA-1, with the fuel thermo-mechanical behaviour addressed in the TA5 and with the themes linked to structure integrity, in-service inspection and ageing management, which are included in the areas TA-4 and TA-8.

4.1.2 Objectives

The main goal of development of the scope of this technical area was to try to integrate, in a common advanced vision, various aspects of NPP operation. The broad and important area of human and organizational factors, linked with the topics related to implementation of the up-to-date digital technologies can provide the basis supporting such an integrated vision. Other goals are more related to selected specific operational targets (core management, water chemistry management and radiation protection). All these aspects converge in the improvement of the operation economics.

Operation economics is a result of balanced organisational and technical factors. In order to maintain the nuclear energy as one of the power generation sources it is needed to reach as low as possible cost of operation in different operational modes (base load, load following – that is allowed by power plant flexibility capabilities).

Human and Organizational Factors (HOF) are key subjects of analysis made with the aim to improve safety, performance and efficiency characteristics of nuclear power plants operation. After the Fukushima accident, the focus of studies on HOF has been moved towards safety culture, managerial aspects of plant operation, risk impacts of human and organizational factors and emergency management. High importance of human and organizational performance in emergency conditions is a key lesson learned from the Fukushima accident.

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Implementation of modern digital technologies offers a unique opportunity for improving operational performance, enhancing nuclear safety and supporting life extension of NPPs. Although some European plants have gone through relevant modernization programs, many interventions still have been performed under the constraint of minimizing the impact of the traditional way of operating, maintaining and managing the plant. Digital technologies are often implemented to provide solutions to specific problems and immediate needs, rather than to keep a long term perspective. Due to the rapid evolution of these technologies, this approach does not appear as being sustainable in the long term, and it needs to be modified in the future.

Improvements in core management are currently based on the continuous updating of the design and analysis tools, with the aim of achieving higher accuracy with established uncertainty evaluation, through a strengthened understanding of the underlying physics and associated modelling requirements, combined with enhanced computational efficiency. This task can be directly translated into large challenges in basic nuclear data, neutronics, material science, thermo hydraulics, fuel fabrication and fuel storage. Coupling all these aspects (multi-physics) with the help of up-to-date advanced software is the driver for replacing the current systems of codes used for simulation of individual processes related to reactor operation. Acquired results should be applied to engineering practice, namely to core design, core reload optimization, core monitoring and safety analysis.

Water chemistry and LLW management activities have the main target in optimization of chemical parameters of the primary, secondary and auxiliary cooling systems and in development of the optimum technologies for LLW treatment. Water chemistry is actually one of the most powerful tools which operators can use to improve the lifetime of plant components and systems. A good shape of water chemistry can significantly reduce “randomly” occurred operational problems, including corrosion, erosion, deposition of corrosion products etc.

Radiation protection is a specific area dedicated to the protection of humans and the environment against harmful impact and/or consequences of radiation. This is achieved first through improving the scientific understanding on how ionizing radiation interacts with living matter (humans and non-human biota) in representative conditions of exposure and how such interactions promote effects on health (for man) and ecosystems (for the environment). On this basis next, regulations, guidelines, methods and tools are constructed in view of reducing/eliminating the risks to personnel, population and environment exposure, as arising from natural and man-made (i.e. medical and industrial) sources of radiation. This operational goal is especially driven under ALARA principles, i.e. to limit exposure "as low as reasonably achievable".

4.1.3 State of the art

Improvement of the operation economics and NPP flexibility - Owners of nuclear power plants currently mostlyoperating in deregulating competitive markets are under pressure to reduce operation cost to be more competitive with other energy production options. Various aspects are considered – flexibility of operation, management of outages, efficient use of nuclear fuel, optimization of labour resources, sharing of resources among several plants, etc.

During last two decades, the human and organizational factors community has been promoting various themes for R&D in several international subjects and platforms as, for example, in the OECD/NEA Working Group on Human and Organizational Factors (WGHOF), within various effort organized by IAEA, in international cooperation organized by OECD - Halden Reactor Project (HRP), etc.

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A roadmap addressing the human and organizational factors has been recently developed within the EU 7th Framework programme MMOTION (2009-2011). In addition to the detailed roadmap, the MMOTION partners developed the guide-lines for four research programs oriented to:

- risk-informed decision-making in design and operation of NPPs,
- culture and practices for safe operation of NPPs,
- integrated design approaches addressing and including human and hardware-based in operation of NPPs,
- specific technological requirements in nuclear and other high risk industries

The MMOTION roadmap have mainly contributed to the NUGENIA roadmap of the technical subarea TA-3.2 "Human and organizational factors", detailed in Section 3.2 here below. In addition, some ideas from MMOTION roadmap have been also used in Section 3.3.

Defining a strategy for the implementation of fully digital systems into the processes of NPP operation control is becoming a more and more urgent issue for the life extension of the Generation II reactors, as well as for the deployment of the Generation III. A huge research program in this field is being prepared in US by the Idaho National Laboratory, potentially influencing the R&D activities also in Europe. This program intends to develop standards and guidelines to facilitate the transition to digital technology and its deployment across the US nuclear fleet. Five key technical areas have been identified: Highly Integrated Control Room, Highly Automated Plant, Human Performance, Improvement for NPP Field Workers, Integrated Operations, Outage Safety and Efficiency.

Core management has been subject of ongoing research and development activities, as it has direct impact on the economy of the fuel cycle. For the purpose of this roadmap, it is divided into four distinct areas: 1) core and fuel system design related calculations and numerical modelling, 2) evaluation of uncertainties, 3) core reload optimization and 4) core monitoring and instrumentation. Core design covers Neutron-physical design, Thermal-Hydraulic design and Fuel Rod and Mechanical design. In engineering practice, these topics have been yet treated either independently or with very simplified interrelated feedbacks. The multi-physics approach is an important subject of active R&D work. The same is valid for the core monitoring and core reload optimization. The core design and safety assessment calculations are to move from conservative approaches to the so-called best-estimate plus uncertainty methodologies. Uncertainty analysis is based on comparison to experimental data and on propagation of uncertainties of basic data and physical models through the system of computational tools. In the areas of system similarity for the purpose of validation against experimental data and on uncertainty propagation, active research is ongoing.

The water chemistry topic has been the subject of large basic and applied research programs recently. Nowadays, it is difficult to find simple control management for NPPs. Problems differ not only from one power station to another but also from one unit to another. During operation of reactors generation G II and G III, when lot of problems have been identified, were fixed and solutions have been put into practice, nevertheless some issues still remain valid for another research. There are demands for writing a complex of standard specifications for each nuclear power plant since only general regulations for various types of power stations (VVER/PWR) are not sufficient with regard to individual conditions of each unit.

Radiation protection has been recently re-emphasized at European level leading to the establishment of the Multidisciplinary European Low Dose Initiative (MELODI) platform, with the overall aim of

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integrating European initiatives on low dose research including epidemiology. Scientifically, this is aimed at resolving the sustained controversy about the existence/non existence of radiation effects at low dose (and/or dose rate) which may lead to reorient the current system of radiation protection. In practice, this is meant to improve efficiency by means of mutualisation of efforts and infrastructures and also by opening better access to critical mass of research teams on key scientific issues. This platform, initially oriented to better understanding the health effects of man's exposure to low dose radiation, issued in 2011 a Strategic Research Agenda, which defines a series of topics suitable to be considered in the long term research program. Most recently, in 2013, it has been enlarged to include also the effects on the environment itself (Radioecology Alliance platform, supported by the STAR and COMET EU projects) as well as issues related to emergency preparedness and response in case of accidents (European platform on preparedness for nuclear and radiological emergency response and recovery (NERIS), all gathered within a wide radiation protection integrated association (called OPERRA).

The radiation protection rules as developed by ICRP are based on a renewed system framework which proposes a design and application of limits suitable for both personnel, population and most recently non-human biota protection, as documented by ICRP guidelines 103 issued in 2007, and 109 issued in 2009. The renewed approach largely results from lessons learnt after Chernobyl. Currently, it appears to be applied consistently to the radiological impact of the Fukushima events in March 2011, and is therefore continuing its worldwide testing in order to improve technical, organizational and methodical issues suitable to better mastering emergency preparedness and post-accidental longer term response.

Other activities carried out worldwide (e.g. OECD-HRP) are focused on virtualization of real world with added information, so called Augmented Reality. These methods and tools serves as a contribution for optimization of maintenance and repairs of equipment in contaminated areas (e.g. disassembly during decontamination), for training of the personnel in view of reducing and/or eliminating human failures and finally for optimization of interventions and radiation monitoring activities during emergencies.

4.1.4 Challenges

4.1.4.1 Improvement of the operation economics and NPP flexibility

Owners of nuclear power plants currently mostly operating in deregulating competitive markets are under pressure to reduce operation cost to be more competitive with other energy production options. To recover huge initial investment cost and to maintain necessary level of profitability it is reasonable to prolong operation of plants where it is feasible, naturally without compromising safety and security. Along with traditional safety and reliability parameters, economic and financial factors are needed to be taken into account in new perspectives nowadays that is incomparable with former regulated markets where utilities provided complex service with inclusion of all reasonable costs.

4.1.4.2 Human and organizational factors

The aspects/topics, which are considered as utmost priority in this field, are "Human factor related risk-informed decision-making in design and operation of NPPs" and "Culture and practices for

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safety", which include the topics of human reliability analysis, operational culture, work practices. The important challenges are to strengthen the objectivity of safety judgments by using methods of risk-oriented decision making in human reliability area, to improve the cost-effectiveness and the balance of safety provisions, to harmonize human factor and safety culture operation principles across Europe and to minimize the negative impacts of complexity on safety and efficiency of plant operation.

Operational experience, safety culture and operating practice can significantly influence the current level of the operational safety; that's why it is necessary to optimize the performance and robustness of processes, which are integral part of operating NPPs at present, to be safe, reliable and efficient also in the future. The research should define the conditions required for ensuring the robustness of the organizations in charge of operating future NPPs and to postulate the optimum way of management of change, based on a deep understanding of practices and culture in the operation of existing plants and also how changes can impact the complex socio technical system and its performance.

An important challenge is to consider how individuals, teams and organizations function and interact within the plant in the environment defined by specific safety culture, and how they are supported by tools, artefacts, procedures, rules, etc. Focusing on the human and organizational components of the socio-technical system should start from psychological dimensions with the idea to be extended up to analysis of social and cultural dimensions of NPP operation. The Fukushima accident revealed significant human factor related weaknesses in the on-site and off-site response to extreme, unforeseen events, in particular when the functioning of the emergency organization itself is weakened by the event.

4.1.4.3 Integration of digital technologies

Digital technologies are nowadays deployed in all modern power generation plants and also in large industrial plants and devices characterized by a relevant risk level. The situation in the nuclear power sector is peculiar, and differs from other sectors in the following key aspects:

- the use of analogical systems is being extended beyond their expected service lifetime,
- complexity of safety demonstration and cost issues are the main barriers to wider application of digital technology in the I&C systems

Many digital technologies are already on the marked and widely used in industrial installations. They offer a huge potential for application in NPP, ranging from normal operation to maintenance optimization, from radiation protection to emergency management.

For achieving better integration of technical, human and organizational aspects in the various activities of the new advanced digital information and control architecture design process, more focus needs to be put on efficient integration of HOF requirements in the design by the prediction and timely assessment of the future work conditions and through active participation of HOF specialists and end users. Introduction of digital technologies in analogue control rooms requires application of human factor engineering principles, which are in continuous evolution. The development of an advanced alarm systems is also a possibility offered by digital technology. The complementary aspect of an advanced alarm system is the evaluation of how advanced visualization techniques could improve operator recognition and comprehension of alarms, influencing their performance.

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Several kinds of mobile communication/computing devices may be used to provide continual plant status and control capability anywhere in the plant, improving the performance o field workers. Tools using virtual reality models and technologies could be used to develop computer aided maintenance procedures and to train maintenance personnel.

In recent years, a shift in technology has led to the use of programmable digital electronic systems in nuclear safety applications, to overcome the difficulty in maintaining analogue electronic assemblies and to take advantage of functions enabled by digital logic. This however increases the potential for component-level faults due to engineering mistakes.

New systems based on different kind of sensors, data storage and analysis can help in optimizing maintenance activity, on the basis of the real need, preventing faults and hence enhancing plant reliability. These systems could be coupled with advanced methodologies and modern tools for reliability calculation (including both quantitative and qualitative reliability variables) and risk assessment, based on probabilistic approach, under development in various laboratories. Information technology could support an advanced centralized online monitoring centre to collect all information from each plant and conduct long-term plant asset management.

4.1.4.4 Core management

The management of a LWR core has an objective to maximize the cycle energy production at minimal fuel cycle costs while it ensures sufficient margins to relevant safety criteria of the fuel system designs (i.e. nuclear, thermal-hydraulic, thermo-mechanical and mechanical) and relevant core operational limits. Several topics were identified as the principal challenges.

The first topic is oriented to improvement of precision of core calculations and numerical modelling, as it has direct influence on the economy of core management. All aspects of modelling have to be concerned: precision of criticality and power distribution predictions, modelling of fuel behaviour and thermal-hydraulics, and predictions of used fuel isotopic composition. Improvement of core monitoring and instrumentation comes hand in hand with these challenges; it is to include detector signal interpretation capabilities, power reconstruction process and uncertainty evaluation of reconstructed power distribution.

Improvement of robustness and precision of uncertainty estimations has the same effect on the economy of the core management as the improvement of core calculations. Principal challenges include development of methods for determination of system similarity, in order to obtain the basis for specification of the range of applicability of calculated uncertainty and development of methods for propagation of uncertainties through the chain of computational models.

Fuel cycle efficiency can be enhanced by the successful core reload optimization: Multi-cycle core reload optimization, combined optimization of loading pattern and fresh fuel profiling and improvement of reliability of the optimization belong to the most important challenges in the area of core management.

4.1.4.5 Water chemistry and LLW management

Water chemistry is generally one of the key aspects for safety management in all light water nuclear power plants. Chemistry of primary coolant is crucial in the issues of protection against corrosion and build-up of radiation fields, and also against deterioration in heat transfer due to the formation of deposits on the surfaces of primary components.

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For higher effectiveness and also for prolongation of long term operation of each NPP it is necessary to develop new analytical methods for monitoring of corrosion processes and its products in every power plant circuit. It is also important to monitor the effects of corrective chemicals and boric acid solution on structural materials. In case of BWR reactors it is important to correlate the relations between oxidative-reductive potentials and water radiolysis.

Chemistry of primary water coolant is closely linked with LLW management, so in case of poor adjustment of water chemistry there would be an increased need for disposal of low-level waste. Likewise, radiation field build-up leads to decontaminations and so as to disposal of produced wastes from these processes. Radioactive wastes liquidation goes hand in hand with increasing operational costs. The main aim of this procedure is to minimize the amount of final rad-waste products, which is presently done by system of evaporators and that is very costly. The replacement of evaporators by membrane's technologies could lead to higher effectiveness and lower financial losses. Finally, reducing of significant amount of LLW would cause reduction in follow-up costs, such as rad-waste transportation, storage etc.

4.1.4.6 Radiation protection

Today, the overall context for radiation protection is such that, in the one hand, there is a need to account for the development and deployment of new nuclear technologies and renewed interest in large scale nuclear energy production, and in the other hand, geopolitical instability and global terrorism, create a black market for radioactive materials and ongoing attempts to acquire capacity in nuclear weapons. If the world is to realize the potential of radiation-based technologies for peaceful purposes, each country must be prepared to face the associated risks. Given this overall global context, one cannot escape from acknowledging that a significant threat can arise virtually at any time, and anywhere. Thus, there is a pressing need to strengthen the safety and security network at every level.

In normal operation normal operation of NPP, RP consists in various elements such as national regulatory infrastructure, radiological protection in occupational exposure, radiological protection in medical exposure, public and environmental radiological protection, all driven with concerns on good practice based on ALARA principles.

Important challenges are:

- to prepare effective methods, tools and regulations to keep the radioactive track into the environment as low as possible, with demonstrated minimum detrimental impact on life, e.g. to limit radioactive waste during operation of radioactive sources like NPPs,
- to prepare effective methods and tools to keep the exposure of personnel and/or population to radiation and other co-stressors as low as possible, with demonstrated minimum detrimental impact on life during normal operation,
- to avoid and or to drastically minimize the risk of human failures in the process of designing, operating and maintaining different kinds of radiation sources, including securing all these sources against thefts and misuse by criminals and/or terrorists and/or unstable governments,

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- to improve and periodically evaluate, review and inspect the technical, organizational and safety-cultural issues of radiation sources,

4.2 Sub Technical Areas (STA)

4.2.1 Improvement of the operation economics and NPP flexibility (STA 3.1)

4.2.1.1 Scope

Nuclear power plants are characteristic by enormous unit investment cost (EUR/MW), while operation (per electricity produced) is relatively cheap in comparison with other energy production alternatives. Investment cost reflects many factors (technical complexity, redundancy of systems, etc.) and generally gradually raised in last decades however there is potential for decrease under some conditions (competition, learning and fleet effects, modularization, standardization, construction schedules with sanctions, etc.). Detail investment cost analyses are outside of the scope of the NUGENIA Roadmap. Operational costs are usually split in two parts: a fixed part (EUR/MW/y), that is incurred whether or not the plant is generating electricity, and a variable part (EUR/MWh), which vary in relation to the output.

It is worth mentioning that actual full cost of production reflects also other factors like radioactive waste and spent nuclear fuel disposal costs, insurance of possible liabilities, reserves for decommissioning and rehabilitation or costs arising from additional requirements of regulatory bodies. For new projects, the LCOE (homogenized cost of electricity) is usually calculated based on a discounted cash flow model. LCOE is the price at which electricity must be generated to break even over the lifetime of the project (including an acceptable return on investment).

Optimization of operation offers potential to reach better competitive costs and is subject of huge engineering effort worldwide. Nevertheless, still there is potential for research and development to bring new approaches.

Most of the NPPs in operation were designed for base load mode of operation, but new market conditions could induce necessity or advantageousness to operate in the load-follow mode. Flexibility of NPPs could represent new factor of their competitiveness, especially in regions with highly expanded intermittent renewables.

4.2.1.2 State of the art

Nuclear power plants area benchmarked by many indicators in international organizations like International Atomic Energy Agency or World Association of Nuclear Operators. For example, the IAEA use the following indicators relevant to the NUGENIA TA3 (among many others):

- Nuclear operation and maintenance cost (USD/kWe)
- Refuelling outage or overhaul planned maintenance period (days)
- Fuel cost (USD/MWh)

Owners and operators of nuclear power plants currently mostly operating in deregulating competitive markets are under pressure to reduce operation cost to be more competitive with other energy production options.

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Therefore various aspects are considered to optimize operation of NPPs and maximize electricity produced:

- management of outages and maintenance,
- efficient use of nuclear fuel,
- power uprate (design reserves utilization, efficiency,…)
- other organizational factors like sharing of resources among several plants, etc.

NPPs could play important role in the stabilization of grid due to ability of operation in abnormal ranges (overload, frequency deviations, disorder close to disintegration of grid, etc.) and flexibility that is integral part of the Gen III reactors.

4.2.1.3 Challenges and research topics

4.2.1.4 Flexibility of nuclear power plants

Nuclear power plants as high capital intensive power production facilities were designed for operation to generate the maximum energy output in order to assure the appropriate return on investment and therefore nuclear power plants are to be operated preferably in base load mode. However, due to high proportion of nuclear energy production in the national electricity mix or high enlargement of intermittent energy sources the load following operating mode can be either an option or a necessity. Older NPP have limited eflexibility, on the other hand, modern light water reactors are designed to have large manoeuvring capabilities.

The load following ability of dispatchable power plants in comparison (OECD NEA, 2012)

	Start-up time	Maximal change in 30 sec	Maximum ramp rate (%/min)
Open cycle gas turbine (OCGT)	10-20 min	20-30%	20%/min
Combined cycle gas turbine (CCGT)	30-60 min	10-20%	5-10%/min
Coal plant	1-10 hours	5-10%	1-5%/min
Nuclear power plant	2 hours - 2 days	up to 5%	1-5%/min

NPPs under some conditions could provide various grid services:

- Primary (frequency) control
- Secondary and tertiary services
- Daily and weekly load-following
- Black start

ENTSO-E (European Network of Transmission System Operators for Electricity) has recently published new Network Codes. These new requirements will be incorporated into European Utility Requirements (EUR) which have been developed as a minimal expectations for new built NPPs.

The change of power level in pressurized water reactors could be perfomed by control rod movements and by changing the concentration of the boric acid in the primary circuit. Boiling water reactors make such change make this by changing coolant flow rate and control rods.R&D topics

Inherently, there are strong interlinks in many topics with the NUGENIA TA4 (aging of materials,…..). Flexibility issues are also solved in the TA1, mainly from the safety point of view.

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- modelling of scenarios of future energy mix in state and regional scale to derive various aspects: grid stabilization role of NPPs, potential requirements on flexibility of nuclear power plants, etc.
- to identify and assess impacts of flexible operation on reliability of critical components of NPPs (degradation, accelerated aging,..) and express it into lifetime management strategies
- to assess consequences of flexible operation on maintenance incl. modifications in monitoring and diagnostics and additional expenses induced by this mode of operation

4.2.1.5 Optimization of outages and maintenance

Maintenance activities include servicing, overhaul, repair and replacement of parts and may, as appropriate, include testing, calibration and in-service inspection.

Important factor directly influencing availability and then operation economics is outage duration and quality. Every nuclear power plant operator develops strategy for short and long term outage planning. Effort is directed on optimization of maintenance, refurbishment and diagnostics o for outages operation during outage while minimizing radiation exposures to personnel. Detail planning and training along with post-outage evaluations and feedbacks are integral part of overall strategy.

Outages could be categorized as follows (IAEA):

- refuelling and standard maintenance
- refuelling and extended maintenance
- special outages for major backfitting or plant modernization

R&D topics

- ✓ **development of new tools, organizational and technical measures for optimization of scope of activities performed and minimization of duration of outages**
- ✓ **minimization and effectiveness of cooling period during reactor shut-down and heating period during reactor start**

4.2.1.6 Advanced and integrative approaches to maintenance and lifetime management of systems

NPPs are managed by asset management principles that mean using resources to create maximum value for owners. Asset management should reflect possible long-term operation - ageing nuclear power stations will have to cope with the important task of replacing and refurbishing major components and systems in a cost-effective way. EPRI delivered variety of approaches and tools in this area, important information also comes from INPO (Institute of Nuclear Power Operations). EPRI asset management programme research and product development have addressed many key important aspects. Integrated Life Cycle Management (ILCM) for NPPs was elaborated by EPRI to

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complementary supplement existing programmes that maintain reliability, safety margins, design adequacy and licensing basis.

In spite of this effort, some ways and aspects should be further complementary explored and refined.

R&D topics

- ✓ **nuclear power plants extension cost and effectiveness analysis**
- ✓ **advanced asset management tools related to control of equipment ageing, control of sufficient safety margins and control of conformity with existing safety requirements**

4.2.2 Human and organizational factors (STA 3.2)

4.2.2.1 Scope

Human and Organizational Factors (HOFs) are key elements of the process of improving safety and efficiency of nuclear power plants operation. Both for the operated plants, and for those newly built, development of methodology to assess, monitor and improve performance of individual humans, as well as teams and organizations as a whole, is therefore a very important area for R&D to ensure that safety and operability of NPPs is maintained at the highest level.

All the individual functions and processes passing during plant operation are controlled by people, supported by technology. That's why any designed project carried out in this context should be considered as an act of creation of socio-technical system, not just as a design of hardware or as fitting the technology to the humans' needs. Consequently, an important factor for every design project (related to nuclear energetics) is a link with the physiological and cognitive properties both of the human designing the system and the human working inside and with the system and how these properties are shaped by the organizational framework and other non-technical features such as safety culture and management practices.

The organizational structures of the plant may either support or impede good interaction among individuals or teams. Other factors influencing interactions, which belong to the scope of HOF, are communication skills of the individual staff members and availability, quality and user friendliness of support tools, procedures and methods.

The evaluation of the dependability of the socio-technical system is one of the major issues of current R&D. Diverse methods should be applied to achieve credible and adequate assessment. In particular, human reliability considered in the space of risk scenarios is important element of safety and has to be addressed by adequate means. Since the variability of methods used for human reliability analysis all over the world is nowadays very high, harmonization of such approaches should be a challenge for the next future.

A significant overlap regarding risk oriented analysis of human factors/reliability may be found between TA-3 and TA-1, which may be solved the following way:

TA-1 should cover, basically, all the methods and processes internal to risk analysis. That means that TA-1 shall take all human factors related information (provided by TA-3!) and find the way, how to transform this information into the human error probabilities, frequencies of events initiated by human factors etc. and how to combine this information with the rest of PSA model to get the overall, plant risk level, risk contributors (including those related to human), importance measures

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(including importance measures of human actions) etc. The final product of TA-1 initiated and driven activities should be to provide a set of recommendations, how to decrease plant risk, related to human reliability and the factors, which influence it. The activities and outputs carried out in TA-1 should be mostly quantitative and risk related, significantly connected with PSA needs and outputs. The qualitative aspects of human factor analysis, which are much broader than those, which can be to covered by PSA in detail, should belong to TA-3. The features of (even risk oriented) decision making activities at NPP covering all qualitative aspects of conditions and factors influencing operators' work (training, procedures, MMI…..) should be covered by TA-3.

4.2.2.2 State of the art

A road map in the field of HOF was recently developed within the MMOTION project (2009-2011), funded in the FP7. The MMOTION partners identified four research programs (RPs), two of them being particularly relevant to the present topic:

RP1: "Risk-informed decision-making in design and operation" dedicated to balancing human and technological contributions to minimize the risk in the operation of nuclear installations

RP2: "Culture and practices for safety" aiming at better understanding the conditions for achieving robustness in the organization to minimize the risks in the processes of nuclear installations' operations.

The MMOTION outputs as the programs that also have a tight relationship with other safety and risk issues as discussed in the roadmap for NUGENIA TA1, provide the background for definition of challenges and research topics in this section. The results of MMOTION account for and integrate the outputs from various international projects, expert groups work etc., representing current state of the art in the human factors area, as for example:

- OECD/NEA working Group of Organizational and Human Factors (WGHOF)
- Industrial Safety Technology Platform (ETPIS)
- US DoE and National Laboratories effort
- IAEA initiatives on consideration of human factors in new NPP projects
- EDF research
- UK nuclear research index
- Nordic nuclear safety programme
- Broad MMOTION related activities of Halden Reactor Project.

4.2.2.3 Challenges and research topics

4.2.2.4 Application of risk informed decision making in human factor area

The goal is to strengthen the objectivity of safety judgments by using methods of risk-oriented decision making in human reliability area and to improve the cost-effectiveness and the balance of safety provisions. It is evident that understanding of the strengths and weaknesses of available methods to model human reliability within probabilistic risk assessments will be needed.

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An important challenge is to harmonize operation principles across Europe. As a prerequisite to this, the benefits and drawbacks of different operational principles and their feasibility need to be analyzed. Regarding this, it would be beneficial to put increased effort to development of advanced and harmonized methods and tools for evaluation of operating experience linked to human and organizational factors.

Another need is to understand the complexity of socio-technical systems, and to minimize the negative impacts of complexity on operation and safety, including design methodology.

It would also be important to develop deeper understanding of the MMO related integrated concepts for safety and risk management and to develop/improve tools for risk-informed decision making support in the human reliability area.

R&D topics

- ✓ **to develop tools and methods for analysis of operating experience feedback (OEF) particularly oriented to human and organizational factors, with the main goal to limit occurrence of the "repeated" events/errors,**
- ✓ **to develop consistent methodology for combination of human factor related operational experience with generic quantitative parameters used by the well-known human reliability analysis methods,**
- ✓ **to apply risk management techniques and practices as a part of decision making in the area of HOF and to support it by development of generally valid high quality methodologies.**

4.2.2.5 Safety culture and operating practices for safety analysis and improvement

The main objective in the research covered by this sub-area is optimizing the performance and robustness of the organizations in charge of operating nuclear power plants. The focus is on the human, organizational and management dimensions of the NPP system and underlying processes. The research should define the conditions required for ensuring the robustness of the organizations operating NPPs in future on the base of a deep understanding of practices and culture common for operation of existing plants. For the purposes of management of change, a theme how changes can impact the dimensions of this socio technical system and, consequently, its performance, will have to be addressed.

A deep understanding of how individuals, teams and organizations function and interact within the processes related to plant operation in the environment formed by a specific safety culture, and how these are supported by tools, artefacts, procedures, rules, etc. is necessary. Here, the emphasis is not put upon the original design or design changes affecting the technical system or its components, but, instead of that, the focus is on the human and organizational components of the socio-technical system, starting from psychological dimensions and reaching up to social and cultural dimensions of plant operation.

Currently, it has been found as important task to set up of theoretical models of representation of the functioning of the socio technical system, to develop methods and tools, and to benchmark operational practices in order to have a better knowledge and understanding of a number of key

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issues. Hence, research needs are currently found in understanding better the factors influencing safety culture, how to measure them as well as how to estimate the impact on the plant performance. The safety culture influence is estimated (and proved in practice, particularly nowadays) as strong, but it still deserves further deep analysis. In addition, it would also be necessary to improve understanding of the impact of changes in the organization's (NPPs) external conditions on their robustness and vulnerability.

Another issue, in which knowledge needs are clear, is management practices in different NPP's, teams and European countries and their generic strengths and weaknesses. Further needs for knowledge can be found in understanding the potential effects of new technologies on staffing and the dependence of staffing requirements on the plant's lifecycle. Methodical studies should be accomplished to understand and significantly improve the effectiveness of tools applied to assess human performance. One of the safety critical issues would be assessment of workload in complex tasks with multiple operators engaged in multiple and potentially conflicting sub-tasks.

R&D topics

- ✓ **to set up of theoretical models of representation of the functioning of nuclear power plant as the socio technical system and to use these models directly for safety culture enhancement,**
- ✓ **to develop methods and tools, and to benchmark operational practices in order to have a better knowledge and understanding of a number of key issues:**
    - o **safety culture of nuclear organizations, safety management policies and practices,**
    - o **teamwork in operating organizations and in different operational conditions,**
    - o **occupational health, robustness and vulnerability of organizations,**
    - o **staffing and changing concepts of operations,**
    - o **impact of dynamically changed external conditions,**
- ✓ **to accomplish methodical studies to understand effectiveness of the tools commonly applied to assess human performance:**
    - o **individual human performance assessment tools,**
    - o **training effectiveness assessment tools,**
    - o **teamwork assessment tools,**
    - o **organizational safety culture assessment tools,**
- ✓ **to develop a methodology for assessing the impact of safety culture on plant performance,**
- ✓ **to handle socio-technical systems complexity at the design level and in management of modifications.**

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4.2.2.6 Development of socio-technical approach to improve management of human resources under circumstances with dynamic features

The Fukushima accident revealed significant weaknesses in the on-site and off-site response to extreme, unforeseen events, in particular when the functioning of the emergency organization itself is weakened by the event. Better and more resilient socio-technical solutions are needed for inter-organizational collaboration (including plant, utility, emergency services, regulator and government agencies), for making time-critical decisions based on dynamic and partially unreliable information, and for rapidly transporting all kinds of resources (e.g. generators, equipment, robots, experts) to a damaged plant.

Emergency may not be the only plant condition which could require a specific organization of dynamic processes. Plant outage conditions taking advantage from “ad hoc” organization, control and management are also typical with higher level of dynamics and not fully predictable course. There are different practices how the outage is planned, organized and controlled in different companies: in some companies there is a specific organization element acting during the outage, whereas in the others there are no basic changes in respect to the normal operation from this point of view. It is likely (and it should be one of the goals of the work) that also interactions with other company functions (procurement) may be optimized to make the plant outage management more effective.

It is evident that the role of human factor in dynamic situations, which cannot be completely planned and covered by detailed instructions, is critical and that the potential deficiencies in human and organizational factors area may impact both safety and operational efficiency during these sensitive time periods of not fully pre-determined plant configurations. For that reason, human factors within the context of outage or emergency situation are of extreme importance, but, also typical with higher level of uncertainty.

R&D topics

- ✓ **to develop better and more resilient socio-technical solutions for both intra- and inter-organizational collaboration for making time-critical decisions based on dynamic and partially unreliable information,**
- ✓ **to address issues connected with (human factor related) requirements on moving and rapidly transporting resources to the location of need, taking into account:**
  - o **the current emergency preparedness practices adopted by different actors in Europe,**
  - o **decision-making process in emergency-management centres of utilities from the of human and organizational factors point of view,**
  - o **the principles of functioning of shared resource centres for rapid response,**
- ✓ **to improve (optimize) outage planning and management using socio-technical approach.**

4.2.2.7 Improvement of training based on simulation of plant processes

Numerical simulation and use of high performance computers may represent an important tool for achieving new targets in the development of simulators and computerized operation support systems. The calculations requiring several months years ago need just few hours, minutes or less

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now and will require few seconds in the next future. As more and more plant information will become available in a digital form, it will be possible to develop tools able to provide real-time situational awareness for operators and to predict the future plant state based on the most current conditions and trends. It will also become possible to use advanced nuclear, thermal-hydraulic, and electrical models to assess the actual plant performance relative to the predicted plant performance and to report deviations and trends to the operators, using directly-measured parameters for analysis of plant performance and recognition of the effect of instrument malfunctioning.

A faster-than-real-time simulator could predict the effect of operator actions prior to taking them, in order to check their consequences. This would detect interactions that might not be apparent to the operator due to unusual plant configurations and other operating restrictions. For instance, it could project the timing of the gradual effect of actions such as boration and dilution on reactor power. Depending on the fidelity of the simulator, such predictions could be very helpful in off-normal conditions where the emergency procedures cannot anticipate every combination of component unavailability's and emergency conditions including severe accidents.

R&D topics

- ✓ **to use simulator data collection in systematic manner for human factor related data collection and analysis, to develop methodology and processes for this task,**
- ✓ **to use the data from simulator data collection systematically for searching for week points in emergency procedures used by operators and for improvement of these points, to develop methodology for this task,**
- ✓ **to collect and use the data from simulator data collection systematically for up-date of human error probability values in plant PSA project,**
- ✓ **to provide more advanced tools for operator training, capable of direct visualization of the phenomena as they occur in the circuit, thus improving the understanding of any situation,**
- ✓ **To provide more advanced 'post Fukushima' engineering/training simulators oriented to the scenarios of extreme natural events (earthquakes, tsunamis, hurricanes, tornadoes, external fires…..), including the multi-events and multi-unit NPPs aspects.**

4.2.3 Integration of Advanced Digital Technologies (STA 3.3)

4.2.3.1 Scope

Implementation of modern digital technologies offers a unique opportunity for improving operational performance, enhancing nuclear safety and supporting life extension of LWRs.

Digital technologies are nowadays deployed in all modern power generation plants and also in large industrial plants and devices characterized by a relevant risk level. The situation in the nuclear power sector differs from the rest of the industry in the following key aspects:

- the use of analogue systems is to be extended far beyond their expected service lifetime,

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- complexity of safety demonstration and cost issues are the main barriers to wider application of digital technology in the I&C systems.

As a consequence, R&D work is still necessary before utilities are ready to take full advantage of digital technologies to achieve performance gains. Even if several European plants have gone through relevant modernization programs, many interventions have been performed under the constraint of minimizing the impact on the traditional way of operating, maintaining and managing the plant. Digital technologies are often implemented to provide solutions to specific problems and immediate needs, and not in a long term perspective. Due to the rapid evolution of these technologies, this approach is not sustainable in the long term, and it needs to be modified in the future.

4.2.3.2 State of the art

Defining a strategy for the implementation of fully digital system is becoming a more and more urgent issue for the life extension of the Generation II reactors, as well as for the deployment of the Generation III. It offers an opportunity to test the capability of European nuclear institutions to introduce a higher level of harmonization and eventually to reduce the insularity of the nuclear industry promoting the interaction with other sectors, where the practice is more advanced.

A huge research program in this field is being prepared in US by the Idaho National Laboratory, potentially influencing the R&D activity also in Europe. Among other targets, this program intends to develop standards and guidelines to facilitate the transition to digital technology and its deployment across the US nuclear fleet. In this program five key technical areas have been identified, as follows: Highly Integrated Control Room, Highly Automated Plant, Human Performance Improvement for NPP Field Workers, Integrated Operations, Outage Safety and Efficiency. For each area various projects will be performed, each one in cooperation with a hosting utility, for a total of 18 pilot projects scheduled in the decade 2012-2021.

To maintain competitiveness Europe (industrial and research organization) needs to develop its own expertise and vision. Research in this field has been carried on in Europe mainly by utilities and vendors and by some excellence centre, while very few projects have been funded by the European Framework Programs. The already mentioned MMOTION roadmap is a part of this effort; it identifies two research programs to cover the HOF research needs with regard to the design and upgrading of NPPs for the future. These were:

RP3: "Integrated design approaches" focusing on better integrating human and organizational factors within the design of future nuclear installations or the renewal of existing I&C systems,

RP4: "Technological requirements in nuclear and other high risk industry" aimed at achieving better coherence between the products offered by the industry for I&C systems and the specific needs of the nuclear and other high risk industries.

The content of RP3 and RP4 has been integrated into the description of the following challenges.

4.2.3.3 Challenges and research topics

4.2.3.4 3.3-1: Innovative I&C Design and Architecture

Many digital technologies are already on the market and widely used in industrial installations. They offer a huge potential for an application in NPPs, if used as new plant infrastructures, suitable to be

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used for different applications, ranging from normal operation to maintenance optimization, from radiation protection to emergency management. The main goal for a R&D program will be to develop advanced digital architecture for information and control systems and for protection systems.

R&D topics

- ✓ **to develop a new advanced digital information and control architecture to integrate different plant application,**
- ✓ **to prepare a general framework for I&C design specification (including SW design unification and protection function technical specification unification),**
- ✓ **to define principles of Reliable Design - Dependency minimization regarding Common Cause Failures (CCFs) - "free" design, low consumption equipment + local backup power supply, no Air Condition requirements, …):**
  - o **to develop methods supporting the suitability of fault modes and effects analysis for regulatory assurance of complex logic in digital instrumentation and control systems,**
  - o **to develop methods and tools to assess if the level covered by the functional tests of the software is appropriate,**
  - o **to develop methods and tools for assessment of propagation of numerical errors,**
  - o **to develop methods and tools to assess multi tasks and real time software,**
  - o **to develop methods and tools to assess Field Programmable Gate Array (FPGA).**

4.2.3.5 Integration of new technologies in I&C

Application of new modern electronic technologies to increase operability, decrease the implementation and maintenance burdens with concurrent comfort enrichment and without threatening of safety.

The particular challenges in new technologies are:

- ➢ new materials,
- ➢ smart equipment, smart instrumentation,
- ➢ highly qualified instrumentation,
- ➢ new digital technologies (like Field Programmable Gate Array (FPGA) based devices,
- ➢ modern communication means (e.g. wireless; in this case it is worth considering the cyber-security challenges, too).

Many innovative digital technologies are broadly and rapidly implemented in the modern industry including electricity production except the nuclear ones. The reason is long life cycle of NPP equipment in general and complicated and time consumed qualification of any equipment which is either safety related or could interfere with them.

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Wireless communication is an example of technology, which could be very helpful both in the frame of routine operation of the plant, when increase the comfort and lower burdens on cabling and could be the only way how to get information in the case of severe accident. This appears as a relatively low cost infrastructure able to rapidly improve the communication capability inside generation plants, and it is actually already used in some plants, for example during outage for improving communication between the field and the outage control centre. More generally, wireless communication will help in improving the communication between all individuals, in particular in the field, thus reducing the risk of mistake due to bad understanding of each other. Starting from functions not classified as relevant for safety, we expect that the technology may be deployed in the NPPs, demonstrating the feasibility and the reliability of such a protocol. However, the strong opposition among plant operators exists to minimize the use of any wireless equipment at least within the restricted area. The advantage of wireless technology implementation in the primary circuit equipment and sensors diagnostic as well as source of the information in the case of the loss of networks is nevertheless obvious.

Application of digital technologies in high radiation environment, which is often combined with high temperature, requires the development of specific devices. Various technologies are under consideration for the construction of devices capable to operate in this kind of hostile environment:

- by materials such as Silicon Carbide (SiC) or Silicon On Insulator (SOI),
- by design such as Field Programmable Gate Array (FPGA)'s,
- by packaging of commercially available electronics.

Other technologies have not yet fully developed their potential, but they will be certainly able to extend their possible application in the future. Development of nano-sensors may provide a cheap and sustainable way to increase diversity and redundancy of measurements.

R&D topics

- ✓ **to research applicability of the use of new technologies in the NPP unit operation, specifically (but not limited to):**
    - **smart equipment and instrumentation (Self-diagnostic, Self-healing, Auto-calibration, …),**
    - **the wireless technologies in nuclear power plants with following research targets wireless technologies:**
        - **segmentation of the applications,**
        - **benchmarking of relevant technologies (lab tests and in-situ tests),**
        - **qualification of wireless solutions for regulation bodies acceptance (coexistence with I&C systems, wireless networks coexistence),**
        - **standardization (EMC emissions, communication protocols, data models…).**
    - **field programmable gate arrays (FPGAs), highly qualified instrumentation for post and severe accident monitoring,**

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- **qualification approaches for the using off-the shelf products and the underlying technologies for basic applications in the nuclear field, to develop new digital technologies th**
    - **high level of the irradiation;**
    - **cheap and sustainable way to increase diversity and redundancy of measurements.**

4.2.3.6 I&C life-cycle

To cope with discrepancy between long life cycle of nuclear installations in the comparison with short life-cycle of electronic I&C components.

The main objective is the optimization of NPP I&C lifecycle through a better consideration of the specific needs of relevant installations by I&C suppliers with special regard to integrating standardized "high-risk industry" requirements with respect to safety, availability and usability. The relevant installations include nuclear facilities and facilities of other high risk industries which generally use similar equipment.

The research should deal with a range of current challenges. While more powerful, digital technology is also more complex, requiring expertise in several fields such as automation, computer science, networks, ergonomics, etc., digital technology also has much shorter lifecycles. Complexity impact on design, industrial arrangements, maintenance, potentially operation and may significantly increase the costs. Finally, engineering and validating nuclear I&C systems, for renewal or new build, is much more complicated, time-consuming, costly and less sustainable than before. Moreover, compared to other industries, the I&C domain in nuclear and other "high-risk" industries is constrained by stronger requirements, has a very long life cycle, cannot change rapidly, and represents a comparatively small market. A specific challenge is managing an existing fleet of power plants presently with an average age of 30 years over an extended period of time when the basic technology has grown more complex, with a volatile market stipulating even more severe financial conditions.

R&D topics

- ✓ **to perform cost effective I&C upgrade (including future upgrades) including upgrade implementation planning during the plant operation,**
- ✓ **to develop approaches for I&C equipment qualification (reasonable and cost effective) including equipment lifetime prolongation, equipment calibration period maintenance, developing sustainable I&C specifications and design methods do deal with shorter I&C lifecycle and capable to be shared across Europe in order to ease future I&C modernization programmes.**

4.2.3.7 Advanced Diagnostics

To enhance I&C systems and NPP equipment reliability and availability by the use of advanced diagnostic, condition monitoring with prediction features.

The particular sub-challenges in advanced diagnostics are:

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- Advanced diagnostic tools including prediction features which increase I&C systems reliability,
- Equipment condition monitoring (including data reconciliation analysis) to increase NPP equipment reliability and lower maintenance costs,
- Specific architecture to embed and gain advantage from data from different types of sources of condition-monitoring data.

Ways of reliability of systems improvement are limited. The increase of redundancy is not very efficient from certain level and is very expensive both from the investment and operation (maintenance) point of view. The use of diversity improves reliability but it is even more costly. The early diagnostic with repair or inhibition of failed equipment/instrumentation could improve reliability of systems significantly.

NPPs include thousands of components which are currently not monitored continuously. New technologies may in fact allow to extend monitoring and diagnostics activities to a series of very numerous small components, like electric devices, valves, pumps, etc…, to predict their performance and to organize the data in a large data base.

New systems based on different kind of sensors, data storage and analysis can help in optimizing maintenance activity, on the basis of the real need, preventing faults and hence enhancing plant reliability.

These systems could be coupled with advanced methodologies and modern tools for reliability calculation (including both quantitative and qualitative reliability variables) and risk assessment, based on probabilistic approach, under development in various laboratories. On the other side, information technology could support an advanced centralized online monitoring centre to collect all information from each plant and conduct long-term plant asset management. To this purpose it is necessary to develop a specific digital architecture to collect and organize data from all types of sources of condition-monitoring data, and to provide guidance and associated technical studies to develop guidelines for a wide implementation.

R&D topics

- ✓ **to develop advanced diagnostic tools including prediction features etc.,**
- ✓ **to prepare methodology and means for equipment condition monitoring (data reconciliation analysis),**
- ✓ **to develop a specific digital architecture to collect and organize data from all types of sources of condition-monitoring data,**
- ✓ **to provide guidance, associated technical studies and guidelines for a wide implementation of the new ideas.**

4.2.3.8 Cyber security

Establishing of common European stand position in the field of protection of nuclear installations against cybernetic attacks, to harmonize Cyber Security Policy in the nuclear area via Cyber Security Program template and pilot Cyber Security.

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Protection of computer based systems against malicious acts is more and more an issue in all the industry and services. NPP operation was felt to be untouched for quite a long time due to its specific design of information systems and analogue equipment in protection systems. However the digital systems and even Commercial-Of-The-Shelf technologies are implemented. It is necessary to protect the NPP systems against cyber-attacks both from the safety and the performance point of view.

The necessity to protect programmable systems important for safety against malicious cybernetic attracts is well known and very often emphasizes. However common European approach to this issue is needed. It must be structured and different for particular types of threats and potential consequences and must cover basic building elements like Cyber Security Policy, Program, Plan, etc.

R&D topics

- ✓ **to develop Common European Cyber Security Policy for NPP,**
- ✓ **to develop Utility Cyber Pilot Security Plan,**
- ✓ **to develop NPP Cyber Security Program template and guidance**
- ✓ **cyber security of non-safety related systems information and control systems,**
- ✓ **cyber security of safety related (including protection) systems,**
- ✓ **cyber security for integrated design – safe intersystem communication.**

4.2.3.9 Human performance support

The challenge concerns two following areas:

- operator support - Highly Integrated Digitalized Control Room, advanced alarm systems, advanced visualization techniques, standardization of operator interface screens and control board layout,
- maintenance and field workers support.

For achieving better integration of technical, human and organizational aspects in the various activities of the design process, more focus needs to be put on the efficient integration of HOF requirements in the design by the prediction and timely assessment of the future work conditions and through the active participation of HOF specialists and end users. In order to integrate human and organizational factors in to the overall design process a specific Human Factors Engineering process needs to be organized to manage the design-oriented human factors research and development.

Introduction of digital technologies in analogue control rooms requires application of human factor engineering principles, which are in continuous evolution.

There is a need to develop guidelines based on these principles, shared by regulators, to allow the standardization of operator interface screens and control board layout. The development of an advanced alarm systems is also a possibility offered by digital technology. The traditional alarm systems are facing reliability and obsolescence issues, and are not optimized for highest operator performance, since they are not designed to put information in the context of actual plant conditions and mode of operation.

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Modern digital technologies can be more helpful for further progress in the operator training improvement and lead to a significant reduction of human errors.

Several kinds of mobile communication/computing devices may be used to provide continual plant status and control capability anywhere in the plant. Plant workers equipped with suitable portable devices will have direct access to any information they need, eliminating the need for carrying huge paper packages, and will report quickly about the results of their job once it is finished. Tools using virtual reality models and technologies can be used to develop computer aided maintenance procedures and to train maintenance personnel. Use of e.g. Radio Frequency Identification (RFID) technology could allow making sure that people are working on the right piece of equipment.

By maximizing the “collective situational awareness” of the entire plant, technological advancements will greatly improve critical decision making at all levels.

R&D topics

- ✓ **to develop Highly Integrated Digitalized Control Rooms (including control room task analysis regarding weaknesses),**
- ✓ **to develop maintenance and field workers support (task automation, electronic procedures etc.),**
- ✓ **to develop and promote design methods and the associated organization principles that are able to effectively integrate the “non-technical” design requirements (e.g. human and organizational aspects) into the design process by anticipating future work situations at very early design stages,**
- ✓ **to develop modelling framework supporting and connecting the viewpoints of the various categories of professionals involved in the design activities,**
- ✓ **to define a set of practical criteria to predict the adaptation of different Human System Interface (HIS) solutions to the different operational principles,**
- ✓ **to develop methods and tools for using work practices and task analyses as a basis of HSI specification and early evaluation,**
- ✓ **to define concepts and requirements for the automation system and the system complexity ensuring that the operator is kept in the loop,**
- ✓ **to develop methodology and schemes for cross identification of the relevance of different HSI characteristics with respect to operational activities and work practices,**
- ✓ **to construct approaches, methods and tools for designing and maintaining systems in a way leading to avoiding HF problems such as information overload and lack of system transparency,**
- ✓ **to prepare methods and tools to assess the usability during the design process and to validate the usability of the final product in its context of use,**
- ✓ **to develop and apply advanced methods for the verification and validation of complex socio-technical systems both during the design process (to provide early feedback to the design) and for final verification and validation (V&V.),**

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- ✓ **to develop suitable procedures and advanced tools to support field workers,**
- ✓ **to develop advanced alarm systems,**
    - **provide advanced visualization techniques for better management of information,**
    - **develop guidelines allowing standardization of operator interface screens and control board layout,**
- ✓ **to evaluate emergency operating procedures structure, coordination and interactions.**

4.2.3.10 Pre-normative research in I&C

Even if standardization in Europe concerning the I&C is placed under the responsibility of the European Committee for Standardization - European Committee for Electro-technical Standardization (CEN CENELEC), no real and major pre-normative R&D oriented actions are presently conducted at the industrial level.

Nevertheless, spending R&D efforts on pre-normative activities would be very worth in the fields of endeavor where technologies invented and applied in other industrial sectors, such as avionic and high technology, with different regulatory requirements and relying upon different safety approaches and levels - could be relevant to nuclear industry and likely to be extended or adapted and implemented in the nuclear sector.

For instance, the work is underway for a standardization of reliability assessment and the safety demonstration of programmable logic devices, like FPGAs (Field Programmable Gate Arrays). Another candidate for such a pre-normative R&D is the standardization of safety-grade graphic man-machine interfaces. Moreover, the nuclear industry could take benefit from the work already done in the aerospace industry on the same topic.

Among other R&D Challenges in the field of reliability assessment of safety and protection systems of NPPs, we mention:

- ➢ Investigating the bulk logic of multi-core processors, which are foreseen to be used in NPP safety I&C. That engenders problems of fully novel nature, because, in such devices, different cores access a common memory unit, nevertheless the component features do not guarantee that they see the same information (so-called "memory consistency problem");
- ➢ Investigating the cyber-physics interaction phenomena of hybrid systems. NPP discrete-time control systems (not operating in continuous) interact with a physical environment which is fully deployed through continuous differential equations. How can robustness and of such systems be investigated and which kind of proof may be provided (e.g. guarantee that the closed-loop system will stay within a given physical sub-domain)? The question is open now and merits for further investigation.

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4.2.4 Core management (STA 3.4)

4.2.4.1 Scope

The management of a LWR core has an objective to maximize the cycle energy production at minimal fuel cycle costs while it has as constraint to minimize the power peaking factors in order to ensure sufficient margins to operating and safety criteria.

Several crucial activities influence the success of core management: core reload optimization and core reload safety analysis prior to the core reloading, and core monitoring during the core operation. Prediction of used fuel isotopic composition, including its uncertainty, is important mainly for the fuel cycle back-end applications. All these activities rely heavily on numerical simulation of core and fuel behaviour during the operation: methods of numerical modelling are thus of big importance for core management, as well as methods used for estimation of uncertainties of evaluated parameters.

The broad topic of core management can be thus subdivided into several distinct areas, which, even if partly overlapping, will be further discussed independently:

- core and fuel design calculations and numerical modelling including normal and degraded operation and evaluation of isotopic composition of unloaded fuel,
- evaluation of uncertainties,
- core reload optimization,
- core monitoring and instrumentation.

The above mentioned areas are described more in detail in the following paragraphs.

4.2.4.1.1 Core and fuel design related calculations and numerical modelling

The core and fuel system design calculations include Nuclear Design, Thermal-Hydraulic design and Fuel Rod and mechanical design (thermo-mechanical design and structural behaviour of fuel assemblies and of the core as a whole).

Nuclear Design serves for verification that during the normal operation the core behaviour will be within the assumptions of safety analysis. It covers evaluation of core characteristics, such as local pin powers, integral pin powers, control rod worth, reactivity feedback coefficients and cycle length (reactivity change with burn-up).

Thermal Hydraulic design serves for assessment that thermal and hydraulic conditions guarantee reliable core and fuel cooling. It covers analysis of core flow and analysis of departure from nucleate boiling. Fuel Rod design and Mechanical design serve for verification that the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences. It covers for example evaluation of fuel temperature and fission gas release during core operation.

The Fuel rod design and Structural design are covered mainly in the Technical Area TA-5 ("Fuel development"), but they are discussed here as well for the sake of completeness. The scope of "Core and fuel design related calculations and numerical modelling" includes also methods for evaluation of isotopic composition of spent fuel, which is important for criticality, shielding and residual heat calculations.

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4.2.4.1.2 Core monitoring and instrumentation

The Core monitoring system provides the plant personnel (reactor physicists, core operators) continuous real time core power distribution and technical specification surveillance, based on in-core instrumentation data and numerical simulation of reactor operation.

The core monitoring system has got three main functionalities:

i. monitoring functions - on-line information on the core state,
ii. analytical functions – analysis of the core state, discrepancies measurements/predictions etc.
iii. predictive functions – optimization and analysis of anticipated transients (load swings etc).

As stated above, the core monitoring is dependent on the data from core instrumentation: the core instrumentation itself, the interpretation of measured data and estimation of measurement error (including non-design fuel system conditions – e.g. geometrical distortions) are also covered in this topic.

4.2.4.1.3 Uncertainties

The trend in reactor simulations and deterministic safety analysis is to move from conservative approaches to so-called best-estimate plus uncertainty methodologies. The objective is to perform design calculations and safety assessments using more accurate physics-based simulation tools supplemented with rigorous quantifications of the uncertainties associated to the predicted physical parameters.

The focus here is put upon uncertainties in core design calculations (nuclear physics, thermo-mechanics and thermo-hydraulics) and relevant safety assessments, including criticality safety calculations. In addition, non-design fuel system conditions – e.g. geometrical distortions – are also observed in this topic.

4.2.4.1.4 Core reload optimization

This topic covers the optimization of the rearrangement of all the fuel assemblies (the old and fresh ones) and design of fresh fuel assemblies (enrichment, profiling, distribution of burnable absorbers) via optimization of the suitable objective function (typically maximization of cycle length) together with set of safety and operational constraints (power peaking factors and other safety parameters, operational restrictions or requirements).

The main goals of core reload optimization are:

- maximizing the length of cycle,
- maximizing the fuel utilization,
- minimizing the vessel fluency and other “aging” effects,
- fulfilment of all operational and safety constraints.

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4.2.4.2 State of the art

4.2.4.2.1 Core calculations & numerical modelling

In engineering practice the nuclear design, Thermal-hydraulic design and Fuel rod and Fuel assembly (thermo) mechanical design are done either independently or with very simplified interrelated feedbacks.

The Nuclear Design calculations are commonly done through solution of diffusion equation by 3D steady state multi group nodal codes. The constants for the diffusion approximation are evaluated by 2D lattice codes in the approximation of the infinite array of fuel assemblies. The core simulators use generally simplified model for thermo-hydraulics, which accounts for 1D flow in the fuel assembly, the cross flow among neighbour assemblies is neglected. Tabulated fuel temperatures (or "effective fuel temperatures") as a function of power and moderator temperature are used. They are pre-calculated offline by fuel performance codes. Further refinements of above described approach can be applied specifically by:

- introducing transport corrections with the diffusion approximation (for example SP1 or SP3 method) or direct 3D transport calculations,
- developing advanced thermal-hydraulics models (cross-flow among fuel assemblies),
- increasing number of energy groups (4 or more instead of 1.5 or 2),
- refining lattice calculations by going beyond the approximation of infinite array of fuel assemblies,
- implementing direct 3D pin-wise calculations instead of nodal calculations.

The Thermal Hydraulic design is done using the so called "sub-channel analysis" tools, used to predict the 3D velocity, pressure and thermal energy fields for single and two-phase flow. The codes are commonly based on solution of finite different equations for mass, energy and momentum conservation for and interconnected array of channels. The DNBR margin is checked using the correlations for critical heat flux – empirical functions based on extensive experiments with a given fuel assembly design.

The tools for the fuel modelling are discussed in TA5, but the summary is given here for the completeness. Fuel Rod design is done using so called "Fuel performance" codes based on the inputs generated by the nuclear design. Currently, most of the codes employ the "1.5D" approach, i.e. the equations (heat transfer, diffusion and restructuring, mechanics...) are solved radially in the idealized cylindrical geometry at several planes along the fuel rod height and coupled though the integral quantities such as rod internal pressure and the mechanical axial equilibrium balance conditions.

The Fuel Assembly mechanical design is performed mostly using the finite element based codes or models based on beam theory. These calculations are currently performed only as a part of the Fuel Assembly design and they are not taking into account specific reload schemes.

The main problem of estimation of used fuel isotopic composition resides in software validation: experimental data are relatively scarce and the error estimation (both of measurements and predictions) is difficult. That is why the traditional approach in the areas where the isotopic composition of fuel is accounted for explicitly is very conservative: approach to the criticality safety analysis is based on the principle that the analysis is done for the most adverse conditions – the fuel burn-up, burnable absorbers, diluted boron etc. are not taken into account, conservative shielding

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and decay heat analysis results should be verified by measurements. Improvement in analytical methods allows to reduce the conservatism (for example to take into account the reactivity decrease due to the fuel burn-up or presence of burnable absorbers). The conservatism is nevertheless still relatively important due to large error estimates.

4.2.4.2.2 Core monitoring and instrumentation

The core monitoring system gives the best available information on the in-core power distribution. It relies on two main data sources. The first are the in-core measurements. The scope of measured data depends on the reactor design. In current generation of nuclear reactors the main sources of measured data are the moveable or fixed detectors of neutron flux and the thermocouples. The second data source is the independent numerical simulation of core operation. The core simulator is usually based on the same code as the one used for core design.

The estimation of in-core power distribution is done by so-called “power reconstruction” process, where the measured data are validated and then “extrapolated” to non-instrumented positions using the results of numerical simulation. Alternatively, the measurements are used to estimate the uncertainty of the results of the numerical simulation. The measurement can also be used to estimate the extent of the fuel and core reconfiguration (fuel assembly bow, etc.).

The main purpose of the core monitoring system is the validation of computational model used for core design and on-line control of the core operation safety criteria: these are usually given indirectly by the plant technical specifications such as power peaking factors, fuel outlet temperatures, fuel temperature rise, axial flux difference etc. Some core monitoring systems proceed to direct monitoring of the limiting physical phenomena such as Departure from Nucleate Boiling (DNB).

4.2.4.2.3 Uncertainties

There are two approaches for uncertainty quantification.

The first approach – “analytical” - relies on quantification of upstream sources of uncertainties (composition, data libraries …) and modelling uncertainties and their propagation through the chain of computational codes – either directly or via the sensitivity analysis. The main challenges become therefore threefold: to identify the main and relevant sources of uncertainties; to determine objectively the uncertainty ranges and distributions, and to develop methods to propagate these uncertainties and to quantify their effects on the output quantities of interest. The second approach relies on uncertainties quantification via comparison to experimental data. Here the main challenge becomes the determination of validity range of uncertainties determined in this way. In practice, when it comes to uncertainty determination, both above stated approaches often overlap.

For core physics, the uncertainties in predicted fuel assembly power distributions, feedback coefficients, criticality conditions and other parameters can be, for steady-state conditions, rather well quantified by means of comparisons to measurements. For this purpose, the data collected during the physics start-up tests and reactor operation is used. Nevertheless, as mentioned above, important question arises in connection with the extent of applicability of such uncertainties: for example if they can be applied for core states in abnormal conditions i.e. during transient/accident conditions.

For more detailed information such as pin power distribution no measurements are available, so combination with “analytical” approach is necessary, where uncertainties coming from pin power

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reconstruction process, fuel assembly bow, material and size tolerances, thermo-mechanics (fuel temperature), thermo-hydraulics (moderator density) etc. should be accounted for.

The evaluation of uncertainties coming from nuclear data is currently an active area of research to advance state-of-the-art approaches. For simple cases with a well-defined target quantity such as the effective multiplication factor, a state-of-the-art approach is to apply perturbation-theory based methods to estimate sensitivity coefficients to individual data components (e.g. a cross-section for a particular nuclide, reaction and energy group) and use these in conjunction with nuclear data variance/covariance matrices (VCM) to estimate the final output uncertainty. For more complex cases alternative methods relying on statistical sampling are being developed nowadays, which are based on Monte Carlo calculations over nuclear data. The simple idea is to repeat the same reactor physics calculation a large number of times, randomly varying the nuclear data models to produce a new nuclear data library each time and use each distinct library in a downstream unique reactor physic calculation. Each of these calculations will produce a different result, allowing thereby from the combined set of results, to estimate probability distributions for quantities such as Keff (eigenvalue), material inventory, void coefficient and so on.

For thermal-hydraulics, uncertainty quantifications are currently mainly aimed at applications with system or sub-channel codes. In that framework, a certain maturity has already been achieved in applying statistical sampling methods for perturbation and propagation of input model parameter uncertainties in simulations of transients or design basis accidents. However, the remaining challenge is that these codes rely on a wide range of closure laws/constitutive relationships to complement and solve the mass/momentum/energy conservation equations. This includes empirical correlations, mechanistic “physical” models and semi-empirical correlations. These components constitute one of the main sources of uncertainties in thermal-hydraulic codes. And since such codes are applied not only for pure safety analyses but also with coupling to neutronics for conventional core analyses, these uncertainties will also play a major role in predicting accurately core reactivity and/or power distributions.

In thermo-mechanics, best-estimate approach combined with conservative input data is currently applied. The uncertainty determination for key parameters, such as fuel pin internal gas pressure is problematic due to limited amount of experimental data and complexity of the problem. Concerning multi-physics coupled neutronics/thermal-hydraulic simulations, appropriate uncertainty quantifications can only be achieved once methodologies have been developed and assessed for each of the separate areas.

In the area of criticality safety calculations the uncertainty is estimated during the code validation process against the experimental data, which serve for quantification of computational biases and the uncertainty in the biases. The safety system must be sufficiently similar to the experiments to be within applicability of the experiments chosen for validation. Traditionally, the similarity of systems is determined via comparison of several parameters deemed adequate estimators of the likeness between two systems. The determination of the “measure of similarity” of two systems is currently an active area of research.

Apart from core calculations, the prediction of isotopic compositions during depletion is another area highly affected by nuclear data uncertainties related not only to cross-sections but also to fission yields, branching ratios and decay constants. Therefore, several organizations are now working on extending the uncertainty propagation methods to be applicable also for burn-up calculations.

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4.2.4.2.4 Core reload optimization

Core reload optimization is a discrete optimization problem, which is computationally infeasible by current combinatorial methods, due to the huge number of permutations and the complexity of required neutronics computation. Many numerical methods have been proposed for solving number of commercial software packages have been written to support fuel management.

Current core reload optimization algorithms require typically evaluation the characteristics of tens of thousands reloads, which is infeasible by current 3D core design solvers: that is why the calculations are done either in 2D, or by simplified 3D solvers: both approaches induce error in the selection of the "optimal" reload scheme. The operators usually use a combination of computational and empirical techniques to manage this problem. The choice of fresh fuel inventory is also based on "expert opinion": currently the fresh fuel inventory optimization is not part of the core optimization software.

4.2.4.3 Challenges and research topics

4.2.4.4 Improvement of precision of core calculation and numerical modelling

The economy of core management is directly influenced by the precision of core calculations and numerical modelling. All aspects of modelling are concerned: precision of criticality and power distribution predictions, modelling of fuel behaviour and thermal-hydraulics and predictions of used fuel isotopic composition. Several examples are given below.

Predictions of power distribution: The economy of core operation is driven by the efficiency and reliability of its operation and by efficiency of usage of nuclear fuel. The best fuel economy is obtained using the low leakage loading patterns (low power on the core periphery and high power in core centre), thus loading patterns with maximal allowed power peaking according to the technical specifications. Further improvement of the fuel economy can be thus reached either through the increase of the power peaking limits, which means margin reduction in safety analysis, or alternatively through the margin reduction during the reload safety assessment and core monitoring. Both margins are directly dependent on the precision of core calculations: the precision of the core calculations directly influences the economy of plant operation. Analysis show that the increase of 1% of power peaking limits can lead approximately to cycle length increase of 5 days without any change of core inventory.

Predictions of criticality: Prediction of criticality is closely related to the prediction of cycle length. Today prediction of cycle length within 10 FPD is considered as acceptable, which corresponds approximately to the cost of 1 fuel assembly.

Modelling of fuel behaviour and thermal-hydraulics directly influences the limits of power peaking factors.

Correct prediction of used fuel isotopic composition directly influences the costs of the back-end of the fuel cycle: the higher uncertainty means more conservatism of the criticality analysis, and thus more expensive facilities (need of more casks, thicker shielding etc.)

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These are the reasons why significant efforts worldwide are directed towards the improvement of precision of core calculations, which is a non-trivial task because further improvement cannot be achieved through better modelling of single phenomena, but improvement of several phenomena at once would be necessary (for example error of numerical methods used in current 3D core neutronics solvers is of the same order of magnitude as error caused by methods used for data homogenization: both should be improved in order to improve the precision of core design calculations).

R&D topics

- ✓ **Developing computer code chains for reference core calculations (accurately avoiding limitations on running time through e.g. massive adoption of parallelisation and or assumptions on symmetry of system layout). Those chains should be suitable for evaluation of long-time kinetics of core reloads. The results of such calculations should contribute to the enrichment of data-base for qualification of computer codes adopted for design and safety assessment,**
- ✓ **Developing numerical methods for the multi-physics space–time coupling among neutronics, thermal-hydraulics and thermo-mechanics codes as well as for multi-level coupling between homogenized coarse-mesh lower-order and heterogeneous fine-mesh higher-order solvers in order to establish efficient computational schemes for full-core pin-wise predictions of operational and safety criteria such as DNB, PCI, PCMI (subject of NURESIM and CASL/MOOS projects),**
- ✓ **Implementing suitable methods for cross section libraries preparation accounting for vicinity effect in cell calculation (overcoming the “infinite array of assemblies” approximation in 2D of current lattice codes),**
- ✓ **Developing advanced thermal-hydraulic and computational fluid dynamics solvers for predictions of 3-D two-phase multi-field flow and heat transfer phenomena at the level of individual fuel rods, taking into account global effects such as cross-flows or influence and interactions with assembly/core structural components,**
- ✓ **Implementing mechanistic models for system and sub-channel codes or advanced computational fluid dynamics based methods with the aim to replace correlations for predictions of departure from nucleate boiling or dry-out phenomena,**
- ✓ **Collecting operational data from the NPPs and/or consolidate the availability of experimental test facilities for validation of modern advanced core calculation methods.**

4.2.4.5 Improvement of core monitoring and instrumentation

The margin taken into account during the core design process consists of two components: the first is related to limited precision of core design software, and the second to the core monitoring: the error of the “measured” power distribution is generally smaller than uncertainty of the core design software, but the measured and predicted power distributions can slightly differ and the power peaking of the measured distribution can be higher compared to the predicted one.

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Precision of "measured" power distribution is limited by detector precision, precision of interpretation of measured data, and precision of the core calculation data entering 3D power reconstruction algorithm. Improvement of detector precision, capabilities of detector signal interpretation, point-wise power and power-peak reconstruction, uncertainty evaluation of reconstructed power distribution including the improved methods for factoring the impact of the fuel assembly and rod bow and twist, as well.

R&D topics

- ✓ **to explore, develop and define new strategies and approaches to core monitoring (optimization of in-core and ex-core instrumentation, direct monitoring of DNBR, missloading detection and prevention),**
- ✓ **to develop and implement novel types of detector systems allowing simpler and more reliable interpretation of signals (i.e. with low dependence to fuel burn-up); enlarging the range of measured parameters and to reducing the error of the measurements, e.g.; avoiding use of calculations to access the targeted parameters.**
- ✓ **to estimate and reduce the measurement and power reconstruction uncertainties (as a function of fraction and type of available detectors, their distribution within the core, detector burn-up, …),**
- ✓ **to develop advanced techniques (method and tools) for the calculation of the sensitivity coefficients of the integral quantities measured with the detectors with respect to parameters representative of the hot spot, and taking into account the errors associated with the measurements,**
- ✓ **to assess the effect on the quality of the hot spot detection system following possible failures of the measuring devices during the core life cycle and to define an adequate protection strategy in terms of quality, number and distribution of the detectors,**
- ✓ **to apply the diagnostics methods to measured signals in order to diagnose in-vessel anomalies, such as mechanical vibrations of structural components or fuel assemblies or process disturbances propagating to the core and affecting its dynamical properties or vice versa,**
- ✓ **to improve interpretation of instrumentation signals (i.e. readings of thermocouples, which can be biased due to incomplete coolant mixing or flow irregularities, readings of in-core neutron flux detector signals, the ex-core signals, …),**
- ✓ **to link the core reconfiguration (fuel assembly bow, increase of water gaps among the fuel assemblies) to the measured data and to improve methods for factoring the impact of the non-design fuel system conditions – e.g. geometric distortions of fuel assemblies and rods (interlink with the Challenge 3-3 for the impact on uncertainties).**

4.2.4.6 Improvement of robustness and precision of uncertainty estimations

Improvement in estimation of uncertainties has the same effect on the economy of the core management as the improvement of core calculations. Similarly to the improvement of precision of

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the core calculations, the improvement of uncertainty estimations is a highly non-trivial problem. The principal challenges are:

- development of methods for determination of system similarity, in order to clearly specify the range of applicability of uncertainty determined via comparison to experimental data,
- development of methods for estimation of uncertainties related to abnormal core operation,
- estimation of uncertainties arising from physical models,
- development of methods for propagation of uncertainties through the chain of computational tools,
- enhancement and extension of methods for propagation of uncertainties in burn-up calculations.

R&D topics

- ✓ **developing advanced techniques (method and tools) for the calculation of the sensitivity coefficients of activities measured with the detectors vs. parameters representative of the hot spot, and taking into account the errors associated with the measurements, including impact of the non-design fuel system conditions – e.g. geometrical distortions of fuel assemblies and rods - into the applied measurement uncertainties (interlink with Challenge 3-2),**
- ✓ **implementing efficient and optimized sensitivity/analysis (S/U) methods for the propagation and quantification of nuclear data related uncertainties in reactor analyses,**
- ✓ **enhancing statistical sampling methods for producing sensitivity information and enhance perturbation-theory methods for providing uncertainty quantifications for a large number of output responses,**
- ✓ **applying S/U methods in order to identify and screen relevant nuclear data uncertainties (e.g. nuclide, reactions, energy groups) for target applications (e.g. core reactivity, local pin powers, kinetic parameters, reactivity coefficients, isotopic compositions) and guide thereby, refinements and enhancements of variance/covariance data,**
- ✓ **developing methods to assign uncertainties in physical models of thermal-hydraulic codes, along with a consolidated validation of thermal-hydraulic codes against experimental data, and to propagate these in safety analyses and multi-physics simulations.**

4.2.4.7 Multicycle core reload optimization, combined optimization of loading pattern and fresh fuel profiling and improvement of reliability of the optimization process

The core-loading pattern is crucial for both fuel cycle economics and core operation safety: it influences the discharged fuel burn-up, power peaking, temperature feedback parameters and fast neutron fluence on reactor vessel. Several fuel assemblies for fuel cycle can be saved thanks to good core optimization.

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The core reload optimization process could be significantly enhanced by solution of following beyond-state-of-the-art topics:

- enabling simultaneous optimization of the loading pattern and the fresh fuel inventory (number of fresh fuel assemblies, their enrichment and profiling),
- allowing multi-cycle optimization accounting for fresh fuel enrichment and discharged fuel burn-up constraints: finding the optimal core reload pattern for the next cycle does not assure long term optimum for the fuel economy,
- improving the reliability of the optimization process, so reduce gaps between the optimized and the design load patterns.

Also, all nuclear power plants will eventually approach end-of-life and be permanently shut-down. A critical issue will therefore be to address the core design of the last operating cycles of a nuclear power plant approaching shutdown. The design of such cores, referred to as "End-of-Life" cores will imply several challenges such as fuel procurement and utilization to avoid stockpiles of low burned highly reactive spent fuel, evaluation of modified reactor operation strategies to maintain safety and economic targets, assessment of impact on waste management and spent fuel facilities. The objective function of core optimization process shall be redefined for this purpose.

R&D topics

- ✓ **comprehensively addressing the unsolved problem of multi-cycle optimization,**
- ✓ **enabling simultaneous optimization of loading pattern and fresh fuel inventory,**
- ✓ **improving the robustness of core optimization process by improving the agreement of optimization solver and reload safety assessment software,**
- ✓ **developing and assessing strategies along with computational methodologies for the optimization of end-of-life cores.**

4.2.5 Water chemistry and LLW management (STA 3.5)

4.2.5.1 Scope

Water chemistry and low level waste processing at nuclear power plants are one of the key technologies to establish safe and reliable operation of nuclear power plants. Water chemistry conditions can impact corrosion rates, fuel performance, and radiation management. Low level waste management can impact source-term reduction, effective radiation protection and economical aspects.

Water chemistry affects all materials in contacts with cooling water and at the same time it is affected by the materials themselves. Improved water chemistry can reduce the fre¬quency of transient fault conditions and overall impurity concentrations. Continued improvements are needed to optimize water chemistry and balance the resulting impacts on system materials corrosion, fuel performance, and radiation fields. Greater effort is also needed for the safe processing, handling and disposal of low level waste with respect to regulatory, economic and environmental requirements to

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optimization technologies for waste disposal volume minimization and reduction, dose and radiation field reduction and nuclear plant decommissioning.

In the following sections, several potential research topics are identified, both at general level and specifically for the different plant systems (primary, secondary and cooling/auxiliary systems). Many of the topics defined have potential interactions with research items in the field of structural integrity. For this reason, attention is given here mainly to the activities related to water chemistry and less to water interaction with structural materials.

The continuously increasing pressure used to reduce release of radioactive and other hazardous substances into the environment requires constant improvement of processes and technologies for treatment and conditioning of liquid radioactive waste. Severe measures to minimize generation of radioactive waste in nuclear power plants are introduced because of the rising costs and requirements for treatment, storage and disposal of this waste.

LLW management as a part of the TA3 is presumably focused on liquid waste while the TA5 part is dealing with mainly non-water based radioactive waste (organic, solid, etc.).

4.2.5.2 State of the art

Continuous and collaborative efforts of plant manufacturers and plant operator utilities have been focused on optimal:

- Water chemistry control, for which, a trio of requirements for water chemistry should be simultaneously satisfied - better reliability of reactor structures (e.g. fuel cladding, pressure boundary structures, core internals), lower occupational exposure of workers due to accumulated corrosion products and fewer radwaste sources,
- Low level waste management to minimize and reduce the generation of low level waste, improve technologies and tools, safe and efficient on-site storage of low-level waste.

The importance of the role of the chemistry and low level waste management in the nuclear power plants has been recognized many years ago, and research programs have been carried on in several laboratories. The largest basic and applied research program in this field has likely been carried out in USA by EPRI; started several years ago, it continues nowadays with the following objectives:

- development of cost-effective optimization tools and techniques to improve plant availability and safety,
- demonstrate new technologies through first-of-a-kind applications,
- development of dedicated software tools to improve control, diagnostic capabilities, and staff productivity,
- enhancement of technology transfer through plant-specific collaborations,
- on-site assessment support to benchmark plant controls and identify opportunities to optimize protocols,
- minimizing of the generation of low level waste
- safe and effective on-site storage of low level waste

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- development of tools for improved flexibility and risk-informed regulation for low level waste

The EPRI Water Chemistry Program also develops and updates water chemistry and low level guidelines for nuclear reactors based on industry research and plant experience, as well as optimization tools to mitigate corrosion, achieve, and maintain design fuel performance standards, minimize plant radiation fields and reduce and minimalize low level waste

As a result of this coordinated effort, water chemistry and low level waste management in the nuclear power plants can be applied to mitigate the corrosive environments inherent in plant operation. Chemistry and low level waste management systems have been developed to support and assess the correct implementation in the nuclear power plants. The systems can provide effective and sophisticate functions such as data acquisition, evaluation, analysis, calculation, trending reporting, and so on. The systems can also contribute to confirmation of reliability during nuclear power plant operation. Immediate investigation and countermeasures against anomalies are required. In addition, high similarity and accuracy are desired in the methods of investigation and taking countermeasures.

4.2.5.3 Challenges and research topics

4.2.5.4 Barriers protection (incl. crevice corrosion)

An effective barriers protection is essential for maintaining the reliability of plant systems and components, reducing system corrosion (to extend the life of the plant and reduce dose rates) and limiting the amount of radioactive material and harmful chemicals released to the environment.

There is potential research issue aimed at optimizing of water chemistry control methods, improving of monitoring techniques, and chemical additives to control corrosion, reduce radiation fields, and maintain fuel performance. Guidelines for improving related topics should be developed too.

R&D topics

- ✓ **to experimentally develop a method for dosing of precious metals and zinc in primary water circuit,**
- ✓ **to develop new and thrifty decontamination methods of primary components,**
- ✓ **to find new methods and techniques for reduction of radiation fields and collective effective doses,**
- ✓ **to develop a reliable method of sampling from crevices,**
- ✓ **to find a reliable analytical method for measuring of impurities in crevices in primary and secondary circuit materials.**

4.2.5.5 Biological and inorganic deposits reduction

The development of corrosion inhibitors and focus on human dangerous microorganisms in presence of organic and inorganic compounds is another topic for research as well as monitoring of biological and inorganic (CaCO3) deposits. Microbiological population could grow in biofilms, which cause considerable changes in oxygen concentration and pH. Biofilms could have also impact in worsening

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of heat transfer and consequently lowering of cooling effectiveness. Each nuclear plant or unit has different structure of biological activation and so has different chemistry for its clean-up, as well.

R&D topics

- ✓ **to develop new corrosion inhibitors,**
- ✓ **to monitor biological and inorganic deposits in order to improve heat transfer and cooling effectiveness,**
- ✓ **to develop the technical basis for fouling caused by organic growth or silt deposits**

4.2.5.6 Environmental and radiological protection by using new technologies

Many membrane technologies are already on the market and widely used in industrial installations. They offer a huge potential for an application in NPPs. The new technologies as membrane processes could be the crucial one for reducing of low level waste (liquid) volume and significantly cost of operation. There is potential research issue aimed at reducing liquid waste volume e.g. radioactive laundry and technology waste.

Modern membrane technologies can be more helpful for further cost-related technology – decontamination and clean-up of boric solutions for recycling.

There is a need to develop guidelines based on these principles, shared by regulators, to allow the continual treatment process.

R&D topics

- ✓ **to develop cost-effective polishing technology with membrane technologies,**
- ✓ **to develop new cost-effective methods for clean-up of boric acid and decontamination**
- ✓ **to increase effectiveness of purification systems,**
- ✓ **to develop guidelines based on the new membrane technology.**

4.2.6 Radiation protection (STA 3.6)

4.2.6.1 Scope

Radiation protection is an area dedicated to the protection of humans and the environment against harmful impact and/or consequences of ionization radiation. This is achieved first through improving the scientific understanding on how ionizing radiation interacts with living matter (humans and non-human biota) in representative conditions of exposure and how such interactions promote effects on health (for man) and ecosystems (for the environment). On this basis next, regulations, guidelines, methods and tools are constructed in view of reducing/eliminating the risks to personnel, population and environment exposure, as arising from natural and man-made (i.e. medical and industrial) sources of radiation. One of the SNETP Technology Working Groups on Gen II and III identified reducing the radiological dose to the plant employees and workers as one of the highest priority objectives. A parallel objective is reducing the radiological impact on both the local population and the environment.

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In the framework of the NUGENIA Area 3 “Improved Reactor Operation”, the scope is tightly related to the everyday plant management activity. Emergy preparadness and response issues are addressed in the TA2-5.

The resulting elements of RP, as expressed in operational terms, consist in national regulatory infrastructure, radiological protection in occupational exposure, radiological protection in medical exposure, public and environmental radiological protection. Each of these elements is dedicated to individual research, development and/or engineering issues, but they are all driven with concerns on good practice based on ALARA principles, i.e. to limit exposure "as low as reasonably achievable". This principle is quite clear in principle, but the semantics "as low" and "as reasonably achievable" drives to uncertainties and difficulties of appreciation due to the fact that they impact on budgetary issues and available resources.

4.2.6.2 State of the art

Radiation protection has always been and continues to be one important line of activity in the EURATOM research programs. Strong recent emphasis has been dedicated recently in Europe on radiation protection leading to the establishment of the Multidisciplinary European Low Dose Initiative (MELODI) platform, with the overall aim of integrating European initiatives on low dose research, including epidemiology. Scientifically, this is aimed at resolving the sustained controversy about the existence/non existence of radiation biological and subsequent health effects at low dose (and/or dose rate) which may lead to reorient the current system of radiation protection. In practice, this new organization is meant to improve efficiency by means of mutualisation of efforts and infrastructures and also by opening better access to critical mass of research teams on key scientific issues. This platform, oriented on basic research to better understanding the health effects of man’s exposure to low dose radiation, issued in 2011 a Strategic Research Agenda which defines a series of topics suitable to be considered in the long term research program.

In the recent years, two other platforms contributing to radioprotection issues have evolved. The first one, the Radioecology Alliance, addresses environmental issues related to protection of the public and ecosystems from radiation (Radioecology Alliance platform, supported by the STAR and COMET EU projects) and has issued in 2013 a Strategic Research Agenda wich includes both basic and applied research. The second one, the NERIS platform, addresses issues related to emergency preparedness and response in case of accidents (European platform on preparedness for nuclear and radiological emergency response and recovery.

Most recently, in 2013, the effort of European research structuring initiated with the MELODI platform dealing with low dose has been widened to integrate further the Radioecology Alliance platform, dealing with environment protection, and the NERIS platform, dealing with emergency preparedness, within a wide radiation protection integrated association (called OPERRA).

This wide emerging association of platforms gives high priority to basic research aimed at overcoming the stumbling blocks still puzzling the current efficiency of radiation protection which are clearly not within the scope of NUGENIA. However, the more operational research goals as linked to the everyday plant management activity, as expressed in the NUGENIA roadmap fully meet the NUGENIA Area 3 “Improved Reactor Operation” without promoting redundancies.

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The radiation protection rules as developed by ICRP are based on a renewed system framework which proposes a design and application of limits suitable for both personnel and population, and most recently non-human biota protection as well.

In short, the main evolution from ICRP 60 to ICRP 103 is that there is no more distinction between practices and interventions. The two concepts are replaced by three generic exposure situations, which cover all conceivable exposure situations: planned exposure situations (identical to practices), emergency exposure situations and existing exposure situations.

New ICRP recommendations were published in 2009 that are currently being implemented into national legislations[14].

Other activities carried out worldwide (e.g. OECD - Halden Reactor Project) are focused on virtualization of real world with added information, so called Augmented Reality. These methods and tools serves as a contribution for optimization of maintenance and repairs of equipment in contaminated areas (e.g. disassembly during decontamination), for training of the personnel in view of reducing and/or eliminating human failures and finally for optimization of interventions and radiation monitoring activities during emergencies. Visualization of the radiation field and evaluation of the received doses for personnel form a good tool to understand risks and to optimize the human work under these conditions (e.g. for decommissioning planning).

4.2.6.3 Challenges and research topics

One major challenge for radiation protection as tackled in NUGENIA TA 3 "Improved Reactor Operation" is to promote a better synergy with nuclear safety issues since both are interdependent. Technological solutions taken for safety reasons are not all equal with respect to radiation protection.

Besides improving this synergy, Radiation Protection issues tackled within NUGENIA are clearly legitimate when they do not duplicate research lines already addressed through the existing radiation protection platforms, but complement them through bringing additional methodologies, guidance, coordination, developments that are made accessible by virtue of the NUGENIA membership where operators are largely represented (in contrast to radiation protection platforms and associations) . This means essentially an orientation of Research topics towards applied and/or technological research meant to feed operational and practical goals.

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[14]According to the characteristics of the exposure situation, including the degree of controllability of the radiation sources, the ICRP recommends a dose scale (corresponding de facto to a risk scale) with three bands: 0 to 1 mSv/yr, 1 to 20 mSv/yr and 20 to 100 mSv/yr, in order to select dose constraints and reference levels.
For the protection of the public in case of a nuclear accident the ICRP recommends to select reference levels: in the 20–100 mSv/yr band for emergency exposure situations, in the lower part of the 1–20 mSv/yr for existing exposure situations, with the objective of reducing exposure below 1 mSv/yr in the long term and values of reference levels and timeframe will vary from place to place depending on the local circumstances.
One key issue is the transition from emergency exposure situation to existing exposure situation. ICRP Publications 109 (ICRP-109, 2009) and 111(ICRP-111, 2009) propose a flexible framework for guiding actions in case of a nuclear accident or a radiological emergency. The key guidance is to avoid doses above 100 mSv, reduce exposure (ALARA) all the time, engage affected people in the management of the situation, develop radiation protection culture among the affected people and adopt 1 mSv/year as a long term objective. Even if the ICRP-103 has been issued in 2007, these principles have not been implemented into national regulations yet and praxis still keep in track ICRP-60 and ICRP-63 principles. Research activities have been started in selected organizations belonging to the NERIS platform members, like in Germany, UK, Belgium and Finland.

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We can summarize challenges in the field of radiation protection can be summarized as follows:

- to prepare effective methods and tools to keep the exposure of personnel and/or population to radiation and other co-stressors as low as possible, with demonstrated minimum detrimental impact on life during normal operation,
- to prepare effective methods, tools and regulations to keep the radioactive track into the environment as low as possible, with demonstrated minimum detrimental impact on life, e.g. to limit radioactive waste during operation of radioactive sources like NPPs,
- to avoid and or to drastically minimize the risk of human failures in the process of designing, operating and maintaining different kinds of radiation sources, including securing all these sources against thefts and misuse by criminals and/or terrorists and/or unstable governments,
- to improve and periodically evaluate, review and inspect the technical, organizational and safety-cultural issues of radiation sources,

These challenges help to understand the research needs, but the extent is very large and it is obvious, that the research activities should be specified in detail and prioritized from the stakeholders and budgetary points of view.

To achieve the first objective (to keep the doses to workers and to population in accordance with ALARA principle), new technologies and tools may be fed with information related to radiation protection (virtual reality as-build model with data about radioactivity, real-time data about dose automatically sent to a "radiation protection control room", etc.). Reducing doses received by personnel is clearly a target for all utilities with a benefit for safety since it reduces the stress induced by the fact of working in a nuclear power plant.

To achieve the second objective (reducing the radiological impact on the environment), it is advisable that improved tools are developed both for the plant management and for the analysis of consequences of the release of radionuclides, e. g. on the underground water. Updated systems for tritium, carbon C14 and boron management could be developed.

The above mentioned challenges induce R&D topics. There are some common issues like eliminating impacts and consequences of human failures, early identification of initial events indicators, evaluating existing approaches and processes and finally erecting good practices of understanding and compliance with safety culture, including educational, training and drilling issues. Beside of the common issues, we can define following five specific challenges in the field of radiation protection.

4.2.6.4 Radiological protection and occupational exposures

There are potential research issues aim at developing organizational and management tools for real time monitoring of received doses of personnel, their processing, archiving, optimization and visualization. Virtual reality and/or augmented reality tools should be good basis for personnel training, for optimization of the human activities in radiation affected areas and for real time visualization of doses and their values compared to limits. Such tools could deals also with the radiological issues during accidents when interventions, radiation monitoring and countermeasures for population protection are implemented. Mobile communication between center (e.g. radiation

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protection control room) and field workers should be used to erect real time teamwork enabling to reduce and/or minimize the risk of human failure.

Guidelines for improving topic related safety culture should be developed too, to keep the safety track of all human activities in radiation sources technologies.

R&D topics:

- ✓ **to develop organizational and management tools for real time monitoring of received doses of personnel, their processing, archiving, optimization and visualization,**
- ✓ **to develop virtual and/or augmented reality tools for personnel training, for optimization of the human activities in radiation affected areas and for real time visualization of doses and their values compared to limits,**
- ✓ **to develop the means of mobile communication between center (e.g. radiation protection control room) and field workers should be used to erect real time teamwork enabling to reduce and/or minimize the risk of human failure.**

4.2.6.5 Public and environmental radiological protection

The amounts of radioactive substances periodically released into the environment under normal operational conditions should be reduced by researching, development and/or engineering of advanced methods and principles on how to process radioactive waste and aerial and liquid radioactive substances, reduce their volume and store the remaining parts of them after processing.

A specific topic is to develop methods and tools able to reliably assess radionuclides releases (e.g. tritium, 14C, etc…) and to reduce their amount under normal operational conditions, including research on chemical operational conditions which would help to reduce standard operational releases.

In the recent decades, controversial results have been continuously discussed about the health effects of protracted dose of radiation due to the release of radionuclides form NPPs in normal operation. This is also the case with studies related to the health effects form the Chernobyl contamination spread in Ukraine, Belarus and Russia, even sometimes beyond, with claimed results which have difficulties to reach clear statistical significance. Also, the quantification of risks associated to low doses protracted over their whole professional life by workers employed in the nuclear industry is still uncertain.

This difficulty to evolve with conclusive evidence is arising from two major reasons: first, the limited statistical power of most epidemiological studies which have often been conducted on a national basis with cohorts suffering from being too small, and second, the uncertainties associated to dose estimates, which are generally due to missing exposure data, measurement errors and assumptions in the assessment process. Attempts have been devoted in the recent years to build larger international cohorts in order to improve the situation, but this is made complicated by the absence of clear and harmonized guidance with respect to suitable methodologies of data acquisition and overall organization. Regarding uncertainties, progresses have been made over time in the reconstruction of exposures, but the use of error correction approaches remains very rare in the field. The R&D topics to be addressed do not cover epidemiological studies per se (to be conducted

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elsewhere under the radiation protection platforms) but rather the european-wide harmonization of the methods necessary to support them in view of promoting larger cohorts with better statistical power and better control of exposure uncertainties.

R&D topics:

- ✓ **to develop methods and tools for minimizing the release of radioactive substances into environment under normal operational conditions from NPPs and from other sources or radiation,**
- ✓ **to develop methods, guidelines and tools necessary for reliably assessing radionuclides released into the environment with due consideration of their potential impact on human health and on the environment.**
- ✓ **Identification of scenarios leading to potential radiological impacts on health of workers and the public which may benefit from the construction of large European-wide cohorts with high statistical power,**
- ✓ **Development of a generic an harmonized process for data base construction throughout Europe in support to more robust and conclusive epidemiological studies,**
- ✓ **Development of an adequate European methodological guidance.**

4.3 References

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[TA3-6] EPRI: PWR Primary Water Chemistry Guidelines

[TA3-7] EPRI: BWR Water Chemistry Guidelines

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4.4 Annex

The following priorities were devised in brainstorming by subarea leaders. Such expert judgement was used to define priorities in a narrative way instead of mathematical scoring of individual topics due to their incomparability in different sub-areas.

3.1 Operational economics and NPP flexibility	• Integration of economy and safety issues of NPPs operation in market conditions (incl. NPP flexibility)
3.2 Human and organizational factors	• Development of new framework and guidelines for the design and safety demonstration of human-automation interplay in the control of plant operations. This target could be reached as a result of collaborative effort of experts from HOF, I&C and reliability and risk engineering.
3.3 Advanced digital technologies	• Development of FPGA assessment approach for modernization strategy of I&C systems • Development of cybersecurity policy for NPPs at European level complemented with recommendations for implementation.
3.4 Core management	• Development of approaches for creation of additional operational margins for core design and management (mainly code development)
3.5 Water chemistry and LLW management	• Application of membrane technologies for reduction of liquid waste volume • Improvement of water chemistry to decrease corrosion rates.

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3.6 Radiation protection	• Reduction of occupational exposures • Improvement of plant operation to reduce radiological impacts on the environment

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5 TECHNICAL AREA 4 – Integrity Assessment of Systems, Structure and Components (TA4)

Technical Area Leader: Elisabeth Keim (Areva-GmbH)

5.1 Executive Summary

5.1.1 Scope

This section has been compiled from information gained from specialists of utilities, vendors and research organisations working under the NUGENIA Technical Area 4 "Integrity Assessment of Systems, Structures and Components (SSCs)".

While the assessment principles relating to SSCs are generally comparable in Europe, the actual methodologies and codes are different in the various European countries. With the longer term objective of European harmonization in mind, it is necessary that the differences are fully understood and for the lessons learned from Gen II nuclear power plants (NPPs) to be taken into account when developing and/or revising best practice guidance for the safe operation of SSCs with satisfactory, but not over-conservative safety margins. This is required in order to ensure high integrity and high performance in the case of internal and external loads and post-Fukushima lessons imply that investigations of beyond design loads must be considered.

The SSCs that need to be considered are those that are important for safety and availability, those that require high costs to replace and those that cannot be replaced without a significant long term refurbishment programme (including components like Instrumentation and Cables -I&C).

Structural assessments are an important part of management programs (e.g. ageing management programs, maintenance and design changes). These assessments are required to enable and improve the acceptability of periodic safety reviews. Aspects that need to be considered include definition of integrity assessment over the whole life cycle, the various degradation mechanisms, ageing issues, safety margins and harmonization issues.

This technical area roadmap comprises integrity assessment, description of loads, materials performance and ageing, ageing monitoring, prevention and mitigation, functionality and qualification. These items are largely addressed with respect to metallic components, civil works (concrete structures), polymers and I&C.

For TA4 short-term, mid-term and long-term projects have been defined as follows:

Short-term projects have the mission to identify commonalities between national programs, utility programs and bi-/multilateral partner programs. The main focus is operational-related research, mainly using own resources forming in-kind contributions in terms of NUGENIA projects. European validation / knowledge exchange will usually come under the short-term category. It may also be noted that short-term projects may facilitate further projects and build the basis for mid- and long-term projects - Stages 4 and 3, typically 0-2 years.

Mid-term projects are based on the identified gaps of short-term projects. Additionally the rationalization of results by systematically structuring experiments e.g. additional experiments, up-scaling etc. is one mission of these projects which means validation of models. Mid-term projects could also be based on specific or operational related issues; they may be required in order to satisfactorily address new regulatory demands which are not currently addressed. Additionally, such projects may be required in order to support the harmonization process by proposing and improving guidelines. It is envisaged that EC contribution is needed for mid-term projects. - Stages 2 and 1 (More urgent to be picked up to short-term), typically 2-5 years

Long-term projects can be initiated based on the knowledge of short-term and mid-term projects. More precisely, this includes in depth investigation of issues. Additional aspects may include creating and validating new methods, either experimentally of by modelling and pre-normative research. It is envisaged that EC contribution is required for long-term projects - Stage 0 (More urgent to be picked up to mid-term), typically up to 10 years

5.1.2 Objectives

The objective of NUGENIA TA4 is to improve knowledge and methods for integrity assessment of SCCs in order to increase their safety and availability and to improve control of their lifetime.

The research roadmap is an essential part of an industrial initiative and it is seen that the identification of the research needs is also an essential part of NUGENIA.

5.1.3 State of the art

5.1.3.1 Integrity Assessment

Integrity assessment is concerned with establishing the current and/or future state of SSCs by way of taking into account material properties, component geometry, loading and degradation mechanisms. Fracture mechanics methods are an essential part of integrity assessment and they are used for example in the reactor pressure vessel (RPV) integrity assessment under pressurized Thermal Shock (PTS) loading and in leak before break (LBB) analysis in piping. Codes, standards and procedures are commonly used for integrity assessment. These are generally well founded and validated but in many cases they can have inherent levels of conservatism that can be restrictive in particular circumstances, particularly when considering plant life extension where degradation mechanisms need to be considered and taken into account. There is thus a need to properly understand the levels of conservatism with a view to revising the guidance and procedures. Aspects such as effects of load history, crack arrest, treatment of thermal and weld residual stresses and warm pre-stressing effects are important for this aim.

5.1.3.2 Description of Loads

Having a relatively accurate knowledge of applied loads and resulting stresses (and strains in some cases) in SCCs is inherently an important aspect. Increased computing power over recent years, coupled with advanced numerical modelling capabilities, has resulted in many types of loading and stresses being able to be evaluated with a good degree of accuracy. Examples of this include piping system loads and stresses, stresses resulting from pressurized thermal shock loading and residual stresses resulting from fabrication processes such as welding. Nevertheless, there are still issues concerning loads and stresses that need to be addressed and resolved from a R&D perspective.

5.1.3.3 Materials Performance and Ageing

Properly understanding the performance of materials relevant to structural components and the effect that ageing mechanisms can have on such materials is another important aspect. In

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considering materials performance and ageing, it is useful to consider in turn the aspects of material properties, ageing and degradation mechanisms and modelling of ageing.

5.1.3.3.1 Material Properties

Knowledge of many material properties is required in relation to SSCs. Examples of material properties required for “start of life” assessments include tensile properties, fatigue stress versus number of cycles (S-N) data and fracture toughness data for defect tolerance evaluations. In addition, and particularly for in-service evaluation of SSCs, properties relating to aspects like irradiation creep (for high temperature operation), corrosion and environmental fatigue may be required (also see below). Data are fairly widely available on all these materials properties for materials of interest but there remains some concern and debate regarding the treatment of scatter, in corrosion and environmental-fatigue data, to correctly apply in integrity assessments. In addition, there is the complex issue of how manufacturing processes (including welding, thermal and mechanical treatments and coatings) may affect materials and materials' properties and how such processes may be improved in the future.

5.1.3.3.2 Ageing and Degradation Mechanisms

In addition to the good knowledge of “start of life” properties referred to above, it is vital that all ageing and degradation mechanisms that potentially could be active during the required lifetime of SSCs are properly understood and adequately taken into account in integrity assessments. These include: fatigue, irradiation embrittlement, stress corrosion cracking, irradiation assisted stress corrosion cracking, thermal ageing, general corrosion, erosion-corrosion, strain aging, environmental fatigue, creep, creep-fatigue and thermal fatigue. Several of these mechanisms (e.g. fatigue crack growth, creep) necessitate material properties in their own right to represent the actual mechanism. For other mechanisms (e.g. irradiation embrittlement, thermal ageing), it is important to be able to properly understand their positive or negative effect on “start of life” properties like tensile yield strength, impact energy and fracture toughness.

The effects of the ageing and degradation mechanisms need to be considered for the specific type of SSC material being assessed. For example, all the above mechanisms can potentially have a significant effect on steel components and it is known that for polymers, thermal and irradiation embrittlement is of particular concern. Guidance and/or data are generally available in order to account for most of the mechanisms referred to above. However, in some cases (e.g. environmental fatigue), such guidance and data may incorporate unrealistic factors in order to ensure conservatism. There is therefore clearly a need for further R&D to be undertaken in several of the areas referred to above, in order for a better fundamental understanding of the ageing and degradation mechanism to be achieved.

5.1.3.3.3 Modelling of Aging

The proper understanding of aging mechanisms requires a research strategy based on combined experimental and theoretical studies following a multidisciplinary approach, utilizing state of the art experimental and modelling techniques. Material characterization at different length scales (i.e. nano, micro, meso, and macro scales) is necessary, focusing on the physical understanding of the degradation processes (e.g. neutron irradiation, thermal aging, corrosion, etc..., see above) and their influence on macroscopic mechanical properties and structural/functional integrity of the components. In order to extract/recognize the relevance of the most important degradation mechanisms, to facilitate scale-linking methodologies and to validate the models, a variety of

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materials with different structural complexity (e.g. from model alloys to steels in the case of RPV materials) must be investigated. The development and application of both empirical and physics-based models should be pursued in parallel, with the accent towards their refinements which should bring them closer together and eventual merge. Physics-based models include techniques (especially at the atomic level) that allow mechanisms not easily deduced from experiments, or quantities that are difficult or impossible to extract experimentally, to be identified and calculated. To this regard, one important long term aim is to develop fully validated multi-scale models that link the nano-scale to the macro (i.e. structural) scale, using simulation tools such as molecular dynamics, kinetic Monte Carlo, dislocation dynamics and crystal plasticity theories, with the support of ab initio calculations and ever more refined interatomic potentials.

5.1.3.4 Ageing Monitoring, Prevention and Mitigation

5.1.3.4.1 Ageing monitoring

Ageing monitoring is a way to correlate the evolution of microstructure and material damage with applied loadings and environmental conditions. This will be particularly useful in the case of infrequent transients and will enable the operator to verify the suitability of maintenance programs and in-service inspection, thus ensuring that operation remains within allowable limits. The monitoring of ageing is largely in its infancy and much work is needed to satisfactorily meet the overall needs.

5.1.3.4.2 Prevention and mitigation

Prevention and mitigation of both ageing mechanisms themselves and their resulting damage and failure has been a long term issue for engineers and scientists in many industries, including nuclear. Mitigation measures and several processes (e.g. peening and heat treatments) have successfully been developed and utilized on plant components to increase and maintain the plant safety. However, further studies and developments are required, specifically with the longer lifetime of the plants.

5.1.3.5 Functionality

Under this topic, the issues of equipment reliability, industrial obsolescence and maintenance are considered.

5.1.3.5.1 Equipment Reliability

Equipment Reliability is associated with ensuring that all the systems are able to perform their intended function in a reliable and safe manner throughout the lifetime of their required use.

5.1.3.5.2 Industrial Obsolescence

Industrial Obsolescence is at the plant design level when further life extension/optimization is unable to be justified. It essentially refers to components that are no longer manufactured so that replacement can go as far as a complete change of the system or a real change in design. An example is I&C systems, with the forced transition from analogue to digital technology.

5.1.3.5.3 Maintenance

Maintenance is associated with ensuring that all the systems are maintained in a satisfactory way that will ensure equipment reliability throughout the lifetime of required use.

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5.1.3.6 Qualification of structural integrity assessment and lifetime estimation procedures for SSCs

Qualification in the context of structural integrity assessment and lifetime estimation of SCCs is the validation and verification of corresponding procedures. This can be accomplished analytically by using theoretical models and/or experimentally by using tests. Analytical qualification also includes comparison of different codes & standards for specific issues. All activities that contribute to the validation and verification of structural integrity assessment and lifetime estimation procedures for SSCs are part of this sub-area. Also the development & revision of best practice guidance documents, up to the contribution to the development & revision of codes and standards is also part of this sub-area, as well as pre-normative research related to the structural integrity assessment of SSCs, in particular the development & standardisation of material and component tests.

5.1.4 Challenges

5.1.4.1 Integrity Assessment

The underlying principles are to develop and validate modelling tools which can be used to obtain a better fundamental understanding of material and structural behaviour, including ageing mechanisms and the effect of flaws.

The objectives of the modelling for integrity assessment are to translate the mechanistic understanding to simulation tools and assessment procedures to predict theoretical margins for the safe operation of NPPs, taking into account structural features, real or postulated flaws, loading conditions and relevant material characteristics including ageing effects, by concomitantly focusing on:

- Different material types (metallic, concrete and non-metallic)
- Different failure modes (brittle fracture, ductile fracture, fatigue, environmental effects, high temperature effects, buckling) depending on materials and loading conditions for the type of reactor system considered
- Developing simulation platforms based on complex phenomena, but which can be reliably used.
- The challenges related to long term operation of different structural components.

5.1.4.1.1 Short term challenge lessons learned from GenII nuclear power plants (validation of the integrity assessment):

When proof of life expectancy requires a demonstration that a component will sustain certain loadings during given conditions, the following issues need to be considered:

- Better knowledge of conditions under which the demonstration must be sought,
- Better knowledge of loadings;
- Better understanding, validation and use of modern codes for assessing loading (e.g. computational fluid dynamics codes),
- Better knowledge of the criteria for end of life component ranking:
- direct comparison between an indicator describing the component status and some acceptable limit value,
- more complex criteria generally related to the capacity of the component to sustain the loadings induced by some operating conditions, which are not necessarily the normal ones.

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- reviewing safety justification methodologies for the possible effect of extended service

5.1.4.1.2 Medium term challenges:

European harmonised methodologies to assess integrity and performance in the case of internal and external hazards, selection of indicators and agreement upon end of life criteria is the ultimate goal.

Test procedures for initial qualification, though fully appropriate at the time when older nuclear power plants were built, might be no longer state of the art. The operating conditions of the plants may also have varied considerably from the planned mode of operation. The first aspect refers to the relevance of the qualification tests. The second aspect treats the extension of this initial qualification to a service life longer than the design life. There is also an aspect relating to the qualification procedures to be used in the case of replacements of obsolete materials or components. Simplifications of some procedures based on state-of-the-art understanding of ageing mechanisms could be very beneficial. In summary, the following is required:

Common position on the relevance of qualification tests and on their extension to cover longer-term operation.

5.1.4.1.3 Long term challenge:

- Integrity assessment lessons learned from Gen II nuclear power plants (validation of the integrity assessment)
- European harmonised methodologies to assess integrity and performance in the case of internal and external hazards (including beyond design events e.g. core melt ...)
- Common vision on the safety issues related to the integrity assessment to be:
- Guidelines for European deployment and
- Provide the technical basis for beyond design assessment and for safety justification specifically for long term operation
- Selection of indicators and agreement upon end of life criteria reviewing safety justification methodologies for the possible effect of extended service

5.1.4.2 Description of loads

The challenge is to create methods that take into account the synergetic effect of the different loading modes, by focusing on:

- Different types of load (external, internal hazards, fatigue, creep, dynamic, vibrations, etc.)
- The interaction between the different types of loads (including elastic follow up)
- Categorization and treatment of thermal loads and secondary stresses including geometry misfit and residual stresses and local stress raisers
- Combining fatigue cycle loading (fatigue usage factor, fatigue crack growth)

5.1.4.3 Materials performance and ageing

Material Properties: The highest priority R&D challenges are: relevant and reliable material properties for extended service and creation of a radiation embrittlement database leading to the development of an improved trend curve for RPV life assessment evaluations. In addition efforts should be put on such aspects as the treatment of scatter, corrosion and environmental-fatigue data to correctly apply in integrity assessments. There is the complex issue of how manufacturing processes (including welding, thermal and mechanical treatments and coatings) may affect materials and material properties and how such processes may be improved in the future.

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Ageing and Degradation Mechanisms: The highest priority R&D challenge is to get a better knowledge of ageing mechanisms. The goal is to anticipate and acknowledge ageing issues that may evolve during the foreseen extended life. Identified priorities are corrosion fatigue, irradiation embrittlement, stainless steel cracking and concrete ageing. In case of very long times, possibly exceeding 60 years of operation, several ageing mechanisms that previously have been deemed of lesser importance, such as creep and thermal ageing, may become life limiting factors that need to be addressed.

Improvements are needed in a better physical understanding of all relevant ageing mechanisms and their driving parameters: to identify not only the thresholds for defect initiation and the kinetics for defect propagation, but also the precursor state that leads to defect nucleation. There is a need to be able to make reliable long-term predictions of ageing and its effects. This entails being able to model fundamental phenomena in physics and chemistry at different scales from atomic to macroscopic. Model parameters must be validated against data from laboratory experiments or, most importantly, from operating experience feedback

Short and medium term challenges:

- relevant and reliable material properties for extended service,
- a common understanding of relevant ageing mechanisms on material and component properties from a long-term operational perspective,
- development of advanced multi-scale modelling tools.

Long term challenge:

- European common integrated and qualified physics-based modelling tools.

Modelling of Ageing: The highest R&D challenges concern the needed improvements for a better physical understanding of all relevant ageing mechanisms and their driving parameters. The objective is to identify not only the thresholds for defect initiation and the kinetics for defect propagation, but also the precursor state that leads to defect nucleation. There is a need to be able to make reliable long-term predictions of ageing and its effects. This entails being able to model fundamental phenomena in physics and chemistry at different scales from atomic to macroscopic. Model parameters must be validated against data from laboratory experiments or, most importantly, from operating experience feedback.

Short and midterm challenges:

- understand and simulate neutron/matter interaction and resulting effect on plasticity, ductile/brittle transition, swelling, creep.
- develop physically based fatigue criteria
- investigate on fuel material behaviour: diffusion properties, additive effects
- understand and model the corrosion layer evolution and rupture
- evaluate the structure dynamic behaviour and damaging modes
- understand and control the interaction of metallic materials with their environment: oxidation at high temperature, corrosion in water, stress corrosion cracking

Long term challenges:

- robust simulation of steel behaviour and physically based modelling of fuel behaviour

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- innovative experimental and numerical approaches for dynamic loads (including hybrid testing)
- stress corrosion cracking modelling

5.1.4.4 Ageing Monitoring, Prevention and Mitigation

Ageing Monitoring: R&D activities may contribute to help identifying indicators of ageing phenomena and demonstrating their relevance. Another area for R&D is the development of technologies to monitor such indicators and process the data. Examples may be monitoring of thermal ageing of other components than the reactor pressure vessels, irradiation and embrittlement monitoring of the reactor pressure vessel and water chemistry monitoring by on-line fluid sampling or off-line monitoring e.g. using micro samples or replicas.

Short and midterm challenges:

- demonstration of an intelligent plant condition monitoring systems
- best practices guideline for ageing prevention and mitigation and operational deployment
- advanced repair and replacement technologies
- Long term challenges:
- implementation of intelligent plant condition monitoring systems.
- Implementation of a rule based information and condition information system based on information from different systems

Prevention and Mitigation: Prevention and mitigation methods to avoid initiation or limit propagation of defects and onset of ageing need to be developed. These methods should be common and acceptable to all types of reactors as well as for different regulatory positions within the European countries. Research has to be piloted in this field to increase the understanding of which mechanism are affecting these components.

5.1.4.5 Functionality

Equipment reliability:

Short term challenges:

- Monitor the effectiveness of maintenance and optimize maintenance
- Identify critical equipment and performance criteria
- Optimization of maintenance
- Mid-term challenges:
- Define obsolescence management strategies:
- Develop unified EU guideline for the obsolescence management?
- Develop procedures for the assessment of the adequate change - adequacy of the new component (goal is to not start design change process)
- Equipment qualification
- Qualify under environmental conditions for the LTO
- Functionality qualification (especially analysis of the functionality of safety related motor and air operated valves)
- Harmonization of the qualification and re-qualification methodologies with the selected monitoring techniques (e.g. active power measurement for SSC MOVs)
- Define equipment failure analysis

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- Equipment failure analysis program
- Lessons learned (EU database of active components age-related failures)
- Long term challenges:
- Develop unified EU guideline for the monitoring of the maintenance effectiveness (European maintenance rule)

Industrial Obsolescence: The objectives of R&D are to help to adapt the safety justification (e.g. qualification of software). Furthermore a common approach should be developed either to create versatile technologies, possibly with other industries, or to adapt nuclear procedures to even faster evolving domains, vendors using more and more off-the-shelf technologies and components.

5.1.4.6 Qualification of Structural Integrity Assessment and Lifetime Estimation Procedures for SSCs

Qualification covers pre-normative research related to the integrity assessment of SSCs, in particular the development and standardisation of material and component tests. It also includes the validation and verification of structural integrity assessment and lifetime procedures for SSCs and comparison of different codes & standards for specific issues. These activities support the development & revision of best practice guidance documents, up to the contribution to the development & revision of codes and standards.

The short, mid and long term challenges are:

- Further development and standardisation of small specimen tests, in particular small punch test.
- Comparison of existing codified fatigue assessment methodologies (fatigue curves and/or fatigue models) between each other and with new advanced methodologies on the basis of existing test data from fatigue tests involving specimens or mock-ups with the aim of evaluating their margins.
- Development of empirical/semi-empirical models for prediction of SCC initiation in LWR and super critical water conditions.
- Collection of irradiated and long term aged material samples from real operating reactors for studies on the effects of LTO on reactor materials. This also includes creation and expansion of corresponding databases.
- Comparison of existing codes & standards or other codification approaches in order to determine differences in their degree of conservatism. This could serve as a step towards harmonized European codes & standards and uniform procedures for specific integrity issues.

5.2 Sub Technical Areas (STA)

5.2.1 Integrity Assessment (STA 4.1)

5.2.1.1 Scope

Integrity assessment is to develop suitable methodologies, including the multidimensional computational capability, aiming to take properly into account all the physical phenomena and the different uncertainty sources so as to evaluate the safety margins and their sensitivity to the initiators as well as to the system status and operation [15]. This is required to guarantee the confinement of activity inside the reactor building and NPP with sufficient margins. The principles are

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equally applicable to all generations of reactor designs and they can be either in deterministic or probabilistic terms.

The R&D specific objectives are to collaboratively:

- Identify components and systems that contribute significantly to plant safety and influence the total availability of both new and operating light water reactors, heavy water reactors and gas cooled reactors
- Consider potential damage mechanisms that may contribute to the degradation of components, and consider their consequences under normal operation and design basis emergency (DBE) conditions.
- Thoroughly capitalize on the advances realized in reactor physics, thermal hydraulics, material science etc. to enhance the maturity of the integrity assessment procedures and analysis tools in a well-structured manner.
- Produce global and best practice recommendations.
- Develop the basis for a common understanding between experts at the international level, which may lead to a harmonization of codes and standards and facilitate close collaboration between European and international experts through multilateral frameworks such as IAEA technical working group on PLIM, OECD, NEA group for integrity assessment and ageing (IAGE) and others.
- Common use of the European testing facilities with the objective to develop and validate the integrity assessment procedures and analysis tools focusing exclusively on safety aspects.

R&D topics

5.2.1.1.1 Metallic Components

Short term objectives (0 - 2 years)

- ✓ **Current projects underway that are an important step towards achieving the ultimate goal are associated with:**
- ✓ **Update of design curves including fatigue and environmental effects, indicators and surveillance programs for materials behaviour for long term operation**
- ✓ **Assessing, optimising and developing the use of advanced tools for the structural integrity assessment of non RPV components;**
- ✓ **Development of best practice procedures for assessing structural performance of multi-metal components**
- ✓ **Gaining an understanding of LBB procedures and engineering assessment procedures employed in different European countries;**
- ✓ **Treatment of secondary and residual stresses (including crack closure and load history effects)**
- ✓ **Developing and experimentally validating warm pre-stressing assessment and analytical tools for conditions beyond the present experience & data;**
- ✓ **Further development of unified procedures for WWER components.**

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Mid-term objectives (2 – 5 years):

- ✓ Integrity of RPV internals for long term operation
- ✓ Development of a probabilistic approach of RPV and other safety critical systems integrity assessment for long term operation
- ✓ Harmonization of probabilistic safety assessment with the aim of achieving improved justification for safety factors - strong link to Technical Area 1
- ✓ Implementation of lessons learnt from Gen II NPPs in terms of integrity assessment validation
- ✓ Development of a probabilistic approach of Leak Before Break analyses for long term operation
- ✓ Development of a methodology to include high temperature ageing effects in plant assessments
- ✓ Adapt state of the art methodologies to support the decision for assessing safety and risk impact of changes and modifications of structural components

Long term objectives (up to 10 years):

- ✓ Validated models for the assessment of structural integrity of in-vessel components under high doses of irradiation
- ✓ Benchmarking of safety assessment methodologies including comparison of outputs from deterministic versus probabilistic methods and integration into the safety assessment
- ✓ Integrity behaviour of cladded components
- ✓ Fracture mechanics for thin sections
- ✓ Treatment of non-crack like defects (Corrosion, thinning, pitting, erosion, flow induced corrosion, crevices)

5.2.1.1.2 Civil Works

Short term objectives (0 – 2 years):

- ✓ Modelling of creep behaviour of concrete structures
- ✓ Benchmarking of the impact resistance of concrete slabs

Mid term objectives (2 -5 years):

- ✓ Gain understanding of modelling approaches adopted in different European countries on life time evaluation of civil structures
- ✓ Modelling of cracks opening in prestressed concrete and air and air/steam mixture flow rate through cracks according to different loads, including severe accidents loads

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- ✓ **Determination of the influence of cracking at an early age of the presence of rebards and construction joins**

Long term objectives (up to 10 years):

- ✓ **Develop probabilistic and deterministic methodologies to evaluate the impact of internal hazards (hydrogen explosion, pipe whip impact)**
- ✓ **Develop probabilistic and deterministic methodologies to evaluate the impact of external hazards, (seismic event, aircraft impact, explosion)**
- ✓ **Integration of probabilistic and deterministic methodologies**

5.2.2 Description of Loads (STA 4.2)

5.2.2.1 Scope

The underlining principles are to establish a complete inventory of the loading systems and their categorization as to their effect on the structural integrity of reactor components.

R&D topics

5.2.2.1.1 Metallic Components

Short term objectives (0 – 2 years):

- ✓ **Development of guidance in order to more accurately predict fluid to component wall heat transfer (CFD, Computational Fluid Dynamics) for thermal fatigue analysis**
- ✓ **Dynamic response of reactor internals to Loss of Coolant Accidents (LOCAs)**

Midterm objectives (2 -5 years):

- ✓ **Improve numerical prediction methods for residual stresses in dissimilar metal welds**

Long term objectives (up to 10 years):

- ✓ **Investigation of combined fatigue and tearing fracture resistance under high asymmetry cycles and random high cyclic loads**
- ✓ **Improved methodologies for thermal stratification and mixing assessments (pipings and T-junctions)**
- ✓ **Harmonisation of load evaluation methodologies**
- ✓ **Methodologies for establishing the significance and ranking of external loads for deterministic and probabilistic assessments**
- ✓ **Fluid structure interaction under turbulent flow conditions (resulting impact force)**
- ✓ **Development of improved guidelines for lifetime extension of bolts and flanges (including defect tolerance)**
- ✓ **Enhancement of dynamic safety assessment methods for external events**

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5.2.2.1.2 Civil Works

Long term objectives (up to 10 years):

- ✓ **Increase the understanding of hydrogen production and hydrogen venting during service accident conditions and of hydrogen from zirconium oxidation and from radiolysis in water and hydrogen transport and absorption in water, gas phase and on walls**
- ✓ **Containment phenomena in dry-well stratification and heat transfer into the condensation pool with the aim of reducing uncertainties**
- ✓ **Containment phenomena in dry-well - load assessment from source to loads on walls under different voiding conditions in the condensation pool with the aim of reducing uncertainties**

5.2.3 Materials Performance and Ageing (STA 4.3)

For the properly understanding of the performance of materials relevant to structural components it is important that their properties and the effect that ageing mechanisms can have on such materials are known. In considering materials performance and ageing, it is useful to consider in turn the aspects of material properties, ageing and degradation mechanisms and modelling of ageing.

This sub-section deals with:

i. Material Properties
ii. Ageing and Degradation Mechanisms
iii. Modelling of Ageing

5.2.3.1 Materials Properties

5.2.3.1.1 Scope

A key aspect in safely operating a nuclear power plant is the assurance that the material is able to perform adequately under all the required loading, environmental and time dependent conditions. In particular for long term operations it is essential to reliably assess the change in material characteristics and properties during operation. The main objectives of materials performance and ageing are to address in a collaborative way the following topics:

- Qualification of the materials used in nuclear power plants
- Material behaviour under operating conditions
- Material behaviour under accidental and emergency conditions
- Aging of materials for long term operation
- Improvement of testing and monitoring materials performance
- Modelling and predicting the materials behaviour under various conditions
- Common use of materials testing and other experimental facilities
- Harmonization and codification to quantify materials performance

R&D topics

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5.2.3.1.1.1 Metallic Components

Short term objectives (0 – 2 years):

- ✓ **Development and testing of accelerated (SCC) test methods for life time prediction**
- ✓ **Investigation of the fatigue endurance roots**
- ✓ **Development of characterization methodologies and qualification procedures for dissimilar metals welds (DMWs)**

Midterm objectives (2 -5 years):

- ✓ **Establish the role of manufacturing conditions such as cold work, micro-structure, surface finish on material performance**
- ✓ **Further develop testing and qualification procedures for miniaturized specimens**

Long term objectives (up to 10 years):

- ✓ **Stress corrosion cracking in Ni based alloys**
- ✓ **Development of testing procedures to account for environmental effects and high temperature on mechanical properties**
- ✓ **Effect of dynamic strain ageing on fracture toughness of materials**
- ✓ **Improve identification of radiation-resistant materials**
- ✓ **Guidance for manufacturing processes based on material performance studies**
- ✓ **Development of joining and welding procedures (including mixed welding, welding of thick components etc.)**
- ✓ **Creation of materials database including coolant (primary and secondary), thermal, mechanical and irradiation effects (single and combined effect)**
- ✓ **Improve embrittlement trend curves (on the basis of constantly monitoring relation between the predictions of current ETCs and emerging data)**
- ✓ **Standardization of test procedures and post-test investigation strategies**

5.2.3.1.1.2 Concrete

Long term objectives (up to 10 years):

- ✓ **Development of an inventory (in the form of a database) on the exact composition of concrete used for containment buildings of operating reactors and reactors under construction.**

5.2.3.1.1.3 Polymer Materials

Midterm objectives (2 -5 years):

- ✓ **Development of acceptance criteria for replacement taking into account correlations of all properties (composition versus performance)**

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- ✓ **Establishment of methodologies for testing and characterizing polymers used in nuclear power plants**
- ✓ **Characterisation of coatings and paints for use on structural components and civil structures**

Long term objectives (up to 10 years):

- ✓ **Irradiation ageing of polymers: Investigate the influence of dose rate and irradiation time when performing accelerated irradiation**
- ✓ **Synergistic effects of stressors at polymers: Investigate synergistic effects of irradiation, heat, moisture (water), vibration and other existing stressors affecting the environmental impact of polymers with special considerations taken for elastomers**
- ✓ **Ageing mechanism of organic materials: Investigation of degradation of organic materials in synergy with vibrations**
- ✓ **Irradiation ageing of polymers with various nuclides: Investigate and build a model to assess influence of irradiation sources deviating from normal operation sources and DBE sources**
- ✓ **Ageing affected by manufacturing: Production, storage and transport of polymeric raw materials can impact the life-time of the materials. Identify how this is managed today and possible need for improvements (e.g. state-of-the art report)**

5.2.3.2 Ageing and Degradation Mechanisms

5.2.3.2.1 Scope

The specific mechanisms and phenomena that are responsible for the ageing behaviour of materials in LWRs can be assessed by methods and procedures based on representative experimental data. For the example of the reactor pressure vessel (RPV) it can be stated that during the operation of a nuclear power plant the neutron flux affects the RPV material properties. Irradiation of ferrite steels by (fast) neutrons with sufficiently high energy causes interactions of the neutrons with the atoms of the RPV steel influencing the microstructure of the irradiated material. The mechanisms responsible for these irradiation induced embrittlement effects are well known in terms of matrix damage, Cu precipitation containing Ni, Mn, Si, and grain boundary segregation of P. To assess the real ageing behaviour it is useful to investigate RPVs of decommissioned plants.

Thermal aging refers to hardening caused by thermally activated diffusion of alloying elements or impurities. The potential for thermal aging embrittlement in LWR RPV steels for times of up to 40 years is considered as low, but cannot be entirely dismissed on the basis of available data.

It is known that hydrogen may contribute to the embrittlement of RPV steels, but only under very specific conditions. The resistance of the steel for hydrogen embrittlement is dependent on the chemical composition, fluence, irradiation temperature and the type of irradiation-induced defects.

R&D topics

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5.2.3.2.1.1 Metallic Components

Short term objectives (0 – 2 years):

- ✓ **Investigation of microstructural and mechanical effects in RPV steels caused by long term irradiation leading to the improvement of RPV safety assessment of existing European LWRs under long-term operation and Generation-III reactors under construction (supporting RPV ageing management and plant life extensions)**
- ✓ **Investigation of thermal ageing in low alloy steels**
- ✓ **Investigation of growth and creep under irradiation in internal structures by way of benchmarking different numerical methods for microstructure evolution**

Midterm objectives (2 -5 years):

- ✓ **Investigation of possible negative side effects of injecting zinc into the primary circuit of LWRs in order to reduce dose rates**
- ✓ **Improving the knowledge regarding stress corrosion cracking (SCC) initiation with respect to the field requirements, including experiments with relevant advanced characterization**
- ✓ **Investigation of the fabrication process and initial surface condition of a metallic component on the SCC time-to-failure**
- ✓ **Creation of a radiation embrittlement database leading to the development of an improved trend curve for RPV life assessment evaluations**
- ✓ **Investigation of crack initiation by irradiation assisted stress corrosion cracking (IASCC) and the creation of a database of experimental data**
- ✓ **Investigation of irradiation in tough low activation V alloyed steel for future RPVs**
- ✓ **Study of the crack morphology and its influence on leak rates for provision of defined parameters and further improvement in the accuracy of leak-rate/LBB models**
- ✓ **Development of computational fluid dynamics (CFD) based methodology for more accurate predictions of flow assisted corrosion (FAC) rates for complex system components**

Long term objectives (up to 10 years):

- ✓ **Evaluation of methodologies to determine and optimize fatigue endurance of safety critical components with the aim of gaining a better knowledge of environmental fatigue effects in PWRs**
- ✓ **Investigation of corrosion and hydrogen embrittlement for structural materials**
- ✓ **In situ verification of swelling macroscopic effects by in-plant measurements and the establishment of a suitable database to be used for long term operation justifications**

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- ✓ **Investigation of degradation of buried metallic pipes**
- ✓ **PWR secondary water chemistry: components behaviour in the absence of N2H4 conditioning**

5.2.3.2.1.2 Concrete

Short term objectives (0 – 2 years):

- ✓ **Identification of other concrete pathologies that may occur during concrete ageing and the development of a better understanding of physical phenomena responsible for these (e.g. corrosion of reinforcement, swelling reactions (or due to RAG and ISA), carbonation, attack by chloride ions, leaching of cemented materials, lixiviation)**
- ✓ **Evaluation of chloride initiated corrosion of concrete reinforcements**
- ✓ **Better understanding of the physical phenomena responsible for alkali aggregate reaction (RAG) and internal sulphate attack (ISA)**
- ✓ **Development of accelerated methods to provoke the development of RAG and ISA, and methods to characterize the development of RAG and ISA**
- ✓ **Investigation of galvanic corrosion on reinforcement in the vicinity of the main cooling water pumps**

Midterm objectives (2 -5 years):

- ✓ **Development of simulation methods to predict ageing in critical components and buildings**
- ✓ **In-depth studies on degradation mechanisms like ISR (Internal Sulphate Reaction – delayed formation of ettringite (DEF)) and AAR (Alkali Aggregate Reaction, mainly alkali-silica reaction)**

5.2.3.2.1.3 Polymer and Electrical Equipment

Short term objectives (0 – 2 years):

- ✓ **Evaluation of polymeric material in concrete constructions and sealing applications**

5.2.3.3 Modelling of Ageing

5.2.3.3.1 Scope

The underlying principles of a physically based material and structure management are to get a robust prediction of structure behaviour under normal, accidental or extreme conditions. Faulted condition may result from the material behaviour (vessel, internals, clad), or from the global mechanical behaviour (external hazards, including human aggressions). In addition the efficiency and safety shall be enhanced by modelling basic chemical phenomena, whereby the interactions between the metallic materials and their environment has to be investigated in detail, in order to understand these interactions from the atomic scale up to the scale of components of end products or complex structures, and to be able to make reliable long term prediction of the evolution of the materials in

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service conditions. Thermodynamics and thermo kinetics parameter under severe accidents conditions are to be predicted.

Multi scale modelling and simulation of chemical phenomena, including data bases, is a long term goal. Molecular dynamics tools can be considered at a first focusing step.

R&D topics

5.2.3.3.1.1 Metallic Components

Short term objectives (0 – 2 years):

- ✓ **Better understanding of physical mechanisms driving the ageing of materials, by combining advanced experimental characterization of ageing materials with the use of atomistic and nanostructure evolution modelling tools, that should include the effect of local phenomena and the interaction between them, by using powerful numerical tools, including those required to homogenize in time and space;**
- ✓ **Development/improvement of models to predict crack initiation under various ageing mechanisms**
- ✓ **Performance of representative in-pile experiments and subsequent materials characterization for the validation of the modelling tools**
- ✓ **Investigation of ageing effects, especially irradiation hardening and subsequent embrittlement, of reactor components, especially the vessel, by multi-scale modelling**
- ✓ **Investigation of mechanisms determining irradiation creep and swelling in austenitic steels and development of relevant predictive physical models, based on a multi-scale modelling approach**

Mid-term objectives (2 -5 years):

- ✓ **Validation and parametrisation of the toughness module for VVER steels**

Long term objectives (up to 10 years):

- ✓ **Simulation of welding and manufacturing processes with a view to evaluating ageing effects**
- ✓ **Investigation of the correlation between microscopic properties and macroscopic mechanical behaviour of aged materials on the basis of combined experimental and theoretical analysis of dislocation dynamics.**
- ✓ **Development and parameterisation of advanced oxide film models**
- ✓ **Establishment of relevant and reliable material properties for extended service inspection, maintenance and repair**

5.2.3.3.1.2 Civil Works

Short term objectives (0 – 2 years):

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- ✓ Development of models to represent the effect of RAG and ISA on concrete mechanical properties and civil work structure thermo mechanical behaviour (see above)

Mid-term objectives (2 -5 years):

- ✓ Development of models to represent the effect of other concrete pathologies than RAG and ISA on concrete mechanical properties and on civil work structure thermo mechanical behaviour (see above)
- ✓ Development of simulation methods to predict ageing in safety critical components and buildings

5.2.4 Ageing Monitoring, Prevention and Mitigation (STA 4.4)

5.2.4.1 Ageing Monitoring

5.2.4.1.1 Scope

Component ageing has to be monitored over the nominal and extended service life, in order to determine ageing mechanisms correctly. The overall goal is to monitor and understand environmental conditions in the power plants as well as their impact on the functionality of safety relevant components and structures.

R&D topics

5.2.4.1.1.1 Metallic Components

Midterm objectives (2 -5 years):

- ✓ Development of on-line monitoring tools for advanced water chemistries in BWRs and PWRs
- ✓ Investigation of ultrasonic on-line monitoring of piping in NPPs
- ✓ Investigation of the storage of irradiated and long term aged materials for microstructural and non-destructive studies of RPV steels
- ✓ Improvement of the lifetime justification of SSCs based on dosimetry best practices
- ✓ Improvement of sampling techniques for water chemistry analysis in LWRs

Long term objectives (up to 10 years):

- ✓ Demonstration of the impact acceptability of fractured piping and jet impingement effects
- ✓ Further develop of NDE methods for on-line monitoring
- ✓ Confirmation of the safety operational capability of SSCs beyond design lifetime (long term operation) by demonstrating that NPP equipment will fulfil its function during the entire assumed operation period
- ✓ Development of improved methods for the inspection of internals

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- ✓ Optimum design and realization of pipe whip restraints
- ✓ Investigation into life extension of steam generator pipes, including guide for the evaluation of eddy current testing signals (plugging criteria)
- ✓ Optimum design and realization of barriers against jetting medium

5.2.4.1.1.2 Concrete

Short term objectives (0 – 2 years):

- ✓ Development and provision of effective operational methodologies for inspection and tools for extended operation by consideration of such aspects as tightness of reactor containment, serviceability of waterways and integrity of cooling towers

Mid-term objectives (2 -5 years):

- ✓ Development of methods for assessing the current state of concrete structures by way for example of retrofitting embedded humidity sensors for monitoring the quality of the concrete with respect to penetration of boric acid (and other liquids) in to the structure
- ✓ Development of non-destructive examination (NDE) methods for concrete (e.g. ultrasonic techniques to detect and characterize cracks embedded in the concrete and ultrasonic guided wave methods to assess the pre-stressing state in the tendons)

5.2.4.1.1.3 Polymers and Electrical Equipment

Short term objectives (0 – 2 years):

- ✓ Detection of local and global degradation of cables and establishing a correlation for residual life estimation by using in-situ non-destructive electrical techniques for in-service full length measurements (for medium and low voltage cables).
- ✓ Sharing of best practices and the development of new techniques to detect ageing and to limit the degradation on I&C components and the evaluation of new technology and methods to be used for the purpose of I&C modernization.
- ✓ Development of monitoring techniques from the Motor Control Centre (MCC) providing remote assessment from the complete supply loop of electrical appliances

Midterm objectives (2 -5 years):

- ✓ Determination of a life-time criterion for non-LOCA cables that are important for safety and also for non-safety critical cables
- ✓ Determination of ageing monitoring methods for cables under the fire protection layer
- ✓ Long term objectives (up to 10 years):
- ✓ Development of test methods and devices to predict ageing phenomena and counter obsoleteness (e.g. the adoption of programmable digital automation)

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- ✓ **Definition of requirements, methodologies and assessment of the risks associated to the implementation of new technological solutions**

5.2.4.2 Prevention and Mitigation

5.2.4.2.1 Scope

The issue is associated with components that are usually very difficult and expensive to replace and may not be readily observable. For example, reinforcements that are used in concrete applications are generally considered difficult in evaluating, since they often are found within concrete constructions and non-visible. Nevertheless, these components are essential for the integrity of civil engineering structures and should therefore be considered as essential for the overall plant safety.

R&D topics

5.2.4.2.1.1 Metallic Components

Midterm objectives (2 -5 years):

- ✓ **Investigation into annealing of RPV materials**
- ✓ **Investigation of the effect of water additives on vessels, piping and components in the primary loop of the reactor coolant systems in BWR, PWR and VVER plants**
- ✓ **Optimisation of the secondary side water chemistry to minimize magnetite deposition**
- ✓ **Provision of information and data on HDPE materials for intended use for service water piping**
- ✓ **Assessment of methods to evaluate vibration levels in main steam piping and identification of design modifications to reduce such levels**
- ✓ **Implementation of a condition information system supporting condition based maintenance (CBM) based on worldwide experience and relevant data**

Long term objectives (up to 10 years):

- ✓ **Simulation of residual stresses and development of mitigation measures including overlay welding**
- ✓ **Development of improved mitigation techniques for IGSCC: e.g. modified water chemistries or development of other new techniques**
- ✓ **Assessment of low temperature crack propagation susceptibility for primary circuit and core materials (welds, cast materials: Inconel 182, alloy 690 cold worked,…) in transient conditions with the aim of optimizing the water chemistry**
- ✓ **Development of an advanced primary water chemistry for VVER and PWR systems based on coolant treatment for radioactive waste reduction and lifetime extension of primary systems components**
- ✓ **Investigation of full service decontamination**

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5.2.4.2.1.2 Concrete

Short term objectives (0 – 2 years):

- ✓ **Development of a simple anode system (thermally sprayed coating) for simple corrosion protection of concrete reinforcements**

5.2.5 Functionality (STA 4.45)

Under this topic, the issues of equipment qualification, technology obsolescence and maintenance are considered.

5.2.5.1 Equipment Qualification

5.2.5.1.1 Scope

Qualification of equipment important to safety in nuclear power plants ensures its capability to perform designated safety functions on demand under postulated service conditions including harsh accident environment. Test, analysis, operating experience and combinations of these methods may be applied in a variety of ways to establish qualification. It involves a planned test sequence subjecting equipment first to simulated normal operating conditions, including ageing, and then to PIE service conditions. The qualification process should clearly identify the duration of qualified life or the qualified condition, coupled with condition monitoring intervals and maintenance requirements.

R&D topics

- ✓ **Development of criteria and test methods to compare qualified vs. installed equipment. This will enable to define an applicable procedure to evaluate the equipment equivalency and solve the problems if the "finger prints" between compared materials of installed and qualified equipment differ.**
- ✓ **Development of a standard procedure for detection, measurement and evaluation of the not allowed or limited amount of elements (halogens, S, BP etc.) content in polymers.**
- ✓ **Enhancement of long term operation (60+) simulation for equipment qualification (artificial equipment ageing) based on better and more reliable ageing models to simulate normal operation stressors (temperature, radiation, mechanical stress, chemicals, voltage, ohmic heating, ...).**
- ✓ **Study on reliable qualification and thermal ageing of semi-crystalline materials with inverse ageing phenomena effect, like e.g. crosslinked polyethylene.**
- ✓ **To define convenient and efficient way of the (severe) accident conditions simulation during the qualification test. Proposed solution have to take into account post-accidents periods that may be as long as one year and more (necessity of the test acceleration) and influence of beta, gamma, neutron irradiation on polymers.**
- ✓ **To solve problems with the oxygen supply into LOCA chamber during the qualification tests simulating DBE and post DBE situation. Explanation of**

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oxygenation of LOCA chamber from IAEA NP-T-3.6.: Test chambers used for simulation of the thermal profile of a LOCA during qualification are often quite small, with a limited partial pressure of oxygen present due to their small volume. Diffusion limited oxidation effects can then become significant in terms of restricting the total degradation that will occur during the thermal profile. Oxidation will not necessarily be heterogeneous, but is likely to be significantly less than would occur in during an actual LOCA, where the oxygen content is not restricted. This means that the degradation produced by the LOCA may be significantly underestimated. Consideration should be given to supplementing the oxygen content of the test chamber during the thermal profile to ensure that diffusion limited oxidation does not take place.

5.2.5.2 Technology Obsolescence

5.2.5.2.1 Scope

Obsolescence is the important plant concern not just materials or engineering. There are three main types of obsolescence (possibly impairing nuclear safety) setting up the call scope:

- knowledge erosion,
- current standards and regulation deviation,
- technology obsolescence (lack of spare parts and qualified suppliers).

The long term objective is to undertake activity aimed at mitigation of obsolescence impact on NPPs safety and availability.

This subarea is focused to the problems related to the Technology obsolescence.

R&D topics

Midterm objectives (2 -5 years):

- ✓ **Development of the necessary obsolescence management approaches taking into account:**
- ✓ **Potential impact evaluation (determination and scoring)**
- ✓ **Equivalency evaluation**
- ✓ **Reverse engineering the item**
- ✓ **Modification of the plant**
- ✓ **Proactive obsolescence**

Long term objectives (up to 10 years):

- ✓ **Development of the new (exchangeable) ways of equivalent components or their spare parts production (Cross cutting with TA6).**

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5.2.5.3 Maintenance

5.2.5.3.1 Scope

Maintenance is associated with ensuring that all the systems are maintained in a satisfactory way that will ensure equipment reliability throughout the lifetime of required use.

R&D topics

Midterm objectives (2 -5 years):

- ✓ **Justification of the monitoring systems supporting condition based maintenance (CBM).**
- ✓ **Integration of operational feedback (experience) provided by the monitoring systems into the planning of maintenance activities (CBM)**
- ✓ **Modernization of pre-stressing control units to ensure fluent control of stress during the pre-stressing procedure in order to minimize the influence on lifetime of the tendons, to automate the process and to increase the accuracy**

Long term objectives (up to 10 years):

- ✓ **Implementation of fleet wide monitoring data exchange systems in order to provide more accurate input data to maintenance teams**

5.2.6 Qualification of Structural Integrity Assessment and Lifetime Estimation Procedures for SSCs (STA 4.6)

5.2.6.1.1 Scope

Qualification in the context of structural integrity assessment and lifetime estimation of SCCs is the validation and verification of structural integrity assessment and lifetime estimation procedures for SSCs. This can be accomplished analytically by comparison of the underlying models/procedure that is subject to qualification with a more complex theoretical model and/or experimentally by using tests. Analytical qualification of methodologies also includes comparison of different codes & standards for specific issues.

All activities that contribute to the validation and verification of structural integrity assessment and lifetime estimation procedures for SSCs are part of this sub-area. Also the development & revision of best practice guidance documents, up to the contribution to the development & revision of codes and standards is also part of this sub-area, as well as pre-normative research related to the structural integrity assessment of SSCs, in particular the development & standardisation of material and component tests.

R&D topics

5.2.6.1.1.1 Metallic Components

Midterm objectives (2 -5 years):

- ✓ **Further development and standardisation of small specimen tests, in particular small punch test.**

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- ✓ **Comparison of existing codified fatigue assessment methodologies (fatigue curves and/or fatigue models) between each other and with new advanced methodologies on the basis of existing test data from fatigue tests involving specimens or mock-ups with the aim of evaluating their margins.**
- ✓ **Development of empirical/semi-empirical models for prediction of SCC initiation in LWR and super critical water conditions.**
- ✓ **Collection of irradiated and long term aged material samples from real operating reactors for studies on the effects of LTO on reactor materials. This also includes creation and expansion of corresponding databases.**

Long term objectives (up to 10 years):

- ✓ **Comparison of different numerical methods on microstructure evolution of irradiation creep and swelling.**
- ✓ **Comparison of existing codes & standards or other codification approaches in order to determine differences in their degree of conservatism. This could serve as a step towards harmonized European codes & standards and uniform procedures for specific integrity issues.**

5.2.7 Cross-cutting R&D (Interactions with other Technical Areas) (STA 4.7)

Related activities should be checked with and included in TA 6

- ✓ **Some outcomes of the Fukushima's events could be considered for pre-normative research activities, e.g., for instrumentation, on seals and grease, able to sustain high temperatures and pressure…,**

Related activities should be checked with and included in TA 1

- ✓ **As far as the improvement of methodologies, the above mentioned fields of endeavour could be considered for pre-normative R&D:**
    - o **Boltzmann–Bateman reference calculations,**
    - o **Fuel modelling / neutronics tight coupling,**
    - o **Fluid Mechanics and (two phase flow) thermal-hydraulics,**
    - o **Fully integrated multi-physics simulation ("numerical" fuel assembly, core, steam generator),**
    - o **Advanced Uncertainty Management,**
    - o **Problem solving: advanced tools,**
    - o **Upgraded, innovative instrumentation with advanced interpretation,**
    - o **Ab initio modelling of metallic and concrete structure behaviour**

The standardization of the design of the NPPs safety-classified components is a major step towards an enhanced safety (provided similar operating uses and in service inspection and maintenance

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arrangements), the whole stakeholder community for nuclear energy could profit from. Standardization should widely ease the use and the replacement of these components whenever needed and simplify the safety assessment, as well.

5.2.7.1 Harmonization topics

The progressive process which should lead to the standardization of component design has to rely on the already existing practices; that is why it is worth:

- Elaborating a critical overview-document collecting and analyzing the suitable standards already publicly available (emanating from the International Standards Organizations ISO and IEC, from CEN/CENELEC or other sources, as well) or under preparation, to meet specific needs for products and services in the nuclear sector;
- Contributing to improve synergies between innovation/research projects and civil standardization for nuclear energy, noting the various proposals in the Commission Action Plan for European Standardization.
- Where no suitable standards exist, the R&D carried-out within the NUGENIA Topic Area 7 should contribute to:
- Define the roadmap enabling the definition of these standards relying on suitable pre-normative research programmes (see § 7.2) to be carried-out in collaborative frameworks;
- Assess whether the provisions of the Vienna and Dresden Agreements, or similar arrangements, might be considered in future in this sector;
- Identify and give due consideration to any other specific issues linked to European legislation;
- Attract the stakeholders' attention and advise them on any strategic issue concerning standardization.

According to the Safety Requirements for the Design of Nuclear Power Plants, all Structures, Systems and Components (SSCs) relevant to safety shall be first identified and then classified on the basis of their function and significance with regard to safety. This classification should identify the appropriate requirements, codes and standards to be applied in their design, manufacturing, construction and operation. Currently, the SSCs of similar systems could be differently safety-classified in different countries due to the classification scheme adopted. This turns out a major limitation to standardize the new- built requirements. IAEA is now developing DS 367 on "Safety classification of SSCs". When finalized, this document will help the regulators, the vendors and the licensees to adopt a common ground to establish a nationally-based international safety classification scheme. Accounting for the current trend, the R&D should focus on:

- **Clarifying the technical data that will lead to the safety classification decision and then converge on common criteria,**
- **Focus on common and harmonised understanding of the grey area (mostly safety related SSCs) which engenders is the most conflicting when it comes to international comparisons**
- **Anticipate the need for data on passive systems (criteria, periodic testing …)**

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- Examples of groups/projects are:
- The WGAMA (OECD/CSNI) on analysis and management of accidents, and the SARNET network concerning the assessment of severe accident phenomena,
- The WGRISK (OECD/CSNI) on probabilistic safety assessments,
- The WIAGE (OECD/CSNI) on integrity and ageing of structures and components,
- The ASAMPSA2 (…) on level 2 PSAs,
- The Safety margins Action Plans & the SM2A (both OECD/CSNI projects) aimed at providing a methodology to assess the safety margins variation in case of design and/or plant operation modification, such as the power up-rating and the prolonged operation
- The IAEA CRP "benchmarks".

In the short-term it would be worth:

- Defining a list of existing guidelines and practices, identifying the most urgent ones for harmonisation,
- Defining a list of Best Practices Documents (BPD),
- Updating the procedure for BP documents,
- Transferring the NULIFE, SNETP TWG II&III, SARNET documents on harmonization to NUGENIA (e.g. OPERA map) and possibly update them
- Introducing harmonization targets in the future NUGENIA projects. (All NUGENIA projects should start with a State of the art Report)
- Identifying routes for dissemination, through the organisation of Workshops and Conferences
    - Preparing and addressing Questionnaires.

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6 TECHNICAL AREA 5 - Fuel Development, Waste and Spent Fuel Management and Decommissioning (TA5)

Technical area leader: Steve Napier (NNL)

6.1 Executive Summary

6.1.1 Scope

Technical Area 5-A covers the development of nuclear fuel for existing, advanced and innovative core designs including assembly and control rod considerations, within assembly instrumentation, manufacturing issues, transport, use within reactors (nuclear fuel behaviour mechanisms including post-irradiation examination), spent fuel Interim storage (wet and dry), reprocessing and the production of recycled fuel from the products of reprocessing (reprocessing is covered by TA5-B). It includes the safety issues linked with fuel behaviour in normal operation, transient and accident conditions in addition to the safety of the fuel cycle including criticality prevention, transport safety and interim storage of spent fuel. Of particular importance is that the roadmap takes account of emerging lessons from the Fukushima accident to propose research, development and innovation to improve the safety and resilience of the existing and new build LWR reactor fleet.

TA5-A does not cover any significant aspects of the availability of and mining of uranium (and thorium) ores, extraction from the ore or uranium enrichment. Availability, mining and extraction are considered in other fora and the details of enrichment are closely guarded for non-proliferation reasons, such that NUGENIA is not the appropriate forum for development.

TA5-A has an important interface with TA1 and TA2, which deal with NNP safety and risk assessment (including criticality prevention) and-severe accidents (including those in spent fuel pools) respectively. In addition, TA5-A has connections with TA3 in terms of core optimization and chemistry which is closely linked with fuel behaviour mechanisms. TA5 is also connected to TA6 regarding fuels for innovative LWRs. Finally, TA5-A connects to TA8 regarding post-irradiation fuel inspection methods, which bring real data to use for computational codes validation.

TA5-B focuses on waste management and decommissioning. This incorporates the dismantling and decommissioning of nuclear power plants and fuel cycle processing facilities as a last step in their lifetime. It also includes the treatment of fuel waste by reprocessing which can generate materials suitable for the production of recycled fuel (covered by TA5-A). The subsequent immobilisation of waste and the production of suitable waste forms are also covered. Finally it also considers waste minimisation and recycle of non-fuel materials.

TA5-B has links to TA4 in terms of understanding the integrity of structural components of NPPs and the implications for dismantling and decommissioning. It also has links especially in the fields of waste forms and waste minimisation with IGD-TP (Implementing Geological Disposal of radioactive waste Technology Platform) which is in the process of drafting a research agenda and deployment plan for this crucial area.

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6.1.2 Objectives

The objectives of Technical Area 5 are to improve the operation of NPPs and the nuclear fuel cycle in the fields of in-reactor and out-of-reactor nuclear fuel management and nuclear waste management so as to be more:

- Safe
- Sustainable
- Secure (proliferation resistant)
- Environmental friendly
- Reliable
- Economic

The existing technology is already at a mature stage of development, however further improvements are still sought. Such work will include meeting the safety requirements of regulatory bodies and enacting recommendations of the relevant international organisations such as the IAEA, OECD/NEA and WNA.

6.1.3 State of the art

The main nuclear fuel suppliers in Europe are currently AREVA, Westinghouse, GNF and TVEL. The existing theoretical and experimental knowledge base consists of the vendors' own R&D, the operational experience of utilities, research entities such as national laboratories, technical service providers and universities and international organisations in particular the IAEA, OECD/NEA and WNA. Experimental facilities including research reactors, hot cells and hot laboratories are available for research and testing.

Uranium dioxide (UO2) enriched up to 5% in form of solid or annular pellets in zirconium alloy cladding remains the most widely used fuel in European reactors, primarily LWRs. MOX (mixed uranium-plutonium oxide) fuel also is used in limited quantities mainly in France where large scale reprocessing and manufacturing facilities are available. For LWR fuel assemblies the main construction materials are again zirconium alloys, with nickel alloys and stainless steels also used for some assembly components. Control rods are currently manufactured primarily from either silver-indium-cadmium (Ag-In-Cd) alloys or contain boron carbide (B4C).

The properties of all these materials and fuel assembly design are relatively well established; however the drive for continuous improvement in safety, reliability and performance through improved understanding and evolutionary adjustments necessitates further studies and ongoing development.

Fuel performance and reactor physics codes have also been developed over a number of years and validated using data from operation and dedicated experimental programmes. These are routinely used for simulation of normal operation, transient conditions and accident scenarios. In spite of this, enhancements in simulation methods are continually sought, facilitated to a significant extent by ever improving computing capabilities. In particular, there is a desire for better mechanistic understanding of fuel behaviour in-reactor.

Spent fuel management of the various nuclear fuel types from both commercial and research reactors is a mature practice benefiting from the accumulated knowledge and experience acquired over more than fifty years. Nevertheless, there is a space for improvements in safety, security

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(proliferation resistance), economics and environmental aspects. Spent fuel management is carefully regulated by national regulators, usually reflecting recommendations of international organisations, in particular the IAEA. Within the EU, a range of spent fuel storage arrangements are employed, in some countries fuel is stored primarily at the reactor site where it was irradiated, whereas in other countries centralised facilities exist for interim/long term storage following an initial cooling period at the reactor site. The transport of spent fuel is well established operation.

The reprocessing of UO_2 and metallic fuels has been well established within some parts of the EU. However, there is the potential for the extension of existing reprocessing techniques to more challenging fuels, such as high burn-up fuels and multiple recycled MOX, as well as development of advanced reprocessing techniques such as pyro-processing. In addition, the reprocessing of potential novel LWR fuel compounds such as those proposed for Accident Tolerant Fuels (ATF), needs to be considered in particular by linking with Gen IV programmes where there is currently greater experience of some of the proposed materials. In addition, there are many exotic fuel forms which been produced in Europe as a result of past development programmes including Gen IV pre-cursors which are difficult to reprocess but may not be suitable for direct disposal in a geological repository.

A number of decontamination, waste treatment and conditioning methods and technologies have been developed and are used along with management of special categories of waste were also developed (Tc and C-14 waste, Be, graphite, mixed radioactive and toxic waste, etc.). Nevertheless, potential for improvement to reduce cost and risks is still not exhausted. Methods of reuse and recycling of various materials (metals, concrete) have been also introduced. Experience from decommissioning and dismantling of nuclear facilities is being continually accumulated allowing drafting of guidelines and the first best practices.

6.1.4 Challenges

Technical Area 5 has some important overarching challenges to address, which include:

- Increasing the safety margins of nuclear fuels and improving behaviour under operation and accident conditions including the development of new accident tolerant fuel (ATF) forms;
- Improved economics of nuclear fuels in particular through allowing high burn-ups and potential new high density fuel forms;
- Increased nuclear fuel recycling through the use of reprocessed uranium and improved MOX fuels including multiple recycled MOX, high plutonium (Pu) content and minor actinide (MA)-bearing MOX;
- Other Pu- and MA-bearing fuels including thorium oxide (ThO2) matrix (thorium MOX) and inert matrix fuels (IMF) for Pu and MA burning applications;
- Improvements in assembly design and optimisation including attempted elimination of grid-to-rod fretting and the prevention or mitigation of damage by foreign objects;
- Improvement of fuel performance and safety computer codes and their validation by reducing uncertainties and extending experimental data;
- Introduction of more mechanistic and multiscale modelling packages for the assessment of both existing and innovative fuel designs;
- Improvement of post-irradiation examination (PIE) methods;

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- Maintaining of key experimental facilities (research reactors, hot cells and hot laboratories) and expanding their capabilities to meet future requirements ;
- Handling and storage of leaking fuel assemblies (spent fuel pool and interim wet and dry storage);
- Handling of fuel and casks after longer term storage including the interface to deep geological repository;
- Spent fuel burn-up credit challenges (code validation and licensing issues);
- The reprocessing and recycling of challenging fuels (e.g. high burn-up, multiple recycled MOX) and advanced fuels (e.g. ATF) as well as advanced reprocessing technologies (e.g. pyro-processing);
- Use of advanced IC tools for development of integrative waste management strategies;
- Minimisation of waste production due to design and material selection and operational measures and development of advanced waste treatment and conditioning technologies;
- Development of efficient dismantling technologies for structures and components including remote dismantling techniques;
- Waste minimisation strategies for decommissioning including safe release of material to the environment, recycle/reuse, disposal to VLLW repositories along with reliable and cost effective activity measurement techniques.

6.2 Sub Technical Areas (STA)

6.2.1 Fuel Development, Computational Codes and Spent Fuel Management (STA 5-A)

6.2.1.1 Fuel development for existing, advanced and innovative fuel designs (STA 5-A1)

6.2.1.1.1 Scope

Fuel behaviour in normal operation, operating transient and accident conditions currently is, and will continue to be, a major issue for the safe, reliable and economic operation of nuclear power plants. An understanding of fuel behaviour is underpinned by fuel R&D, which must address new design and safety requirements, increases in uranium enrichment, uranium and plutonium recycling (and potentially in the future minor actinide recycling), power up-ratings, and increased cycle length and burn-up. It must also address differences in behaviour engendered by the incremental changes in the fuel products. Furthermore, radical innovation could lead to an accident tolerant fuel, and if not fully tolerant of severe accidents, then capable of increased grace time.

Sub-area 5A-1 covers evolutionary improvement of existing nuclear fuel designs and the development of innovative fuels for the longer term horizons with links to Gen IV programmes. Nuclear fuel development is strongly mutually interlinked with sub-area 5A-2 on in-pile fuel behaviour mechanisms and computational codes and sub-area 5A-3 on spent fuel management steps. There are also strong links with the reprocessing investigations in 5-B with regards to the production of recycled nuclear fuel.

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5-A1 has an important interface with TA1 and TA2, which deal with NNP safety and risk assessment (including criticality prevention) and-severe accidents respectively. In addition, 5-A1 also has connections with TA3 in terms of core optimization. 5-A1 is also connected to TA6 regarding fuels for innovative LWRs.

6.2.1.1.2 Objectives

Nuclear fuel production and it use in commercial reactors have reached a relatively matured state. Nevertheless there is motivation to improve existing types and to develop innovative fuel. The main goals of fuel development can be summarised as follows:

- To increase the safety margins of existing fuel with respect to design basis accidents (DBA) and to reduce consequences of severe accidents;
- To innovate towards new accident tolerant fuel (ATF) forms;
- To improve the economics of nuclear power, for example through reduced fuel costs, improved burn-ups and/or power uprates;
- To improve the sustainable use of available fissile and fertile nuclear material resources (uranium, plutonium, thorium) and to reduce fuel cycle greenhouse gas emissions;
- To reduce the amount and/or radiotoxicity of spent nuclear fuel and to recycle as much nuclear material as possible in particular reprocessed uranium, plutonium and minor actinides;
- To improve fuel reliability (reduce fuel failures) under normal operation and operating transient conditions;
- To enhance proliferation resistance.

6.2.1.1.3 State of the art

Uranium dioxide (UO_2) enriched up to 5% in form of solid or annular pellets in zirconium alloy cladding remains the most widely used, studied and reliable fuel in European reactors, primarily LWRs (PWR, BWR and VVER). MOX (mixed uranium-plutonium oxide) fuel also is used in limited quantities mainly in France where large scale reprocessing and manufacturing facilities are available. In the UK, AGRs (advanced gas-cooled reactors) use stainless steel cladding. There is also historical experience with uranium metal fuel elements in earlier Gen I GCRs (gas-cooled reactors) especially the UK Magnox type, which used magnesium alloy cladding (the last operating Magnox reactor in the UK is scheduled to cease operation in 2015).

For LWR fuel assemblies the main construction materials are again zirconium alloys including the channel boxes in BWRs. Nickel alloys and stainless steels also used for some assembly components including rod spacer grids. AGRs use stainless steel and graphite as a moderating sleeve. Control rods are currently manufactured primarily from either silver-indium-cadmium (Ag-In-Cd) alloys or contain boron carbide (B_4C).

For excess reactivity control, Gd and Er oxides are used as absorbers integrated into the fuel matrix or zirconium diboride (ZrB_2) coating is applied to pellet surface. Modifications to fuel microstructures, in particular to achieve larger grain sizes, have also been recently introduced by incorporation of

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additives such as chromia (Cr_2O_3) and alumina (Al_2O_3) or by use of advanced manufacturing techniques such as prolonged sintering.

The properties of all these materials and fuel assembly design are relatively well established; however the drive for continuous improvement in safety, reliability and performance through improved understanding and evolutionary adjustments necessitates further studies and ongoing development.

6.2.1.1.4 Challenges

In spite of the large knowledge base on existing nuclear fuel types, there are still data gaps, which necessitate dedicated material property measurement in separate effect experiments in order to fill them and to thus provide experimental data on fuel performance behaviour, which can be used to inform and validate fuel performance code development.

In addition, a number of evolutionary and innovative nuclear fuel types are undergoing development for which the challenges are much greater. In general, some of the key R&D requirements in order to deploy any new fuel type in existing reactors are:

a) Collection of data on unirradiated material properties;

b) Measurement of in-reactor and post-irradiation examination (PIE) data on fuel performance (thermo-mechanical and thermo-chemical behaviour under irradiation);

c) Simulation and understanding of fuel performance using computational codes (considered under by 5-A2);

d) Assessment of spent fuel (5-A3) and reprocessing/direct disposal suitability (5-B)

e) Development of manufacturing techniques on production scale;

In-reactor and PIE data would in turn be obtained from irradiation testing, initially as rodlets in test reactors, then as lead rods in commercial reactors followed by lead test assemblies (LTAs) and ultimately full core reloads.

Some of the main challenges for nuclear fuel development as summarised as follows:

- Increasing the safety margins of nuclear fuels including the development of new accident tolerant fuel (ATF) forms;
- Improved economics of nuclear fuels in particular through allowing high burn-ups and potential new high density fuel forms;
- Improved MOX fuels including multiple recycled MOX, high plutonium (Pu) content and minor actinide (MA)-bearing MOX;
- Other Pu- and MA-bearing fuels including thorium oxide (ThO_2) matrix (thorium MOX) and inert matrix fuels (IMF) for Pu and MA burning applications.

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For fuel assemblies, although the design is well established, evolutionary modifications and optimisations are under development in both design and material selection in order to increase the performance and improve the reliability. The main challenges are:

- Improved assembly robustness in order to reduce the potential for damage as a result of handling or contact with any foreign objects;
- Improved dimensional stability under irradiation to prevent any hindrance in the insertion of control rods;
- Elimination of any fuel rod damage due to spacer grid to rod fretting;
- Predictive and analytical models are also needed for most of these phenomena, especially for the thermal mechanical behaviour of individual assemblies and across the whole core;
- The increased use of within assembly instrumentation to monitor assembly behaviour during irradiation;
- The development of accident tolerant assembly materials (e.g. silicon carbide composites) and control rods (e.g. Gd_2O_3) as part of ATF development.

Fuel assembly dimensional changes are closely related to the properties of the zirconium alloys used in the design with hydrogen pickup under irradiation often being the crucial parameter.

It should be noted here that current nuclear fuel designs are already highly optimised and achieving an improvement in one area may lead to a detriment in others. Therefore trade-offs will need to be considered with optimisation performed where necessary. Concepts which offer the potential for improvements in multiple areas may need to be prioritised over those which only offer improvement in a single narrower area, however the relative importance of these different areas must also be taken into account.

6.2.1.1.5 New Research Topics

The following research topic areas have been identified to form the main basis of nuclear fuel development for Gen II / III reactors in Europe. It should be noted that these research topic areas will unavoidably have a degree of overlap.

6.2.1.1.6 Accident tolerant fuels (ATF) (TSA 5-A1.1)

The Fukushima accident in 2011 has highlighted some of the shortcomings of current nuclear fuel in LWRs and HWRs. Of particular concern is the potential in steam at accident temperatures for the runaway exothermic oxidation of the zirconium alloy cladding and assembly components, which generates very significant heat and potentially explosive hydrogen gas. For this reason, modification or replacement is sought of the zirconium alloys that are currently used as cladding and assembly materials. The low thermal conductivity of UO_2 is also a concern and hence means to improve the thermal conductivity are being researched as well as the use of alternative higher thermal conductivity fuel compounds. Other means for improving the safety margins in accident conditions are also being investigated such as those associated with geometry and design. In order to have a truly accident tolerant fuel then control rod materials must also be accident tolerant. Of the plethora of ATF fuel concepts that have been suggested since Fukushima, some would offer greater accident tolerance than others whilst some would be more economically favourable than others, and as such it may be advantageous to prioritise concepts with enhanced safety and enhanced economics. In

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addition, some would be easier to deploy than others and have a correspondingly lower R&D requirement. It has been identified that the greatest benefit in accidence tolerance would most likely be achieved through an improvement in cladding technology and hence there is justification for such research to be prioritised from a safety perspective, however concepts relating to the fuel materials should not be overlooked, especially where they can also achieve an economic benefit. Moreover, a successful ATF cladding for uranic fuels could also most likely be applied in an accident tolerant Pu-bearing fuel (see 5A-1.4.4).

6.2.1.1.6.1 Oxidation resistant zirconium alloys (STA 5-A1.1.1)

Enhanced corrosion resistance for zirconium alloys used as cladding and assembly materials is particularly desirable in order to prevent or inhibit the exothermic reaction with steam that was a major contributor to the release of radioactive material during the Fukushima event. New zirconium alloy compositions have been developed and introduced in the past, for example or AREVA's M5 or Westinghouse's ZIRLO as a replacement for Zircaloy-4 in PWRs. Zirconium alloys are already highly optimised and so the potential for improving the oxidation resistance by altering the bulk composition may be limited.

6.2.1.1.6.2 Coating or other surface protection of zirconium alloys (STA 5-A1.1.2)

Instead of altering the bulk composition as discussed in the previous section (see 5-A1.1.1), improved corrosion resistance of zirconium alloys could be achieved through the application of a protective coating or surface treatment. Many potential coatings have been suggested for zirconium alloys including oxidation resistant ceramics such as silicon carbide (see 5-A1.1.5), diamond or MAX phases (see 5-A1.1.6) as well as metallic surface alloy-type coatings. The ceramic coatings may prove more oxidation resistant, however the discrete interface likely with the Zr, and significant differences in thermal expansion rates may mean that these are vulnerable to cracking and spalling from the Zr surface. Metallic coatings may be less oxidation resistant but their surface alloying nature with a gradual change in composition may improve their relative adherence.

The development of coatings is still at an early stage and much R&D must be done before these could be used industrially. A particular concern relates to coating "holidays" where the coating was not applied or has fallen off and whether these could prove to be a weak point and an initiation site of fuel failure during normal operation. In addition, outer metallic (such as steel, molybdenum or chromium – see 5-A1.1.3 and 5-A1.1.4) and/or ceramic tubes (such as silicon carbide composites or MAX phases – see 5-A1.1.5 and 5-A1.1.6) have also been suggested with Zr remaining as a fuel facing liner.

6.2.1.1.6.3 Steels (standard and advanced) (STA 5-A1.1.3)

Post-Fukushima, replacement of zirconium alloys in LWRs by stainless steels to improve high temperature mechanical strength and corrosion resistance during accidents is a possibility, through a reversion of the cladding and assembly materials of earlier version of these reactors. In order to investigate this possibility, it would be prudent to review the available data on historical use of stainless steel clad LWR fuel. However, it should be emphasised that it is likely that significant further work would be required to re-adopt this historical cladding material within a modern reactor safety case.

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The use of steels would incur an economic, fuel enrichment penalty due to the increased neutron absorption, though this could be overcome to an extent by the ability to use thinner walled cladding tube on account of the steels' greater strengths. If the steels are to be re-adopted as the fuel cladding and assembly material, then the issue of tritium release through the cladding must be taken into account. While zirconium alloys are nearly impenetrable to the tritium generated in the fuel, the diffusional release through steel cladding might be more significant and thus further studies are required in this field.

Steels with enhanced corrosion resistance are also desirable and in particular the relatively new class of advanced steels known as FeCrAl (or Kanthal) alloys which form a highly corrosion resistant alumina surface layer. Of all steels, FeCrAl alloys may offer the highest potential benefit versus zirconium alloys in accident conditions. So-called "nano-steels" with a highly refined grain structure have also been considered.

For Gen IV and fusion system, mechanical alloying techniques continue to be developed for the production of oxide dispersion strengthened (ODS) steels, which offer the potential for improved high temperature strength, resistance to creep and corrosion resistance. However, ODS steels are still in the early stages of development and the capability for large scale manufacture is not currently available. They have been suggested for LWR ATF, though it is doubtful whether sufficient benefit could be derived to justify the cost.

Steels, in particular FeCrAl alloys and nano-steels, have also been suggested as a coating for Zr alloys or as an outer protective tube (see 5-A1.1.2).

6.2.1.1.6.4 Refractory metal alloys (STA 5-A1.1.4)

In addition to the steels discussed in the previous section (see 5-A1.1.3), some concepts have also considered the use of refractory metal alloys such as molybdenum (Mo) or chromium (Cr). The refractory metals with the exception of already-used zirconium and to lesser extent vanadium (which is unfortunately has quite poor oxidation resistance) have higher punitive relative neutron absorption. This means that economically they are not viable to use on their own as a direct replacement for zirconium in cladding applications, though they could be used for assembly components such as spacer grids (nickel-based Inconel are already used for these applications). In spite of this, refractory metals (in particular Mo and Cr) have been proposed as a coating or a protective outer tube for Zr alloys (see 5-A1.1.2) or as a coating or a liner for proposed ceramic clads such as silicon carbide composites (see 5-A1.1.5) or MAX phases (see 5-A1.1.6).

6.2.1.1.6.5 Silicon carbide composites (STA 5-A1.1.5)

Silicon carbide (SiC) fibres reinforcing a bulk SiC matrix (SiC-SiC) offers the potential for a low activation, low neutron absorption material capable of withstanding extreme temperatures in excess of 1000°C and high dose irradiation. This potential means that SiC-SiC is being evaluated for Gen IV reactors that operate under these conditions such as the gas-cooled fast reactor (GFR) and the very high temperature reactor (VHTR). The potential benefits of SiC-SiC mean that it is also under consideration as an ATF cladding and assembly material for use in Gen II-III reactors including BWR channel boxes and as an accident tolerant control rod sheath.

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SiC-SiC is still relatively underdeveloped as a material and will require significant additional testing and optimisation in order to realise its potential. Its composite nature means that experience with conventional materials is often not applicable. Important technical hurdles include developing a technique for joining of the end caps (given that welding is not possible), maintaining rod hermeticity (possibly necessitating the use of a metallic liner – see 5-A1.1.2 and 5-A1.1.4) and thermal conductivity under irradiation, and the economic production of long cladding tubes.

6.2.1.1.6.6 MAX phase ceramics (STA 5-A1.1.6)

MAX phase ceramics are a family of carbide and nitride ceramics containing two other elements where:

M = a metal element e.g. titanium or zirconium
A = a metalloid element e.g. aluminium or silicon
X = carbon or nitrogen

There are around fifty known combinations originally defined according to the generic formula $M_{n+1}AX_n$ with titanium silicon carbide (Ti_3SiC_2) and the titanium aluminium carbides (Ti_3AlC_2 or Ti_2AlC) being the most well-known, though more are still being discovered including some 'MAX phase-like' compositions that correspond to other formulae. MAX phases are significantly less brittle than all other known bulk ceramics meaning that their behaviour is closer to that of metals and makes them more suitable as engineering structural materials.

They are capable of withstanding extremely high temperatures (over 1000°C), have high thermal conductivity and have excellent corrosion resistance in many environments. Parasitic neutron absorption is also predicted to be low for some compositions with the potential to increase the commercial competitiveness of reactors and fuel. Low neutron induced radioactivity is predicted for some compositions. Initial irradiation tests have indicated that some compositions may also be resistant to radiation damage but such investigations are at a very early stage.

Correspondingly they are of interest as both bulk material and as a coating for accident tolerant cladding and assembly applications for Gen II-III reactors as fuel as well as Gen IV systems and potentially fusion systems. In addition some of the high neutron absorption phases may be of interest as accident tolerant control rod materials (see 5-A1.1.12).

However, in spite of the promise shown by MAX phase materials, their remarkable properties were only discovered in the 1990s and nuclear applications have only really been suggested since about 2010. As such they are currently at an early stage of development in almost fields including synthesis (in particular the challenge of making single phase materials), shape forming (especially long cladding tubes), joining (similar methods to those proposed for SiC-SiC may be useful), property measurement and modelling. As such significant R&D will be required in order to bring these materials closer to application, in particular further irradiation testing is required to establish whether some compositions truly are radiation resistant as well as high temperature steam testing to confirm accident tolerance.

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6.2.1.1.6.7 Annular pellets in LWRs (STA 5-A1.1.7)

Use of annular UO_2 fuel pellets is routine in the UK's Advanced Gas-cooled Reactors (AGRs) and in Russian designed PWRs (VVERs) as well as some Gen IV fast reactor designs. Due to the potential for an increase of power-to-melt, a decrease of stored energy, and reduced rod internal pressures, wider usage in other reactor systems is being considered. R&D is required to quantify the benefits and the drawbacks, primarily the economic penalty due to the loss of fissile material. Annular MOX fuel (see 5-A1.4.2 and 5-A1.4.4) is of particular interest given that the plutonium content can be readily increased to offset any loss of fissile material, and given that rod pressurisation is usually the limiting phenomenon with respect to fuel licensing.

6.2.1.1.6.8 Dual-cooled fuels (DCF) (STA 5-A1.1.8)

Dual-cooled fuels (DCF) offer similar potential benefits to annular fuel as discussed in the previous section (see 5-A1.1.7). DCF would also use an annular pellet but with an inner diameter cladding tube inserted through the centre to allow passage of the coolant and hence internal cooling of the fuel. The additional cooling of DCF means that the potential gain in safety margins is greater than for annular pellets within a single cladding tube. In addition, DCF could allow for reactor power uprates (see 5-A1.3.1). One slight safety drawback of DCF would be the greater surface area of cladding material within the core that is available to oxidise in accident conditions. Manufacturability at a commercial scale remains a significant issue for such fuel. The effects of differential heat flow through the inner and outer cladding tubes and of differential inner and outer tube deformation are also pertinent issues, alongside the arguably greater potential for fuel failure during normal operation. Dual-cooled MOX fuel is of interest for the same reasons discussed as annular fuel (see 5-A1.1.6, 5-A1.4.2 and 5-A1.4.4). Some DCF inert matrix fuel (IMF) designs have also been proposed (see 5-A1.4.5).

6.2.1.1.6.9 Improved thermal conductivity UO_2 (STA 5-A1.1.9)

Improved thermal conductivity UO_2 fuel would increase safety margins on account of lower fuel centreline temperatures. This would also likely have the effect of improving fission gas retention (see 5-A1.2.2). The main means that has been suggested for improving the thermal conductivity of UO_2 is to dope the fuel with a higher thermal conductivity substance. In particular, it has been identified that doping with short ceramic fibres such as silicon carbide (see 5-A1.1.5) or short metallic wires such as Mo (see 5-A1.1.4) may yield the greatest improvement on thermal conductivity on account of the potential high thermal conductivity paths that could be created within the fuel pellets.

If any of these means was successfully deployed then advanced MOX fuels incorporating similar means could be considered. This is not yet the subject of much investigation, however, so this is noted only briefly in the recycled fuels section (see 5-A1.4.2 and 5-A1.4.4).

6.2.1.1.6.10 Non-oxide high thermal conductivity fuels including composites (STA 5-A1.1.10)

A more radical alternative to improving the thermal conductivity of the fuel is to replace UO_2 either totally or partially (in a 'composite' fuel). Potential higher thermal conductivity uranium compounds

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include uranium metal alloys (especially U-Zr or U-Mo), carbides and nitrides as well as silicide and boride intermetallics.

Many of these compounds also have the significant economic advantage of having a higher uranium density, allowing for a higher power density at the same enrichment level (see 5-A1.3.1). In addition, there would be a slightly increased degree of Pu production in the fuel, not enough to give an overall fissile breeding gain as could be achieved in Gen IV fast reactors, but possibly enough to boost the end-of-life reactivity of the fuel and give a more even power generation rate across the irradiation life, reducing the need for fuel shuffling during shutdown and potentially allowing shorter duration shutdowns. The additional generation of Pu would also increase Pu yield should the fuel be reprocessed and could allow for a greater degree of fuel recycling in LWRs or provide additional seed material for a Gen IV fast reactor closed fuel cycle.

If the significant R&D is undertaken to develop and qualify a higher thermal conductivity accident tolerant fuel, then it would be highly desirable to also gain an economic advantage for doing so. For this reason, the alternative higher thermal conductivity uranium compounds currently targeted for ATF are also those with a higher uranium density than UO_2.

However, these potential alternatives currently also have other downsides with respect to UO_2 which may limit their potential accident tolerance, for example lower melting points than UO_2 (~2800°C) especially the metal alloys (~1100°C) and intermetallics (U_3Si_2 is ~1650°C), though this factor must be traded off against the higher thermal conductivity which will reduce the centreline temperature of the fuel. Uranium carbide, nitride and diboride all have melting temperatures that are similar or not drastically lower than UO_2, though their still some uncertainty regarding the exact values, in particular UN which has been documented as both higher and lower than UO_2. In any case the irradiated melting temperature is more significant in particular the extent to which this is decreased by the generation of plutonium.

In addition, some also have significantly worse reactivity with water which is also an issue in terms of a fuel failure in normal operation; metal alloys and uranium carbide are particularly poor in this regard. Uranium nitride is more promising, historic testing has shown significantly reactivity worse reactivity with water than UO_2, however in recent years some success has been shown in improving its water resistance through using advanced manufacturing techniques such as Spark Plasma Sintering (SPS, see 5-A1.3.5) to achieve oxide coating of the grain boundaries (in particular work at KTH in Sweden).

Of all the proposed higher thermal conductivity/uranium compounds, U_3Si_2 has the best resistance to water currently and is not currently known to be significantly worse than UO_2 and it is currently used in dispersion form in Research and Test Reactors. Irradiation swelling is also a major concern with high density fuels, for example precluding the use of the higher uranium U_3Si.

An issue specific to the potential adoption of uranium nitride is the economics of N-15 enrichment to close to 100%. This is because the majority N-14 isotope is a parasitic neutron absorber that would significantly hinder the in-reactor economics of the fuel and also forms radioactive C-14 which is a waste disposal concern should the fuel be reprocessed. In addition, N-14 would cause additional generation of helium gas within the fuel.

Uranium diboride (UB_2), using depleted boron (B-11) to avoid parasitic neutron absorption, may prove an interesting but currently little studied form for fuel applications (they are more studied for

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wasteform applications). UB_2 has a relatively high melting point (~2400°C vs. ~2800°C for UO_2) is insoluble in water, in addition, it could incorporate B-10 burnable absorber (see 5-A1.3.4) without any chemical alteration. The limited study to date with respect to fuel applications may be as a result of the association of boron in the fuel development community with burnable absorbers neglecting the potential for using the B-11 isotope. B-11 is a current nuclear industry by product and as such is already produced industrially, unlike N-15, the isotope required for uranium nitride fuels. UB_2 has also a slightly better uranium loading density than U_3Si_2. The major downside of UB_2 is a lack of knowledge regarding its neutron irradiation behaviour in particular irradiation swelling which is an issue for other uranium intermetallic compounds.

Of the all potential higher uranium density fuel forms, U_3Si_2 is seen (especially in the USA) as the form which could be deployed most readily in the medium term due to its broadly acceptable reactivity with water, the irradiation experience in Research and Test Reactors and no dependency on the isotopic enrichment of light elements.

In addition, to considering the complete replacement of UO_2 in LWR fuels, composites (mixtures) of higher uranium density fuel compounds with UO_2 or other uranium oxides are also considered to counterbalance their limitations but still yield a net benefit in terms of thermal conductivity and uranium density. Furthermore, composites of higher thermal conductivity / uranium density fuel compounds with no oxide can be considered in order to try and achieve an optimised trade off with respect to their properties and yield a significant thermal conductivity / uranium density benefit.

The reprocessing implications of these new fuel forms should also be discussed. Silicides, in particular, and possibly borides, are likely to prove particularly difficult to reprocess and would primarily be suitable in a once through cycle and their adoption would likely represent a path of no return in terms of partially or fully closed fuel cycles. Reprocessing techniques for uranium alloys such as U-Zr or U-Mo are well established from the experience of Gen I (in particular UK Magnox), historic weapons Pu reactors and Gen IV fast breeder reactor (FBR) prototypes (especially those in the USA where the use of metal fuels was continued for a longer period). The reprocessing of uranium nitride and also uranium carbide can also draw on historic experience and ongoing development in Gen IV programmes, in particular ESNII (European Sustainable Nuclear Industrial Initiative) and EERA JPNM (European Energy Research Alliance Joint Program for Nuclear Material). Nitride fuels are foreseen to be easier to reprocess than carbide fuels due increased solubility in nitric acid, lessening the possible need to first oxidise, though there is complication of N-15 recycling for economic reasons. These considerations are an issue for TA5-B.

Overall, all these fuel forms require extensive additional R&D before any of them could be adopted commercially in particular in the fields of manufacturing technology, fuel-clad chemical interaction (FCCI) and irradiation experience.

If any of these fuel forms was successfully deployed as an accident tolerant uranium-only fuel form then similar accident tolerant plutonium bearing fuel forms could be considered as a replacement for MOX. However, such development is quite a long way off and so this is noted only briefly in the PuMA fuels section (see 5-A1.4.4) and there is not the same drive to increase the fissile loading as for uranium only fuels as this can be potentially be increased by increasing the Pu:U ratio.

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6.2.1.1.6.11 Dispersion and microencapsulated fuels (STA 5-A1.1.11)

Probably the most possible accident tolerant of all proposed fuel material concepts are dispersion fuels where the fissile material (the 'fuel meat') is incorporated within another non-fissile, non-fertile material which drastically reduces the potential for fission product release. This can be either a homogenous dispersion where the two materials are intimately mixed such as Zr-U alloys (e.g. the Lightbridge LWR fuel concept) or a heterogeneous dispersion for example UO_2 particles in either a metal matrix such as a zirconium alloy or a ceramic matrix such as bulk silicon carbide. Metal matrix (Al or Zr) dispersion fuels in a plate geometry have been used for many years in Research and Test Reactors and marine propulsion reactors. Silicon carbide matrix fuel was also used in the UK AGR prototype, WAGR (Windscale Advanced Gas-cooled Reactor).

The role of the matrix material in important in terms of accident tolerance with high thermal conductivity desired throughout the fuel irradiation lifetime (possibly an issue for silicon carbide) and high oxidation resistance in accident conditions (an issue for zirconium alloys).

Dispersion fuels and inert matrix fuels (IMFs, see 5-A1.4.5) share a common characteristic in that a non-fissile, non-fertile, and chemically unreactive matrix is used to host fissile material. The distinction is that dispersion fuels are targeted towards providing provide in-reactor safety and/or performance benefits, whereas IMFs are intended primarily for Pu and/or minor actinide disposition.

Another heterogeneous dispersion concept is so-called microencapsulated fuels where Gen IV High Temperature Reactor-type coated fuel particles are incorporated within similar matrix materials. The coated particles may be of a BISO design incorporating one coating material, likely pyrolytic carbon (PyC) or silicon carbide, or a TRISO design incorporating both. Zirconium carbide is also suggested as a replacement for silicon carbide. Microencapsulated fuels would potentially prove even more accident tolerant than a simple dispersion. Oak Ridge National Laboratory (ORNL) in the United States has been particular proponents of microencapsulated fuels.

However, such potential high accident tolerance would come at a significant economic penalty in terms of the enrichment of the fuel meat, with a greater enrichment penalty for a lower fuel meat to matrix ratio possibly even pushing enrichment beyond the 20% lower enrichment limit of the High Enriched Uranium (HEU) classification. It should be noted that the enrichment penalty may be able to be compensated to an extent by the potential to achieve higher burn-up (see 5-A1.3.2 regarding the significance of higher burn-up).

In addition to the economic issues, there are also potential proliferation resistance and security issues associated with dispersion and microencapsulated fuels in that this would require the shipping of fuels of significantly higher enrichments than the current Gen II-III limit of 5%, a regulatory mandation in some countries and an industry standard in others (see 5-A1.3.6 for issues concerning higher enrichment fuels). It would also be against the trend of fuel development for Research and Test Reactors where the US-led IAEA RERTR (Reduced Enrichment in Research and Test Reactors) has been endeavouring to reduce the enrichment of RTR fuel and radioisotope production targets. It can be argued that given the dispersion form, the fissile material is difficult to extract however this argument does not seem to be currently holding much sway in the RTR community.

Finally the use of dispersion and microencapsulated fuels would primarily be suitable in a once through cycle as these forms are likely to prove highly difficult to reprocess and their adoption would likely represent a path of no return in terms of partially or fully closed fuel cycles.

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6.2.1.1.6.12 Accident tolerant control rod materials (STA 5-A1.1.12)

In order to have a truly accident tolerant fuel then control rod materials must also be accident tolerant. In particular it is desirable to replace Ag-In-Cd alloys which have a relatively low melting point at ~800°C, though with a high thermal conductivity, are desired to be replaced. Boron carbide which is currently used is refractory (~2750°C). Other higher melting temperature control rod material options include dysprosium titanate (~1850°C) for which there is Russian experience, gadolinium oxide (~2400°C) which is already used as a burnable absorber (see 5-A1.3.4), hafnium-containing compounds including hafnium metal (~2250°C), hafnium diboride (3250°C) and hafnium-based MAX phase ceramics (see 5-A1.1.6) which are amongst the most refractory of all materials. Tantalum containing MAX ceramics may also be an option. One of the significant issues with control rod options containing more unusual elements is the cost of these rare species this is particularly true for hafnium which has been used in control rods in marine propulsion systems but is potentially prohibitively expensive for commercial reactors.

6.2.1.1.6.13 Other ATF concepts (STA 5-A1.1.13)

Since the Fukushima accident in 2011, a wide variety of new ATF concepts have been postulated. The most significant of these are summarised in the preceding sub-sections. In addition, there are others that have been postulated and may come to greater prominence in future years. These often have a very limited level of development such that the technology readiness which is currently too low to warrant a sub-section in its own right. One example is the category of materials known either as 'amorphous metals' or 'metallic glasses' such as those based on the zirconium-vanadium (Zr-V) system which has been suggested for ATF. The atoms in these materials are metallically bonded but do not have the crystalline structures traditionally associated with metals. These relatively poorly understood materials do show evidence of some remarkable properties but much greater knowledge is needed on their behaviour. Another possibility is the development of other composite materials such as those incorporating zirconium carbide, super strong silica fibres or even graphene.

Furthermore it seems likely that other new concepts will be postulated that are not currently featured in this roadmap. Such concepts should not be overlooked for further development without any consideration but should be assessed on their respective merits alongside the more established concepts. New concepts are perhaps more likely in terms of cladding materials than fuel materials, given the greater flexibility in the potential elemental compositions that can be considered, and the greater likelihood of these being developed for non-nuclear applications.

6.2.1.1.7 Improved safety and reliability in fuels during normal operation (STA 5-A1.2)

In addition to greater tolerance to accident conditions as discussed in 5-A1.1, it is also essential to improve safety during normal operation and to eliminate any potential accident initiating factors. In terms of the fuel, the main potential accident initiator to consider in this section is the distortion of the control rods or the assembly guide tubes. For improved safety, the main focus is to reduce the already very low rates of fuel failure (clad rupture leading to the release of radioactive fission products into the primary coolant), to as close to zero as possible with the target of complete elimination. This will reduce doses for reactor operators and those carrying out decommissioning,

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reduce radioactive discharges and significantly reduce the hazard and complication associated with storage or reprocessing of failed fuel. The measures taken to eliminate fuel failures are in general more evolutionary in nature compared with some of the more innovative ATF concepts discussed in the previous section (see 5-A1.1). It should be noted that a number of the measures that may be taken to prevent fuel failure, may also allow fuel to be taken to higher burn-up (see 5-A1.3.2).

6.2.1.1.7.1 Greater margins to pellet-clad interaction (PCI) (STA 5-A1.2.1)

Pellet clad interaction (PCI) has the potential to cause fuel failure through the swelling UO_2 pellet contacting and causing rupture of the cladding. Therefore is it is highly desirable for there to be greater margins before the potential occurrence of PCI. Greater margins to PCI may also allow fuel to be taken to higher burn-up (see 5-A1.3.2). One of the main means to achieve this is to enhance material creep rates within the fuel. It is also advisable to better understand the role of thermo-chemistry in PCI phenomena.

6.2.1.1.7.2 Improved fission gas retention including grain size dopants (STA 5-A1.2.2)

Excessive rod internal pressure caused by the build of fission gas released from the fuel matrix can cause fuel failure. Therefore it is desirable to increase the proportion of fission gas retained within the structure of the fuel. In addition, this is desirable as it minimises the potential concentration of radioactive fission products released in the event of a fuel failure. Improved fission gas retention (FGR) may also allow for fuel to be taken to higher burn-up (see 5-A1.3.2).

Improved fission gas retention is most commonly achieved by increasing the fuel grain size through the use of dopants. The two predominant dopants which have now reached commercial deployment are chromia (Cr_2O_3) and alumina (Al_2O_3) in particular in AREVA's chromia-doped and Westinghouse's ADOPT fuel (alumina and chromia-doped).

Other dopants that have investigated extensively include niobia (Nb_2O_5), magnesia (MgO), titania (TiO_2) and silica (SiO_2). There is interest in the use of beryllia (BeO) as a dopant, however handling issues are major concern as beryllium is strongly carcinogenic. Alumino-silicates and phosphorus are also considered as dopants.

Improved fission gas retention could also be achieved by improving the thermal conductivity of the fuel and thus reducing fuel centreline temperatures (see 5-A1.1.9) or by advanced manufacturing techniques such as large grain or spark plasma sintering (see 5-A1.3.5).

6.2.1.1.7.3 Improved resistance to clad corrosion (STA 5-A1.2.3)

Corrosion of either the inner or the outer clad surface can be caused in normal operation by oxidation or by hydriding, in particular in the case of zirconium alloys. Such corrosion can initiate a fuel failure and therefore it is desirable to reduce the potential for zirconium alloys to be corroded in normal operation. This could be achieved either by developing new Zr alloys compositions as has been achieved in the past or by surface treating (coating) the Zr alloys. These are also the means proposed for improving the tolerance of Zr alloys to accident conditions as discussed in the previous ATF section (see 5-A1.1).

6.2.1.1.7.4 Reduced crud formation (STA 5-A1.2.4)

Crud formation in LWRs is related to water chemistry and corrosion of primary circuit components during normal operation and during decontamination procedures. The accumulation of deposits on

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the fuel rod surface can increase the core pressure drop, change the axial power profile and limit the core power (an economic issue). The presence of crud can also accelerate clad oxidation and hydriding, with the potential for loss of clad integrity resulting in a fuel failure. The specific factors which lead to crud formation (reactor system, primary circuit materials, coolant chemistry, coolant mass evaporation rate etc.), crud alterations, and the resulting mechanisms of clad oxidation and hydriding behaviour need greater understanding through further investigation and simulation (see 5-A2).

In the UK AGRs, crud can also be created in the form of carbonaceous deposits on the surface of the stainless steel cladding caused by reaction with the CO_2 coolant and/or carbon from the graphite moderator bricks and assembly sleeves. These deposits degrade the heat transfer properties of the fuel and increase the potential for fuel failure. One means by which may be reduced is by pre-oxidation of the cladding to thicken the natural surface oxide. Such a means is currently being trialled in lead test assemblies in a number of UK AGRs.

6.2.1.1.7.5 Improved assembly and control rod robustness (STA 5-A1.2.5)

Improved assembly and control rod robustness is desirable in order to reduce the potential for damage as a result of handling or contact with any foreign objects during manufacture, transport and reactor operations. Any such damage has the potential to be the initiating point for a fuel failure. More important any damage to or distortion of the assembly, the control rods or their guide tubes has the potential to hinder the insertion of the control rods which could be a potential accident initiator. Approved robustness could either by achieved by improved manufacturing techniques, improved design likely in a trade-off situation with the improved economics assembly designs considered in 5-A1.3.3 or through the use of improved or novel materials such as those discussed previously in the ATF section (see 5-A1.1).

6.2.1.1.7.6 Reduced grid-to-rod fretting (STA 5-A1.2.6)

Fretting between the rods and the spacer grids during insertion or during operation (in particular as a result of vibration) can cause fuel failure and thus it is desired to reduce the potential for damage to be caused in this way. This can be reduced by improved design, improved manufacturing and/or improved materials. In particular fretting has been reduced in the past by improved spacer grid designs and manufacturing in particular those incorporating fewer welds. More recently, 3D printing technology is being investigated for its potential to produce near net-shape formed grids with minimal machining and consistent, homogenous material. Such additive manufacturing techniques may also allow for new designs in new geometries which are not possible to produce by conventional line-of-sight methods.

6.2.1.1.7.7 Improved assembly instrumentation (STA 5-A1.2.7)

The increased use of within assembly instrumentation to monitor assembly behaviour during irradiation is desirable from a safety perspective during normal operation for example for its potential to give advanced warning of fuel rod becoming vulnerable to failure, in order to allow

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corrective action to be taken. It would also prove very valuable as a source of data for the validation of assembly scale predictive models and their application as part of whole core models.

6.2.1.1.7.8 Improved safety during fuel manufacturing (STA 5-A1.2.8)

Whilst UO_2 fuel manufacturing techniques are highly mature and safely produce thousands of tonnes of nuclear fuel every year. There areas in which safety could be improved, in particular the following are desired: reduction in the use of hazardous chemicals (such as hydrofluoric acid), less hazardous sintering atmospheres (lower levels of hydrogen) and reduced generation of grinding dust. The latter two issues are also areas where there is an even greater desire for improvement in the manufacture of MOX fuel (see 5-A1.4.3).

6.2.1.1.7.9 Reduced activation of clad and assembly materials (STA 5-A1.2.9)

In general, the activation of clad and assembly materials is not a significant problem, if the spent fuel assemblies are for direct disposal as the radioactivity of the spent fuel itself dominates. Reduced activation however is desirable in the case of reprocessing as the cladding and assembly materials will be disposed of separately. This would be primarily through reducing the activation of zirconium alloys by the reduction or replacement of niobium in the composition.

As a side note, reduced activation steels continue to be developed with the replacement of alloying elements such as molybdenum and niobium leading to development of grades such as Eurofer and reduced activation ferritic/martensitic (RAFM) steels. However, the reduced activation of steels is primarily desired for structural materials and development has been led by fusion programmes where this is a greater concern. They are not particularly relevant for LWR fuel development unless steels are adopted as accident tolerant fuel (see 5-A1.1.3).

6.2.1.1.8 Enhanced economic fuels (STA 5-A1.3)

In addition, to safety improvements, the nuclear industry is constantly seeking for economic improvements which can increase the competiveness of nuclear power, especially with respect to fossil fuels as a base load power source, and hence assist with lowering greenhouse gas emissions. In addition to improved economics for the individual companies involved, this is desirable in order to contribute to overall economic growth in Europe. Such economic improvements would need to be at least safety neutral in order to satisfy regulators and it may be advantageous if advanced LWR fuel concepts which combine enhanced safety and enhanced economics are prioritised.

6.2.1.1.8.1 High power density fuels (STA 5-A1.3.1)

Fuels capable of sustaining high power densities are desirable in order to allow power uprates for reactors, increasing the volume of electricity generation per reactor. Alternatively, high power density fuels can allow for greater safety margins at the same power. As such most of the potential options for high power density have already been discussed in the previous accident tolerant fuel section (see 5-A1.1), in particular the sub-sections on annular pellets (see 5-A1.1.7), dual-cooled fuels (see 5-A1.1.8) and non-oxide high thermal conductivity fuels (see 5-A1.1.10). Some potential

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advanced cladding materials which would allow for reduced neutron absorption such as silicon carbide composites (see 5-A1.1.5) and MAX phase ceramics (see 5-A1.1.6) could also allow for a higher power density. The only means not currently discussed in previous sections is a reduction in fuel porosity of UO_2, thereby giving a higher density of fissile atoms with a density closer to the theoretical maximum, which could also have the added benefit of improving fission gas retention. This could be achieved through the deployment of dopants such as chromia or alumina (see 5-A1.2.2) or enhanced manufacturing techniques, for example Spark Plasma Sintering (see 5-A1.3.5).

6.2.1.1.8.2 High burn-up fuels (STA 5-A1.3.2)

In spite of the large knowledge base for UO_2 in zirconium alloy cladding, there are still unknowns, necessitating dedicated material properties, separate effects, semi-integral and integral testing to provide experimental data on fuel performance behaviour. These data can then be used to inform fuel development and to elucidate key mechanisms to improve understanding and simulation of fuel performance (the latter is discussed in 5-A2). In particular, the continual drive towards higher burn-ups requires high burn-up data, which can be used to develop fuel licensing methodologies to take account of high burn-up effects such as the so-called 'high burn-up structure which develops'. In addition, the improved use of burnable absorbers can facilitate higher burn-up (see 5-A1.3.4) and/or an improvement in fission gas retention, for example by using dopants (see 5-A1.2.2) or enhanced manufacturing techniques (see 5-A1.3.5).

Operation to high burn-up is extremely important in that it reduces both reactor operating costs and the amount of spent fuel per unit of energy generated, and is therefore more sustainable. Experimental data on fuel performance during both normal operation and accident conditions is imperative to ensure fuel safety. In particular, the fuel must be able to withstand limiting design basis accidents, which are generally RIAs (Reactivity Initiated Accidents) and LOCAs (Loss Of Cooling Accidents). Research is ongoing in the OECD/NEA Cabri Water Loop Project and the OECD/NEA Halden Reactor Project (HRP), for example, to set new safety criteria for these two scenarios that extend to high burn-up. PIE (Post Irradiation Examination) data to make the link between the microstructural changes of the fuel and the integral behaviour are indispensable.

6.2.1.1.8.3 Advanced economics assembly designs (STA 5-A1.3.3)

For fuel assemblies, although the design is well established, evolutionary modifications and optimisations are under development in both design and material selection in order to increase the performance and/or burn-up (see 5-A1.3.2).

6.2.1.1.8.4 Advanced burnable absorbers (STA 5-A1.3.4)

The increase of fuel burn-up (see 5-A1.3.2) leads to extended utilisation of burnable absorbers (also known as burnable poisons) in UO_2 fuel, most commonly gadolinia (Gd_2O_3) or erbia (Er_2O_3) incorporated within the fuel or zirconium diboride (ZrB_2) coating is applied to pellet surface. In particular, the incorporation of burnable absorbers into the fuel pellets has effects on the fuel material properties (notably thermal conductivity and fission gas conductivity) and the fuel performance (notably the densification). Decoupling the effects of the power depression from the

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intrinsic effects of the burnable absorber material, and quantifying the effects at high burnable absorber loadings associated with extended burn-ups, require further experimental data as well as the greater use of modelling (see 5-A2) and associated validation. Alternative absorbers have also been considered such as dysprosium (Dy) and europium (Eu). The use of burnable absorbers in MOX fuel is discussed in 5-A1.4.2. One potential unique burnable absorber isotope for reactivity control is thorium which is discussed in 5-A1.5.

6.2.1.1.8.5 Advanced UO_2 manufacturing (STA 5-A1.3.5)

Higher throughput and reduced rejection rates are desired in any manufacturing plant, as is increasing the use of recycled material, and whilst improvement in these fields is constantly being achieved, it should remain a priority for this to continue. Similar issues apply to the manufacturing of MOX fuels (see 5-A1.4.3)

The grain size is one of the crucial properties of oxide fuel pellets. Increasing the fuel grain size has been one of the principal means for increasing the fission gas retention of the fuel with its associated safety benefits (see 5-A1.2.2). This can be achieved through the use of dopants (also see 5-A1.2.2) or alternatively through the use of advanced manufacturing techniques such as long sintering times or oxidative rather than reductive sintering, which has the additional safety benefit of removing hydrogen from the sintering atmosphere (see 5-A1.2.8).

Conversely rapid sintering techniques such as spark plasma sintering (SPS) where a high electric field is passed through the pellet via conductive dies to achieve a very small grain size with very minimal porosity, which has associated economic benefits (see 5-A1.3.1). In addition SPS, has the potential to achieve a near net shaped formed pellet, which could simplify the overall manufacturing process and reduce the generation of grinding dust along with the associated safety benefit (see 5-A1.2.8). This has the potential to be an even greater benefit for higher radioactivity fuels in particular MOX (see 5-A1.4.3).

Significant additional R&D is needed for fuels made by these advanced manufacturing techniques. In particular, data for material properties and the behaviour under irradiation are sparse for large grain sintering and are not currently believed to exist for SPS fuel.

For SPS in particular, there two significant issues that must be addressed. Firstly, throughput needs to be improved through the design of advanced machines, likely based multiple punch or rotary presses, for which links to other industries which may be quicker to adopt the technology should be sought. Secondly, the fuel pellets currently get contaminated with a thin outer diffusion layer of carbon as result of contact in sintering with graphite dies, which are the currently favoured material. This layer can be removed by grinding but this would negate one of the main potential benefits of SPS. Current options for avoiding carbon contamination include alternative die materials such as refractory metals and silicon carbide or carbon-carbon composites as well as post sintering furnace atmosphere treatments.

6.2.1.1.8.6 Higher enrichment fuels (STA 5-A1.3.6)

The enrichment of commercial UO_2 fuel is currently limited to < 5%. This is in order to satisfy the safety requirements associated with fuel manufacture and handling facilities and with fuel transport,

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and is in many cases inherent in the design of facilities and/or transport containers. The result is that achievable fuel burn-ups are limited.

The introduction of fuel with higher enrichment may need new, or modified, facilities and/or transport containers to prevent criticality during the fuel fabrication, handling and transport. From a fabrication perspective, criticality control issues will determine equipment geometry and will require a re-evaluation of process flow sheets with respect to scale down and performance for reduced scale operations. Dose control for operations is a key issue. Proliferation resistance will also require evaluation.

Since the fuel enrichment does not significantly affect the chemical behaviour of the fuel or its material properties, the need for experimental testing of high enrichment rods may be limited (given also that experimental rods with high enrichment fuel are routinely irradiated in test reactors). However, the effects on the radial power profile, on formation of high burn-up structure, and the general effects of high power / high burn-up operation would require further investigation.

In addition, much work may need to be performed on a regulatory basis to permit the manufacture, shipping and irradiation of higher enrichment fuels.

6.2.1.1.9 Recycled fuels (STA 5-A1.4)

PUREX-type reprocessing extracts both uranium and plutonium and separates them from fission product and minor actinide wastes. In Europe, in addition to fuels containing uranium enriched from ore, recycled fuels bearing reprocessed uranium (Rep U) and plutonium (Pu) have been used, continue to be used and are to be planned to be used into the long term future. In Gen II-III reactors to date, in addition to UO_2, these fuels have also included mixed uranium-plutonium oxide (MOX) fuels in LWRs. In addition, in the future, it may be desirable in order to facilitate their partial destruction, to incorporate a proportion of the minor actinide (MAs) elements americium (Am), neptunium (Np) and curium (Cm) into recycled fuels for LWRs, in addition to interest in using them in Gen IV fast reactor fuels to facilitate a more complete destruction. Potential research topics for the use of recycled fuels in Gen II/III reactors are detailed below. Rep U is discussed in 5-A1.4.1, whilst Pu-bearing fuels are discussed in 5-A1.4.2 to 5-A1.4.5 and minor actinides in 5-A1.4.6.

6.2.1.1.9.1 Reprocessed uranium (Rep U) fuels (STA 5-A1.4.1)

Rep U has a residual U-235 content equivalent to around 0.9% enrichment. In addition, it contains other less common uranium isotopes in higher proportions than natural uranium such as U-236 and traces of U-232 which slightly increase the dose associated with its use compared with the equivalent material enriched from ore and cause slight complications (reactor physics originated) for its use in recycled fuel.

The primary option for Rep U use in recycled fuel is currently re-enrichment for use in LWR fuel with careful management of dose and reactor physics issues.

In addition, there is another potential option for Rep U use in Gen II/III reactors, which is its direct use in Heavy Water Reactors (HWRs) without re-enrichment. As a result of their increased moderation, HWRs can fission fuel at this low level of residual enrichment (0.9%) or even as pseudo-natural uranium (0.7%) by blending with depleted U. Rep U could also be used as the uranic component of HWR MOX fuels (see 5-A1.4.2).

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However, the opportunity to exploit this option in Europe is currently limited, with only two fully commercial HWR reactors currently operating, the CANDU (CANada Deuterium Uranium)-type Cernavoda-1 and -2 reactors in Romania, with three other partially built CANDU reactors at the same station of which two currently seem likely to be completed. CANDU reactors have also been postulated for the UK in recent years in addition to other new build proposals. It should be noted that a number of other semi-commercial HWR prototypes also operated in the past in Europe, most notably the ~100MW SGHWR (Steam Generating HWR) at Winfrith in the UK which operated for over 20 years until the early 1990s, originally accompanying plans for commercial units which were ultimately cancelled.

In addition, the possibility exists for the export of Rep U fuels produced in Europe to HWRs in other parts of the world such as Canada, where the predominant CANDU HWR technology was developed, as well as CANDU and CANDU-derived reactors in India and China.

6.2.1.1.9.2 Advanced MOX fuels (STA 5-A1.4.2)

Like UO_2 fuel, $(U, Pu)O_2$ mixed oxide (MOX) fuel has reached a high degree of maturity, albeit not as high as for UO_2. In spite this level of experience, there are still unknowns, necessitating dedicated material properties, separate effects, semi-integral and integral testing to provide experimental data on fuel performance behaviour which can be used to validate models (see 5-A2). In particular, experimental data is required to better quantify the effects of UO_2/PuO_2 heterogeneity and of high Pu content (both on material properties and on fuel performance), and data to enable extension of MOX operation to higher burn-up are needed. In addition, helium generation in, and release from, the fuel pellets is not well understood, and can provide a significant contribution to the pressurisation of the fuel rods. This applies equally in-pile, during interim storage, and after final disposal. The helium generation also raises the possibility of degradation of the fuel matrix over medium to long post-irradiation timescales (covered by 5-A3).

Burnable absorber materials are generally incorporated into the fuel pellets in MOX cores to improve reactivity control. However, since existing cores containing MOX fuel have only around one third of the core occupied by MOX assemblies, with the remaining core locations occupied by UO_2 assemblies, the burnable absorber material can be incorporated into the UO_2 pellets without the need for incorporation into the MOX pellets. This is clearly not possible for a 100% MOX core as proposed for some Gen III+ reactors, and may also not be feasible above a certain MOX core fraction threshold. Since higher MOX core fractions are desired for increasing the throughput of plutonium in order to reduce stockpiles, development of burnable absorber bearing MOX pellets is attractive. In addition to the general fuel R&D requirements described above, core neutronics studies would be needed to investigate the inherent feasibility, and any limitations, of this approach.

Annular or dual-cooled MOX fuel is of particular interest given that the plutonium content can be readily increased to offset any loss of fissile material, and given that rod pressurisation is usually the limiting phenomenon with respect to fuel licensing. Both concepts would give the potential for increased safety margins or possibly improved economics whilst maintaining the same safety margins (see 5-A1.1.7, 5-A1.1.8 and 5-A1.3.1). In addition, the methods for improving UO_2 thermal conductivity described in 5-A1.1.9 to improve accident tolerance (SiC or Mo doping) could potentially be applied to MOX fuels.

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A further type of advanced MOX fuels, though currently of limited development, would be MOX fuels for HWRs. The reactor situation regarding HWRs in Europe and the use of Rep U fuels in HWRs is discussed in 5-A1.4.1. HWR MOX fuels are also known as CANMOX fuels on account of their applicability to the predominant CANDU designs of HWR. HWR MOX fuels would have a much lower Pu content than standard LWR fuels, ~3% compared to ~10%. This would have the advantage of potentially allowing easier fabrication, though an industrial plant is not yet developed, for example less restrictive criticality control and reduced operator dose per tonne of MOX fuel produced. In addition, there is the potential to accept a more relaxed PuO_2 feedstock specification given the increased degree of blending with UO_2 which could reduce the volume of high Pu content residues requiring immobilisation and disposal (covered in TA5-B). It would also allow the possibility of using Rep U (see 5-A1.4.1) directly as the uranic component or directly using the products of an advanced, proliferation resistant COEX (co-extracted U and Pu) reprocessing methodology (reprocessing is covered by TA5-B), or even fractured and unreprocessed LWR fuel which is then re-pelleted or vibro-packed in the so-called DUPIC (Direct Use of PWR In CANDU) fuel cycle concept. However, the disadvantages of HWR MOX would be the much larger volume of fuel (~3x greater) that needs to be fabricated and shipped compared with LWR MOX to re-use the same volume of Pu and the corresponding much larger quantity of spent fuel.

Advanced MOX fuels that are also of interest include those that use unconventional Pu feedstocks which have correspondingly unusual isotopics (Pu vector). These include ex-weapons Pu, ultimately drawing on the experience being gained in the US-Russian 'Megatons to Megawatts' programme especially from US LWR MOX, though overall this is less of an issue in Europe. In addition, the use of long-stored Pu civil stocks is desired which have a relatively high Am content, which is particularly a challenge for operator dose during manufacturing (see 5-A1.4.3). Multiple recycle of Pu is also desired through the reprocessing of MOX fuels and presents similar challenges. Deliberate minor actinide additions to the fuel are discussed in 5-A1.4.6. Thorium-plutonium MOX fuels are discussed in 5-A1.5.

6.2.1.1.9.3 Advanced MOX manufacturing (STA 5-A1.4.3)

Advanced MOX manufacturing focuses on a number of key areas: reduced operator dose, less hazardous sintering atmospheres, reduced grinder dust, ability to accept a wider variety of feedstocks, higher throughput, reduced pellet rejection rates, increased levels of recycled material incorporation (addback) more reliable operations and more reliable commissioning of new plants.

Operator dose can principally be reduced through a greater degree of automation and shielding within plants, though this will have consequences for the level of access available in order to perform maintenance and hence plant reliability must be improved simultaneously.

Less hazardous sintering atmospheres principally refers to reduction in the levels of hydrogen. Reduced grinding dust can be achieved by increasing the degree to which the pellets are near net shape formed (reduced wheatsheafing) or by reducing scrap pellets rates with the corresponding reduction in the volume of unnecessary grinding. Both these areas link to similar issues with UO_2 (see 5-A1.2.8), but for MOX fuels the potential hazards are greater (though fuel volumes are lower). Spark plasma sintering has been identified as one advanced manufacturing technique that could significantly reduce the need for grinding through its potential for near net shape forming (see 5-A1.3.5).

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In terms of accepting a wider variety of feedstocks, then in addition to ex-weapons Pu, this includes older or other more problematic PuO_2 powders, which may otherwise have to be converted into a wasteform (see TA5-B). These will have high Am in-growth in the case of older material (with the associated dose issues during manufacturing), and may suffer from other issues such as high levels of impurities (for example gallium in ex-weapons Pu) or a low SSA (specific surface area), all of which may be exacerbated by limited characterisation in the case of older stocks. Multiple recycle of Pu is also desired through the reprocessing of MOX fuels and presents similar challenges to older and problematic materials.

The relaxation of MOX fuel specifications is very difficult to justify as this would necessitate straying outside of the current envelope of safe operating experience and would hence significantly increase the costs and timescales of fuel qualification. Blending strategies may be deployed up to a point to deal with some of these issues. However, if this cannot be achieved, then issues, such as high impurities and low SSA, may be resolved by: remediating pre-treatments in order to meet existing powder feed specifications; or alterations to current manufacturing processes to allow relaxation of powder feed specifications, whilst still meeting current as-manufactured MOX fuel specifications. The potential advantages and disadvantages of manufacturing HWR MOX (CANMOX) are discussed in 5-A1.4.2 including the potential for a relaxed PuO_2 feed specification.

Higher throughput and reduced rejection rates are desired in any manufacturing plant, as is increasing the use of recycled material, and these are also issues for UO_2 (see 5-A1.3.5). In the case of a MOX plant, then this may allow historic MOX residues or unirradiated MOX fuels to also be recycled.

Finally, a number of relatively new MOX plants have suffered difficulties in achieving reliable commissioning and in achieving a high throughput once commissioned. This situation must be addressed in order to successfully bring new MOX plants on line and to allow a high degree of recycling in the current Gen II/III fuel cycle.

6.2.1.1.9.4 Accident tolerant Pu-bearing fuels (STA 5-A1.4.4)

An ATF cladding suitable for uranic fuels (see 5-A1.1.1 to 5-A1.1.6) would most likely also be suitable for Pu-bearing fuels providing that the presence of Pu did not cause a significant adverse reaction. In addition, some more evolutionary methods of improving the accident tolerance of MOX fuels discussed in 5-A1.4.2 (annular pellets, DCF and thermal conductivity doping). The accident tolerance of plutonium-bearing LWR/HWR fuels could be improved by investigating alternative plutonium compounds or composites with higher thermal conductivity similar in form to those discussed for uranium-only fuels discussed in 5-A1.1.10, except the presence of plutonium will act to significantly depress melting points such that U-Pu-Zr or U-Pu-Mo alloys and even silicides may not even be potentially viable. Mixed U-Pu nitride and carbide (MX) fuels may be the leading candidates given that there is past and current experience of the manufacture and use of these fuels for Gen IV prototypes, though the water reactivity and N-15 issues discussed in 5-A1.1.10 will still apply. The potential economic benefits of these high power density fuel forms as discussed in 5-A1.1.10 and 5-A1.3.1 will also still apply.

Overall such concepts are much less developed than the uranium-only concepts and it is not likely that significant further practical research effort will be invested in them until one or more of the

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uranium concepts are brought much closer to deployment. However, the potential similar Pu-bearing concepts should not overlooked entirely. At this stage, the most prudent course of action may be to gather relevant literature, in particular on MX fuels from fast reactor programmes, linking with these programmes in the process and to assess where major data gaps currently exist.

6.2.1.1.9.5 Inert matrix fuels (IMF) (STA 5-A1.4.5)

Inert matrix fuels and dispersion fuels (including microencapsulated, see 5-A1.1.11) and dispersion fuels share a common characteristic in that a non-fissile, non-fertile, and chemically unreactive matrix is used to host fissile material. The distinction is that dispersion fuels are targeted towards providing in-reactor safety and/or performance benefits, whereas IMFs are intended primarily for Pu and/or minor actinide (MA) disposition (see 5-A1.4.6). The recent emphasis of work programmes on IMF has been on the burning of plutonium to reduce inventories, without creating additional fissile material.

Like dispersion fuels, IMFs can be homogeneous (solid solution) or heterogeneous (two phase). Two phase IMFs can either be duplex metal or duplex ceramic alloys (duplex alloys naturally form two phases), or a composite can be constructed i.e. cercers (ceramic-ceramic), cermets (ceramic-metal) or metmets (metal-metal).

A huge number of potential matrices can be considered including metal alloys, oxides, carbides, nitrides etc. IMFs targeting the geological disposal of spent fuel and/or high fuel temperature resistance in reactors are primarily considering ceramic, mineral-based, 'rock-like' oxide matrices based on zirconia (ZrO_2), which also have low neutron absorption. Higher thermal conductivity nitride and carbide ceramics (such as SiC, ZrC or ZrN) or metal alloys (such as Zr) may be considered for increased thermal safety margins in the fuel and/or high power usage.

To date, R&D on IMF has included potential use in most Gen II, III and IV reactor types. The choice of matrix and the Pu and MA concentrations vary with the intended reactor type. MA IMFs (see 5-A1.4.6) have been primarily targeted at Gen IV fast reactors as MAs can burned to a greater extent in the hard neutron spectrum; however there could be some interest for LWRs, in combination with advanced assembly designs allowing for an under-moderated, harder, epithermal spectrum in certain locations in the core. In addition, there could be interest in MA IMFs for innovative LWR designs which would operate entirely or partially with an epithermal spectrum (see 5-A1.6 and TA6).

The fuel cycle strategy for IMF also needs to be considered, in particular whether the IMF is to be reprocessed or sent for final disposal as a wasteform (the latter has had the greater focus); and the relative importance of various underlying factors such as manufacturability, material properties, in-reactor performance, proliferation resistance and suitability for final disposal.

The IMF development process has generally been: (1) desktop studies; (2) manufacturing trials with surrogate materials for Pu and MAs; (3) out-of-reactor ion beam testing; (4) manufacturing trials with Pu; (5) irradiation of Pu-bearing fuel rods in a test reactor. Steps 4 and 5 have then been repeated with various Pu or MA loadings where appropriate. Each of the many IMF variants has reached different stages in this process. However, to date, no IMF assemblies have been irradiated in commercial reactors.

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IMF R&D has been performed in the context of national programmes (France, Switzerland, Canada, Japan, Russia, and the USA) and multilateral collaborations (e.g. OTTO, BORA-BORA, OECD/NEA Halden Reactor Project) and European and other international projects such as EFTTRA, FUTURE, CONFIRM, EUROTRANS, ACSEPT, FAIRFUELS and ASGARD.

For LWR/HWR R&D, oxide-based inert matrices have been preferred, although SiC, and more recently ZrC, have also been studied in some detail, partially due to higher thermal conductivity and experience with coated particle fuels. The oxide-based matrices include ZrO_2-Y_2O_3 (YSZ), ZrO_2-CaO, ZrO_2-MgO, corundum (Al_2O_3), CeO_2, zircon ($ZrSiO_4$) and spinel $MgAl_2O_4$. Most, if not all, have been irradiated. In particular, YSZ and ZrO_2-CaO fuels featured in the Halden experiments IFA-651 and IFA-652. Al_2O_3 and CeO_2 are the reference inert matrices for the French PLUTON and APA assembly designs, respectively. It should be noted that some versions of the APA design utilise large DCF rods for the IMF (see 5-A1.1.8).

Overall, further IMF R&D is required to optimise material composition (including the choice of inert matrix material), to demonstrate manufacturability at a commercial scale, to ensure safety under transient conditions, and to confirm satisfactory performance at high burn-up and the stability of the spent fuel as a wasteform (links to 5A-3 and TA5-B). The neutronics of a core wholly or partly fuelled with IMF also needs further assessment.

6.2.1.1.9.6 Minor actinide (MA)-bearing fuels (STA 5-A1.4.6)

Long term radiotoxicity of spent fuel is dominated by the MAs: neptunium (Np), americium (Am) and curium (Cm). Therefore many different concepts have been proposed for destroying these elements (also known as ‘burning’ or ‘transmutation’). MA properties often differ significantly from those of U and Pu, in particular those of Am and Cm. Hence concepts often, though not exclusively, involve the recycling of MAs within a dedicated zone of a power reactor, with the fuel managed in a separate line from the standard fuel.

In broad terms, a harder neutron spectrum is preferable for achieving efficient burning of MAs and thus Gen IV fast reactors tend to be the most suitable systems. However there is still interest in the deployment of MA-bearing fuels in current LWRs and potentially in future, innovative epithermal LWRs (see 5-A1.6 and TA6).

UO_2, MOX or inert matrix fuels (IMFs) could be adapted to burn minor actinides in LWRs. In particular, French R&D programmes are currently active in evaluating the potential to incorporate MAs (in quantities of up to 5 tonnes per year) into both standard MOX and multiple recycled MOX. Inhomogeneity issues during manufacture are also being investigated. MA IMFs are discussed in the previous section (5-A1.4.5).

Fuel manufacture for irradiation tests (primarily Gen IV), has been carried out with MA contents that would be expected in some fuel cycle scenarios. However, this has only been achieved using relatively small scale facilities under laboratory conditions. The need for more remote handling and greater shielding compared with MOX production, means that fabrication on a commercial scale remains a significant challenge.

Some of the key issues which need to be addressed with MA-burning fuels include:

- The high volatility of Am;
- A greater need for remote handling and shielding during manufacture;
- The impact on core reactivity control;
- Residual inhomogeneity from fabrication, enhanced by phase segregation during irradiation, which can lead to 'hot spots' causing high fission gas release and a greater propensity for clad failure;
- The existence of some low melting point phases;
- Greatly enhanced helium generation, which may be a potential issue both in-pile and for long term storage and disposal (see 5-A3 and TA5-B).

6.2.1.1.10 Thorium-based LWR fuels (Thorium MOX) (STA 5-A1.5)

Natural thorium (Th) is not fissile but 'fertile' (like the U-238 isotope). To release energy from fission, it must first be irradiated in order to breed fissile U-233 through Th-232 capturing a neutron and undergoing beta decay. The range of thorium-based fuels (and associated pros and cons) in theory is as broad as those based on uranium and could be applied in all conceivable reactor systems.

To allow thorium-based fuels to begin generating energy immediately when loaded into a reactor, it is necessary to incorporate a more traditional seed fissile material (U-235 or Pu) from which U-233 gradually takes over generation as it is bred. Seed U-235 can be included in the form of unirradiated, enriched U, while seed Pu would be obtained from reprocessed LWR fuel. With U-235 or Pu seeding, the fabrication processes for thorium-based fuel are broadly equivalent to those for standard uranium-based fuel, though currently without the accompanying infrastructure which would largely have to be built from scratch. If such infrastructure was put in place, then such fuels could be adopted in a once-through cycle with spent fuel disposal.

Efficient breeding of fissile U-233 from Th for fuel recycling can be achieved more easily in thermal reactors than breeding Pu from U-238, which would require Gen IV fast reactor technology. This ability to breed with existing reactor technology has led to some avocation of the use of Th in a closed fuel cycle which does not require fresh uranium mining or enrichment, or plutonium breeding ('the thorium fuel cycle'). The fuel form for such a cycle in LWRs or HWRs would be mixed thorium-plutonium and thorium-uranium oxide fuels (so-called 'thorium MOX', in particular mixed Th-Pu oxide), with the former being used initially, and the latter being phased in as the irradiation of ThO_2 blanket material and/or the $(Th, Pu)O_2$ fuel generates U-233 fissile material which can reprocessed and incorporate into recycled fuel.

$(Th, Pu)O_2$ fuel is also of interest as an alternative to conventional MOX or inert matrix fuel (IMF, see 5-A1.4.5) for plutonium disposition purposes, in this scenario, the thoria matrix would be able to breed fissile material to take the place of the fissioning plutonium more effectively than a urania matrix, potentially allowing higher burn-ups but without generating any additional plutonium similar to IMF (though highly fissile U-233 would be generated instead).

Enhanced safety, proliferation resistance and lower radiotoxicity of waste are also cited as benefits of thorium-based fuels, but have been a matter of some international debate, especially as highly fissile U-233 has previously been used in nuclear weapons. However, recycle of U-233 into fresh fuel would require remote handling and greater shielding than even for U-Pu MOX or for minor actinide containing fuels due to the operator dose implications of highly radioactive daughter products of the decay of U-232, which is present as a by-product of U-233 production.

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In Europe, whilst experimental rods have been fabricated for test irradiations, e.g. in the Lingen BWR in Germany and ongoing in the Halden Reactor Project IFA-730 experiment, little has been done on larger scale production and irradiation of thorium MOX fuel pins in Gen II/III reactors, though TRISO coated particle thorium fuels were manufactured on a moderate scale for the Gen IV THTR-300 (Thorium High Temperature Reactor) prototype in Germany using a sol-gel route for the kernels and then subsequently irradiated. Probably the greatest current experience on thorium-based fuels is in India where a moderate number of lead test assemblies have been irradiated in commercial HWRs with similar plans being explored in China. The reactor situation regarding HWRs in Europe and the use of Rep U fuels in HWRs is discussed in 5-A1.4.1.

The advantages and disadvantages of the wet (sol-gel) and dry manufacturing routes require further investigation, in particular how they are affected by the change from experimental to commercial scale. Factors that need to be considered include the potential greater dose implications of thorium versus uranium fuel manufacture, the potential for recycling scrap pellets and grinding dusts as addback, and disposal routes for thorium-contaminated consumables such as gloves, bags, sample pots etc.

More data on the material properties, focusing on the differences between ThO_2, $(Th, U)O_2$ and $(Th, Pu)O_2)$, is required. In particular, the thermal conductivity is important, with Indian data suggesting that this can be significantly lower for $(Th, Pu)O_2$ than for $(Th, U)O_2$. In addition, the neutronics and fuel performance behaviour of full core reloads in a commercial power plant need to be better understood including the effects of ThO_2/PuO_2 or ThO_2/UO_2 heterogeneity and of high Pu or U content need further investigation (similar issues to conventional MOX fuel). Improved knowledge of helium generation in, and release from, the fuel pellets is also needed.

Following irradiation, utilities would require procedures for dealing with spent thorium MOX fuel (covered in 5-A3) and reprocessing challenges to separate Pu, U and Th should also be addressed by TA5-B (THOREX reprocessing). A particular challenge is remote and shielded manufacturing to deal with the dose hazard associated with the decay chain of U-232.

One additional, potentially interesting application of using thorium oxide in the current fuel cycle, and therefore possibly the most applicable in the medium term, is for reactivity control in an otherwise conventional UO_2 or MOX fuel. The incorporation of ThO_2, displacing UO_2 or $(U, Pu)O_2$ would act initially as neutron absorber as well as displacing a small amount of fissile material, thus reducing excess-reactivity at beginning of cycle. This would occur in a similar manner to any other burnable absorber (see 5-A1.3.4). However, unlike other burnable absorbers, Th-232 as a fertile isotope would gradually breed fissile U-233, boosting end-of-life reactivity in a more efficient manner than then the similar effect from the breeding of Pu from U-238. With optimisation, this could allow for greater reactivity control and more even power generation from the fuel throughout its irradiation life. This could reduce the amount of fuel assembly shuffling needed in the core during shutdown and therefore reduce the duration of the shutdowns and the length of time that the reactor is offline, improving sustainability and reliability.

Finally, and probably most importantly, before extensive further research is performed on thorium-based fuels and the thorium fuel cycle, much greater understanding is required of the economics and the potential advantages and disadvantages.

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6.2.1.1.11 Innovative LWR and LW Small Modular Reactor (SMR) fuels (STA 5-A1.6)

A wide variety of different innovative LWR and LWSMR concepts have been suggested, with their designs developed to varying degrees. These are considered under NUGENIA TA6, though Gen IV SMR designs are not, as these are covered by ESNII if they fast reactors and NC2I if they are High Temperature Reactors (HTRs), under the three pillars structure of SNETP.

All these designs will required fuelling if they are ever to be deployed. Whilst some use standard fuel assemblies in order facilitate more rapid deployment, others stick to established fuel materials (UO_2 and Zr alloys) for similar reasons but alter the assembly design. Some would use more radical fuel types such as those suggested for accident tolerant fuels (see 5-A1.1) or enhance economic fuels (see 5-A1.3). The use of minor actinide bearing fuels in innovative epithermal LWRs has been noted in particular in sections 5-A1.4.5 and 5-A1.4.6.

Overall, it will be necessary for TA5-A and TA6 to link closely as these designs develop and to seek areas of mutual beneficial R&D.

6.2.1.2 Fuel behaviour mechanisms and computational codes (STA 5-A2)

6.2.1.2.1 Scope

This sub-area is focused on research on fuel behaviour, in both normal operation and accident conditions, performed experimentally and with simulation models (computer codes). An understanding of fuel behaviour is underpinned by fuel R&D, which must address new safety requirements and design innovations (as defined by 5-A1) such as Accident Tolerant Fuel (ATF) and advanced recycled fuels and SMR fuels. It must also address differences in behaviour engendered by more incremental changes of the fuel pellets, cladding and assembly structural components.

The understanding of fuel rod behaviour mechanisms is facilitated by modelling using fuel performance (and safety) computer codes. Such codes are also essential for fuel design and licensing. A fuel performance code calculates the evolution of the thermo-mechanical and thermo-chemical state of a fuel rod during its irradiation (and potentially also during any post-irradiation storage) as well as, potentially, the fission gas and fission product behaviour.

6.2.1.2.2 State of the art

The main fuel behaviour mechanisms are currently identified for LWR fuel (UO_2 in Zr alloy cladding) and AGR fuel (UO_2 in stainless steel cladding) for burn-ups of up to 60 000 MWd/t. The knowledge base is formed by operational experience, PIE and dedicated experimental programmes. Based on these data, fuel performance codes are developed and validated, and are routinely used for simulation of normal operation and accident scenarios. Despite the significant effort already undertaken to understand the complex phenomena underlying fuel behaviour, many mechanisms remain uncertain, with empirical or semi-empirical models consequently implemented in performance and safety codes to ensure that the results given by the codes are reliable and validated within their domain of experience.

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Fuel suppliers currently maintain their own databases on fuel behaviour mechanisms which are used for licensing. In addition, experimental facilities (research reactors, hot cells and laboratories) are available for the research such as:

- Test reactors including BR2 in Belgium, the Halden Reactor in Norway, OSIRIS, CABRI and the under construction Jules Horowitz Reactor (JHR) in France, the High Flux Reactor (HFR) in the Netherlands, Maria in Poland and LVR-15 in the Czech Republic;
- Post-Irradiation Examination (PIE) facilities including JRC-ITU in Germany, LECA/STAR, LECI and ATALANTE in France, NNL Windscale and Central Laboratories in the UK and the Studsvik hot cells in Sweden;
- Additional test reactors and PIE facilities in Russia.

Existing international expert groups and associations include:

- International Atomic Energy Agency (IAEA) Technical Working Group on Fuel Performance Technology (TWGFPT);
- World Nuclear Association (WNA) Fuel Technology Working Group (FTWG);
- Nuclear Energy Agency (OECD/NEA) Expert Group on Reactor Fuel Performance (EGRFP);
- Nuclear Energy Agency (OECD/NEA) Working Group on Fuel Safety (WGFS);
- Nuclear Energy Agency (OECD/NEA) Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems (WPMM).

6.2.1.2.3 Challenges

- Improving fuel reliability;
- Improving and extending experimental data on fuel behaviour at high burn-up and in accident conditions;
- Improving fuel performance and safety computer codes through the development of more physically informed models by transferring results from basic research (see 5-A2.6.2);
- Maintaining key experimental facilities and expanding their capabilities to meet future requirements.

6.2.1.2.4 New Research Topics

6.2.1.2.4.1 Fuel pellet behaviour (STA 5-A2.1)

There is a need for further experimental investigation and modelling activities on both existing fuels and fuels under development (as defined by 5-A1) in order to accurately determine thermo-mechanical and thermo-chemical behaviour, to develop new or update existing models, and to incorporate mechanistic models into fuel performance codes. The main R&D effort should focus on:

- ✓ **Thermo-physical and thermodynamic properties (e.g. the thermal conductivity and heat capacity, the fuel melting point, and the Gibbs free energies of the heavy metal and fission product compounds);**

- ✓ **Thermo-mechanical behaviour including the combined effects of fuel thermal expansion, fuel densification and swelling, fuel creep, fission gas release, etc.;**
- ✓ **Microstructural changes due to irradiation (including the concentrations of point and extended defects, and formation of high burn-up structure at the pellet rim) and the associated impact on the mechanical properties and thermo-mechanical behaviour;**
- ✓ **Fission product transport and release, in particular the determination of transport properties, fission gas (xenon, krypton and helium) accumulation in intra-granular bubbles, high burn-up structure pores (together with the consequential fuel swelling) and fission gas release;**
- ✓ **The effect of Pu or minor actinide incorporation on thermo-mechanical properties as well as the defect behaviour and transport properties;**
- ✓ **The high temperature chemistry including calculation of partitioning of fission products into various compounds, fuel-clad chemical interaction (FCCI), etc.**

6.2.1.2.4.2 Fuel cladding behaviour (STA 5-A2.2)

There is a need for further research on current and advanced cladding materials (as defined by 5-A1), with experimental investigations and modelling activities focused on four important phenomena:

- ✓ **Corrosion in normal operation;**
- ✓ **Hydrogen embrittlement;**
- ✓ **Axial growth;**
- ✓ **Creep under both in-pile and post-irradiation conditions (the latter being important for dry storage evaluation).**

The effects of crud formation on corrosion and hydrogen embrittlement are also of interest.

6.2.1.2.4.3 Fuel assembly behaviour (STA 5-A2.3)

Fuel assembly dimensional stability, bowing and vibrational analysis are crucial subjects which require further experimental investigation and model development for integration into simulation tools (i.e. mechanical design codes). Other phenomena such as grid-to-rod fretting, which have historically been analysed only on the basis of operational experience, can also potentially be assessed using modern mechanical design codes if further development is undertaken.

6.2.1.2.4.4 Safety issues: fuel rod behaviour in accident conditions (STA 5-A2.4)

Fuel rod behaviour in accident conditions is a key safety issue in Europe and therefore requires ongoing research and development. Sections 5-A2.1 to 5-A2.3 addressed general fuel pellet, cladding and fuel assembly behaviour, which are also relevant during accident conditions. In addition, there are several research topic areas specific to Design Basis Accidents (DBAs), which, post-Fukushima, may also soon become relevant for Beyond Design Basis Accidents (BDBAs). These include the

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prediction of whether or not sufficient cooling of the fuel can be maintained in-reactor (during a LOCA, steam generator tube rupture, etc.), in spent fuel pools, during dry storage and spent fuel transport as well as whether or not mechanical integrity of the fuel will be maintained, in particular during a LOCA (Loss Of Coolant Accident) or a RIA (Reactivity Insertion Accident). Other important safety issues concern source term assessment in DBA or BDBA.

Some specific topics of interest in accident conditions are listed below:

- ✓ **Fuel mechanical behaviour (swelling, creep, fragmentation, etc.);**
- ✓ **Additional clad oxidation and hydrogen pick-up (including secondary hydriding after a clad burst);**
- ✓ **Cladding mechanical behaviour and clad failure conditions;**
- ✓ **Fuel dispersal and fuel-coolant interaction following a clad rupture under RIA conditions;**
- ✓ **Fuel fragment relocation in the ballooned region of a fuel rod during a LOCA, together with the consequential potential for ejection of fuel particles into the primary circuit through a clad burst, the implications of which also need to be assessed;**
- ✓ **Fuel-clad heat transfer and post-DNB (Departure from Nucleate Boiling) fuel behaviour during fast transients;**
- ✓ **Fission product (in particular xenon, krypton, iodine and caesium) release and transportation in such accidents as LOCAs and RIAs, as well as fission product behaviour in highly oxidising accident sequences (in particular ruthenium and molybdenum);**
- ✓ **The axial loads on fuel assemblies during a LOCA and the mechanical constraint due to rod-to-rod and rod-to-grid contact.**

Further details are available from the OECD/NEA Committee on the Safety of Nuclear Installations.

6.2.1.2.4.5 Integrated test facilities (STA 5-A2.5)

Integrated research reactor and post-irradiation examination (PIE) facilities are essential for R&D programmes on fuel behaviour under operational and accident conditions. A number of facilities exist in Europe for these activities and it is crucial that such capability is preserved or even expanded. Such facilities include:

- ❖ Test reactors including BR2 in Belgium, the Halden Reactor in Norway, OSIRIS and CABRI in France, the High Flux Reactor (HFR) in the Netherlands, Maria in Poland and LVR-15 in the Czech Republic;
- ❖ PIE facilities including JRC-ITU in Germany, LECA/STAR, LECI and ATALANTE in France, NNL Windscale and Central Laboratories in the UK and at Studsvik in Sweden.

In addition, the Jules Horowitz test reactor (JHR) will soon become operational in France.

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6.2.1.2.4.6 Simulation (STA 5-A2.6)

During the last decade, considerable effort has been committed to the development of simulation tools for the purposes of optimising fuel design and operation in reactor, as well as margin evaluation as part of safety assessment. There is a need to develop a pan-European platform, in particular for safety-related aspects, as has been achieved for thermal hydraulics and neutronics.

A fuel performance code calculates the evolution of the thermo-mechanical and thermo-chemical state of a fuel rod during its irradiation (and potentially also during any post-irradiation storage). This involves modelling a large number of phenomena. These include:

- ✓ **Those associated with thermo-mechanical behaviour of the fuel and clad materials including heat transfer by conduction, convection and radiation, thermal expansion, creep, elasticity, plasticity, fatigue and melting behaviour;**
- ✓ **Phenomena related to the presence of a neutron flux including clad hardening, embrittlement and axial growth;**
- ✓ **Phenomena related to fission, neutron capture and the generation of fission products, including (non-uniform) heat generation, the generation and release of fission gases (xenon, krypton and helium) and fission products, as well as fuel densification and swelling;**
- ✓ **Phenomena related to microstructural changes in the fuel, including grain growth and the formation of high burn-up structure (HBS) at the pellet rim;**
- ✓ **Phenomena related to radial temperature gradients in the fuel pellets, including pellet cracking and fuel fragment relocation, pellet wheat sheafing, axial extrusion and dish filling;**
- ✓ **Chemical phenomena including fuel-clad bonding, stress-corrosion cracking (SCC) and clad oxidation.**

The use of fuel performance modelling is legitimised by three important factors:

- The material properties are simulated using experimental data on irradiated fuel pellet and clad materials;
- The predictions are validated against experimental data from Post-Irradiation Examination (PIE) and in-reactor measurements, allowing quantification of uncertainties;
- The phenomenological modelling is as mechanistic as possible, with, where necessary, simulation at different length and time scales.

6.2.1.2.4.6.1 Industrial and engineering codes (STA 5-A2.6.1)

Improvements are still necessary for industrial and engineering codes, especially for simulating pellet-clad interaction in normal operation and transient conditions, for modelling fuel behaviour during RIA and LOCA transients, and also for developing a multi-physics approach by coupling fuel thermo-mechanical codes with thermal hydraulics and neutronics codes in order to have more predictive simulation tools. In these industrial codes, it is necessary to implement more advanced models in particular for clad and pellet thermo-mechanical behaviour, fuel fragment ejection, fuel-

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coolant interaction, and to take more explicit account of the effect of microstructural evolution as well as fission product and fission gas behaviour. These simulation tools also need to include uncertainty evaluation. Validation of such codes against experimental data is vital.

The structural behaviour of fuel assemblies is an area which has not historically received as much attention as the behaviour of the fuel rod. However, with increasing burnups, the fuel assemblies are subjected to more and more demanding conditions. In particular, fuel assembly bowing has been in evidence under some conditions, leading to degradation in control rod drop time. As this is crucial in order to control core reactivity, it is now vitally important to investigate and model fuel assembly structural behaviour in a well qualified code.

6.2.1.2.4.6.2 Multi-scale approach (STA 5-A2.6.1)

Due to the very complex and interdependent phenomena underpinning fuel behaviour, a multi-scale approach from the atomic to the macroscopic scale is necessary to improve understanding of the relevant phenomena during normal operation, transient and accident conditions. This will lead to improved prediction of fuel behaviour, in particular fission gas and fission product behaviour, thermo-mechanical property changes during irradiation, fuel fragment relocation, crack propagation, clad hydriding, the effect of minor actinide incorporation, transport properties and defect behaviour. Such tools also allow for the possibility of predicting the behaviour of fuel beyond the strict validation domain and to allow the techniques to be applied to innovative fuels (as defined by 5-A.1). Furthermore, it will allow integration of all the results of the research on the physical mechanisms involved in the fuel behaviour conducted in 5-A2.1 to 5-A2.4.

Modelling tools from the atomic to mesoscale include electronic structure calculations, classical Molecular Dynamics (MD), Kinetic Monte Carlo (KMC) or cluster dynamics. They have been proven (in the F-BRIDGE project for fuels and in PERFORM 60 for structural materials) to be mature enough when combined with key characterisation techniques to identify and discriminate between the relevant mechanisms that must be taken into account in fuel behaviour simulation at a macroscopic scale. Further improvement of these methods is desired in order to better identify mechanisms, to increase data precision and to study more complex materials e.g. new fuel types (see 5-A1).

There is also a need to show how the results obtained from basic research studies in the context of a multi-scale approach might be incorporated into the fuel performance and safety codes models in order to yield an effective improvement.

In addition, the interaction between the fuel pellet and the cladding during transients is of particular importance, as taking account of mechanical effects in a multi-scale simulation is currently hindered by specific modelling difficulties which remain to be addressed.

6.2.1.3 Fuel treatment, transportation and interim storage (spent fuel management) (STA 5-A3)

6.2.1.3.1 Scope

This sub-area considers spent (used) fuel management following reactor utilisation. Management activities include handling of the spent fuel, associated diagnostics (determination of fuel assembly condition), storage in spent fuel pools at power plants, transport, interim storage in either wet or dry

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conditions before either reprocessing and recycling or transfer for final disposal, which may necessitate repackaging of the fuel.

6.2.1.3.2 State of the art

Spent fuel management practices, for various nuclear fuel types including both commercial and research reactor fuels, have benefited from the accumulated knowledge and experience of over 50 years of wet storage and over 30 years of dry storage activities. Nevertheless, there is a space for improvement and optimisation in individual areas as well as across the entire back-end of the fuel cycle (storage, reprocessing/recycling and disposal activities) in conjunction with IGD-TP (Implementing Geological Disposal of radioactive waste Technology Platform). This can result in better safety, security (proliferation resistance), economic and environmental characteristics. Spent fuel management is carefully regulated according to standards established by national regulators, usually reflecting recommendations of international organisations such as the IAEA.

Within the EU, a range of spent fuel storage arrangements are employed. In some countries, spent fuel is stored primarily at the reactor site where it was generated, whereas in other countries centralised storage is used for interim/long term storage after an initial cooling period at the reactor. Transport of spent fuel is a well established and frequently undertaken operation, employing a variety of means, including lorries, freight trains and maritime transport.

Recycling of UO_2 and metallic fuels is well established within some countries in the EU. The continued development of fuel and the effects of higher burn-up irradiations raise the potential for changes in recycling process parameters. In the main, such changes are anticipated to be incremental, except for proposed fuels of some alternative compounds such as carbides or silicides (see 5-A1) where pre-treatment such as oxidation may be required or alternative reprocessing technologies used.

6.2.1.3.3 Challenges

- Handling and storage of leaking fuel assemblies;
- Optimisation of cask loading;
- Burn-up credit challenges (code validation and licensing issues);
- Handling of fuel assemblies after long term storage e.g. repackaging for disposal or preparation for reprocessing;
- Qualification and monitoring of casks for longer term storage;
- Interface with/requirements for deep geological repository (fuel loading, limits and conditions) and the potential for fuel repackaging;
- Reprocessing and recycling of high burn-up, MOX and advanced fuels (as described in 5-A1) such as Accident Tolerant Fuels (ATF) or advanced recycled fuels;
- Optimisation of spent nuclear management activities.

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It should be noted that severe accident assessment of spent fuel storage is covered under TA2.

6.2.1.3.4 Research Topics

6.2.1.3.4.1 Back-end integration (STA 5-A3.1)

Within the EU, a range of spent fuel storage arrangements are employed. In some countries, spent fuel is stored primarily at the reactor site where it was generated, whereas in other countries centralised storage is used for interim/long term storage after an initial cooling period at the reactor. Recent work in the USA has indicated that significant economic advantages exist for a centralised storage regime, particularly in the light of increasing security requirements associated with spent fuel storage and the decrease in proliferation resistance associated with long stored fuel.

Interim storage, occurring between the minimum time necessary to export fuel from a reactor site spent fuel pool and the time at which fuel is reprocessed or sent for disposal, is a necessary feature of any fuel cycle. However, delays in the provision of disposal facilities and underutilisation of reprocessing have resulted in substantial increases in the duration of interim storage, from a few years up to potentially 100 years. Technical options that have been developed or adapted for providing interim storage capacity include wet storage in ponds, dry storage in casks and dry storage in vaults. In the absence of clearly defined requirements for disposal and timescales for interim storage, it is not possible to be definitive with respect to optimum spent fuel management strategies. Nevertheless, it is important that dependencies within the spent fuel management system and their implications in terms of economic and environmental impacts are understood.

6.2.1.3.4.1.1 Economic and environmental impact (STA 5-A3.1.1)

There is ample evidence that both wet and dry storage provide safe and effective management. The impact of increasing storage times, evolving technical standards and the need to maintain the capability for final export, all affect the costs of spent fuel storage. Although spent fuel management currently accounts for a small fraction of the total cost of generation, due in part to the discounting of deferred costs, the absolute level of expenditure can be significant for utilities and/or national budgets. The optimum solution is likely to vary according to the characteristics of the systems of individual countries, and clear understanding of the sensitivity to cost and the desirability of particular solutions is important in setting spent fuel management policy.

As the timescales for final disposition of spent fuel increase, it is increasingly important to understand the impact of storage system choices and the potential implication of ongoing nuclear generation strategy on the scale and type of spent fuel storage required in the future. It is important to understand the implications of national geological repository characteristics and fuel cycle policy and strategy development programmes, in order to avoid drivers that undermine the system efficiency and hence may increase costs and risks.

The impact of fuel cycle options on the environment, including the use of resources and secondary wastes arising from storage activities, is also an important aspect of back-end optimisation.

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6.2.1.3.4.1.2 Security (STA 5-A3.1.2)

The security requirements of spent fuel management facilities are likely to change over time, because as spent fuel ages, the radioactive content decreases and hence it becomes less self-protecting. There are different challenges for spent fuel management, for example due to variations in the accessibility of spent fuel and the number of people normally present at individual facilities. Therefore it is important to identify the specific vulnerabilities and then put in place the necessary security arrangements.

6.2.1.3.4.1.3 Safety and environmental requirements (STA 5-A3.1.3)

The requirements for safe storage are evolving in response to increasing storage timescales, operating experience and societal expectations. These requirements concern both normal operations and the extent of provision for abnormal events. Some requirements may favour particular solutions and some requirements may arise only when a certain scale of operation is reached. It is important to understand the implications of these requirements on the respective desirability of AR/AFR storage.

6.2.1.3.4.1.4 Legacy and non-LWR fuels (STA 5-A3.1.4)

Although zirconium alloy-clad LWR fuels form the bulk of the spent fuel in the EU, further work is required to ensure that legacy fuels from early development work and other less common designs, are adequately understood so that their safe storage and where appropriate reprocessing can be undertaken efficiently and effectively. Establishing networks in areas of common interest for the exchange of safety information will ensure that duplication is minimised and that such topics can be addressed alongside LWR considerations.

6.2.1.3.4.1.5 Novel fuels (e.g. Accident Tolerant Fuels and SMR fuels) (STA 5-A3.1.5)

As new fuel options are developed for near-term deployment in power generation, it is important that compatibility with current back-end activities is assessed in order to identify sensitivities, incompatibilities and or knowledge gaps. Research into fuel and assembly performance post-reactor operations is to be anticipated as part of fuel development programmes, so as to enable demonstration of satisfactory economic and environmental performance and compatibility with disposal requirements for the end products of the fuel cycle.

As a minimum, compatibility with short term pond storage and interim wet and dry storage should be demonstrated so as to avoid compromising current spent fuel management options. Reprocessing operations may require significant re-engineering for novel fuels and early identification of the potential implications for the production of recycled fuels need to be studied in parallel with novel fuel developments.

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6.2.1.3.4.2 Wet storage (STA 5-A3.2)

6.2.1.3.4.2.1 5-A3.2.1 Fuel assembly ageing (STA 5-A3.2.1)

The effects of helium build-up in spent fuel during extended periods of storage is not fully understood, although research work in this area is underway, primarily driven by dry storage considerations.

The corrosion of LWR fuels is largely well understood and predictable, although some further work may be required to consider severe accident conditions or the effects of low probability mal-operation scenarios. Innovations that reduce crud formation during reactor operation will significantly reduce the release of activity into the pond water and the consequential environmental impact, worker dose and economic cost.

Assembly hardware must also retain its integrity during storage in order to ensure that fuel is retrievable for export or transfer to further storage. Current assembly materials are compatible with storage environments and there are no indications of significant life-limiting behaviours over foreseeable timescales.

6.2.1.3.4.2.2 Fuel integrity monitoring/failure detection (STA 5-A3.2.2)

This topic area includes the development of remotely deployable systems to locate failed fuel and the site of fuel failures will assist in minimising activity arising during pond storage, and will simplify operations associated with a transition of spent fuel to dry storage/disposal.

In addition, the development of techniques and systems that will remain operational with minimal emergency power requirement in the event of major loss of power conditions will reduce risk and aid emergency management.

6.2.1.3.4.2.3 Failed fuel treatment/storage (STA 5-A3.2.3)

Releases from failed fuel provide one of the design bases for activity management systems in pond storage. Developments that improve the efficiency and reduce waste generation from pond-water clean-up would provide environmental, safety and economic benefits.

Understanding of the evolution of the structure of failed fuel and its composition becomes of greater importance as the duration of storage increases because of cumulative effects and because low rate effects can become important over longer timescales. Evolution of fuels in contact with storage environments will also affect activity and material releases during long term storage and potentially during subsequent management activities. There are synergies in this area with work undertaken within IGD-TP and its associated knowledge base.

Failed fuel has often been managed through continued storage in reactor spent fuel storage pools or by reprocessing, since degradation protects are generally small in quantity and compatible with current reprocessing technologies. Development of strategies and standards to deal with failed fuel whilst minimising impact on plant operations, environmental discharges and subsequent fuel management options will support enhancements in the safety of spent fuel storage and disposal.

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Repair of spent fuel assemblies damaged due to mishandling is carried out when necessary to ensure that the assemblies remain retrievable.

6.2.1.3.4.2.4 Pond design and infrastructure (STA 5-A3.2.4)

Most existing fuel storage ponds maintain safety by active means. Some recent designs include passive cooling. Passive safety is one of the NUGENIA priority R&D topic areas and therefore further development is encouraged in order to improve passive safety and the provision of critical status information without external supplies/intervention in new designs and for application to existing storage ponds.

Most pond infrastructure is similar in character to parts of NPPs, e.g. massive concrete structures, and therefore the application of knowledge developed in NPPs is likely to provide adequate underpinning for many aspects of pond ageing management. However, maintenance of water barriers and the integrity of pond liners are likely to remain as storage specific topics additional to such knowledge bases. Remote inspection technologies provide essential information for life extension and assurance purposes, and their continued development will improve coverage, deployability and resolution, all of which will enhance ageing management strategies.

Technologies for sealing containment leaks have been developed, however further work on options to reseal ponds under extreme conditions may be valuable.

6.2.1.3.4.2.5 Severe accidents (STA 5-A3.2.5)

NPP spent fuel pools and interim storage ponds have potentially very different responses to loss of cooling events due to the minimum cooling time of the fuel normally discharged to the pond and the absence of emergency core off-load into interim storage ponds. Thus cooling requirements for interim storage pools may only be required transiently and best estimate assessments are likely to be more appropriate for performance assessment and design.

The drying behaviour of cladding and fuels and their subsequent behaviour in hot steamy air, potentially in the presence of contaminants from emergency water sources, needs to be fully understood in the light of the events at Fukushima. Where events could lead to substantial changes in moderation/neutron absorption and/or fuel relocation, consideration of the implications for criticality safety is required. This activity is covered under TA2 on severe accident assessment.

6.2.1.3.4.3 Dry storage (STA 5-A3.3)

6.2.1.3.4.3.1 Existing LWR dry storage programmes (EPRI ESCP, IAEA, OECD/NEA) – Identification of complementary work programmes (STA 5-A3.3.1)

Dry storage of LWR fuel is widely established in the EU and elsewhere in the world. The move to greater storage times has led to the need to re-license many storage systems that were not originally designed for this timescales and has identified significant knowledge gaps. Substantial work has been initiated internationally, e.g. via the Enhanced Storage Collaboration Programme and activities in the IAEA and OECD/NEA, building on continued R&D in several countries, notably Germany and Japan. Complementary work to address identified knowledge gaps represents the most effective means of continued progress in the safe and effective storage of spent LWR fuels. Knowledge gaps vary

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according to the length of the anticipated storage period. Currently work is focused on demonstrating adequate performance for up to 100 years and to a lesser extent considering longer durations.

Important topic areas include the evolution, qualification and monitoring of:

- ✓ **Fuel matrix and cladding;**
- ✓ **Steel canister corrosion;**
- ✓ **Cask seal integrity and bolt ageing;**
- ✓ **Thermal and age-related degradation of neutron absorbers;**
- ✓ **Concrete overpack and pads;**
- ✓ **Development of improved modelling, particularly for thermal modelling where existing models are non-conservative with respect to some storage behaviours such as condensation on canister surfaces;**
- ✓ **Degradation of fuel baskets during interim storage periods.**

There is a need to develop techniques to monitor evolution of fuel in storage, to qualify fuel for subsequent activities and to provide recommendations on the periodicity of monitoring/inspections.

Current work is at the small sample or whole assembly level (US cask demonstration project). This leaves a gap in experimental data at the level of fuel rod performance, and work to understand the interaction of effects at this scale and to underpin mechanistic modelling of rod behaviour is warranted. The underpinning of approaches to address the mis-match between experiment durations and storage times is also required. Extrapolation of such knowledge to higher burn-up fuels and longer timescales would be beneficial in understanding future behaviours.

6.2.1.3.4.3.2 MOX fuel (STA 5-A3.3.2)

Much recent international review work has focused on uranium dioxide fuel, however further work will be required for MOX fuels in particular on the evolution of the fuel matrix and air oxidation.

6.2.1.3.4.4 Fuel drying (STA 5-A3.4)

Fuel drying affects subsequent storage behaviour but is constrained by the performance of the fuel during drying and the impact on throughput and costs. A number of processes are used to dry fuel, with vacuum drying and warm, inert gas drying being dominant. There is a range of information available on the effectiveness of current drying processes, which indicate that the processes and the effectiveness of drying are yet to be fully underpinned.

Measuring the degree of drying in fuel casks relies on measurement of water content, even though the most common cause of water carryover is ice formation in assembly structures. Experimental evidence on the accuracy of acceptance tests for dried fuel is limited and past experience is often cited to support test methods. Further work to determine effective test methods and instrumentation would provide additional confidence and consistency of approach within EU states.

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6.2.1.3.4.4.1 Acceptable moisture carryover (STA 5-A3.4.1)

A range of acceptable water carryover criteria exist. The basis of such criteria is not clear including the physical processes on which they are based. There are likely to differences arising from variations in fuel cladding and structural materials, however establishing relevant mechanisms and common objectives would enhance the technical basis for criteria used.

Requirements may change with desired downstream operations: transport, storage or disposal.

6.2.1.3.4.4.2 Failed fuel detection (STA 5-A3.4.2)

Many dry storage systems assume that only intact fuel is stored and hence primary containment is provided by the fuel cladding. A variety of techniques for detecting fuel failures exist, however as the fuel ages, the continued effectiveness of these techniques needs to be established or alternatives need to be identified. The demonstrated effectiveness of techniques to confirm that the fuel remains intact after drying, would strengthen the demonstration of safety and compliance.

6.2.1.3.4.4.3 Impact of drying on fuel assembly, cladding and fuel matrix (STA 5-A3.4.3)

Fuel heating rates and maximum temperatures are specified to minimise the risk of hydride reorientation and the risk of embrittlement. Limits are also defined according to incremental damage, stresses and physio-chemical changes.

For spent fuel with cladding defects, further work on how long a failed fuel rod can be exposed to an oxidizing environment before the small defect propagates, would assist in defining safe operations and would provide a means for improved management of fuel with defects. Conditions could be applicable both to in-situ dry storage and to interim storage at a central site.

6.2.1.3.4.5 Transport (routine and post storage) (STA 5-A3.5)

6.2.1.3.4.5.1 Post storage fuel export (STA 5-A3.5.1)

Interim storage requires the ability to be maintained to export fuel long after the engineering and human infrastructure associated with operating reactor stations have gone. Such requirements have implications for the expected time to initiate fuel export and the speed with which exports would be expected to occur. Such considerations also affect infrastructure requirements, minimum equipment availability, capability maintenance and institutional arrangements to ensure that capability is maintained over time.

6.2.1.3.4.5.2 Transport of casks (STA 5-A3.5.2)

Many storage systems are approaching the end of their initial licensing periods. The lessons learned from relicensing need to be examined in order to identify knowledge gaps to provide the technical underpinning required for future licensing reviews and/or extensions to longer storage periods.

6.2.1.3.4.5.3 Post-storage transport challenges to fuel integrity (STA 5-A3.5.3)

Ageing of fuel and cladding during transport is considered elsewhere for storage and loading operations. However the resilience of long stored fuel must also be demonstrated for normal transport activities and information is required on the extent to which failures during abnormal events may be worse either in probability or extent. Given that current storage systems contain fuel in arrangements, frameworks and casks that are often larger than current transport flasks, the differences in applied or possible loadings during transport may need further underpinning.

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6.2.1.3.4.6 Recycling (STA 5-A3.6)

Fuel recycling is well established within some EU countries. Continued development of fuel and the effects of higher burn-up irradiations raise the potential for changes in recycling process parameters. In the main, such changes are anticipated to be incremental, however other aspects remain to be studied in areas such as:

- Recycling of MOX fuel;
- Multi-recycle of fuels;
- Minor actinide inclusion.

The adaptation of existing reprocessing technology may be more challenging for legacy and non-LWR fuels as well as proposed new fuels of some alternative compounds such as carbides or silicides (see 5-A1), where pre-treatment such as oxidation may be required or alternative reprocessing technologies may need to be pursued. It should be noted that, if developed, LWR carbide fuels would have the potential for crosscutting research in the field of reprocessing with legacy fast reactor carbide fuels as well as ongoing fast reactor carbide fuel cycle development programmes e.g. in support of ESNII.

6.2.1.3.4.6.1 Ageing fuel (STA 5-A3.6.1)

Fuel currently in interim storage could, in principle be retrieved for recycle at some future date. The ageing of the fuel may have implications for such activities. Therefore understanding is needed of:

- ✓ **Impact of radionuclide decay of isotopic and elemental composition;**
- ✓ **Impact of degradation products;**
- ✓ **Definition of the suitability criteria for recycling;**
- ✓ **Effects of interim storage on fuel mechanical properties and head end reprocessing activities such as shearing.**

6.2.2 Technical Area 5B (STA 5-B)

6.2.2.1 Scope

This subarea is focused firstly on safe the management of nuclear plant operational waste and decommissioning “waste”, and secondly the decommissioning/dismantling step in the nuclear plant lifetime. The aim is to minimise the arising waste through good design and operational practices, and to develop new technical solutions through focussed R&D on waste management and decommissioning.

6.2.2.2 State of the art

According to IAEA as of 31 December 2013,, 11 NPPs have been fully decommissioned and 149 nuclear reactors have been permanently shut down; many other nuclear facilities have also been decommissioned (research reactors, radioisotopes production facilities, reprocessing plants, fuel fabrication facilities, military production reactors, etc.). Similarly nuclear plants have operated

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successfully for many decades and the arising wastes managed safely, through treatment, storage or disposal.

A significant amount of experience and knowledge has been accumulated and it is important that this is shared as the number of waste management and decommissioning operations increase. This body of knowledge will grow further as experience is gained and will define current best practice. This knowledge should also inform the design for future systems.

Best practices are shared and guidelines for decommissioning are drafted by international organizations (including IAEA, OECD NEA – Working Party on Decommissioning and Dismantling). Identification of optimum decommissioning strategy and reliable costing methods has been recently proposed.

6.2.2.3 Challenges

- To learn from current experience and identify best practice in waste management and decommissioning,
- To develop characterisation techniques for waste inventory assessment and plant and facility assessment to aid planning for decommissioning
- To innovate enhanced decontamination and dismantling technologies for structures and components, incl. remote dismantling techniques
- To establish improved treatment technologies (thermal or other) to reuse/recycle materials, minimise waste volumes and to develop robust and passive waste forms.
- To accelerate the introduction of new technologies and technical approaches through inactive and active demonstrations.
- Waste minimisation strategies for decommissioning, incl. safe release of material to the environment, recycle/reuse, disposal to VLLW repositories (landfills) along with reliable and cost effective activity measurement (assay) techniques
- Organizational aspects: Standardization of processes, Identification of synergy effects for multi-unit sites or fleet-wide D&D projects, optimization of post-operational phase,
- Change Management from operation to decommissioning organisation

6.2.2.4 Research Topics

6.2.2.5 Waste Management (STA 5-B4)

6.2.2.5.1 Waste Management (STA 5-B4.1)

The ideal way to minimise waste arising in line with the principles of the waste hierarchy is to reduce waste during the design phase, to minimise the waste arising during operational and decommissioning activities. The growing experience of waste management and decommissioning of reactors, fuel cycle and research facilities is building a body of knowledge, that should be compiled to inform ongoing operations and the design of future plant: example being the further development of ‘design for decommissioning’ principles and ‘waste management’ principles.

In addition to these design and operational guides, opportunities exist to consider other specific aspects of the future fuel cycle operations which illustrate these principles. For example:

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In the fuel cycle front end:

- ✓ **Alternative UOX/MOX fuel manufacturing processes (co-conversion, denitration, sol-gel, etc.) in order to reduce the number of process steps, the fabrication scraps and the technological waste produced,**
- ✓ **Improving the design of fuel elements to reduce the amount of structural waste, and its activation,**
- ✓ **Processes to recover and recycle uranium present in the fluorination scraps, in the sludge coming from the liquid effluent treatments,**
- ✓ **Processes for generic uranium residue treatment and recovery,**
- ✓ **Processes to recover and recycle uranium present in the solid waste (incineration, decontamination…) produced at each stage, including fuel manufacturing.**

In the back end:

- ✓ **Optimized separation processes to reduce the volume of effluents, the process waste and the technological waste produced (decrease in the number of cycles of purification, increase of factors of decontamination, search for new extractants, substitutes for certain reagents…),**
- ✓ **Processes to reduce alpha emitters contamination of technological waste from the manufacture of MOX fuel,**
- ✓ **Processes to recover and recycle plutonium present in the solid waste (incineration, decontamination, dissolution…) produced during reprocessing and MOx fuel manufacturing,**
- ✓ **Equipment to reduce the waste constituted by filters (decloggable metallic filters, oxide boxes recycling…),**
- ✓ **New glass matrix able to accept more radioactivity (alpha emitters particularly) and chemicals,**

In the fuel cycle:

- ✓ **Matrix able to confine the radioactivity and chemicals recovered by the liquid effluent (cement based or not),**
- ✓ **Processes to treat organic radioactive effluents (oil, solvent), such as "high temperature and pressure oxidation" process,**
- ✓ **Waste conditioning processes to reduce the volume of waste packages and improved containment of radioactive material (melting, new ceramics, etc.),**
- ✓ **New technology choices relating to the materials and the design of structures and equipment of fuel cycle facilities which may reduce the activity and the volume of waste generated during the dismantling and facilitate their treatment, cleaning-up, dismantling, conditioning and sending to existing waste disposal streams**

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(limitation on activation, decontamination of surfaces optimization, minimization of the use of hazardous substances…).

6.2.2.5.2 Waste Characterisation (STA 5-B4.2)

Characterisation of waste material is a critical step in waste management. This represents a significant challenge both to understand the inventory of operational wastes and also to understand the nature of wastes occurring during decommissioning. Knowledge of the waste inventory is the first step in determining the appropriate waste management strategy; treatment, reuse / recycle, storage or disposal. The development of remote and in-situ technologies for waste assay clearly offer a significant advantage. The emergence of technologies for radiological/dose measurement and non-destructive analysis, in combination with remote RAMAN and LIBS offer interesting possibilities for the future.

Effective characterisation can also facilitate segregation of waste streams, with the potential to increase reuse/recycling and to minimise higher volumes of higher level wastes.

Further research and development in waste characterisation is proposed.

6.2.2.5.3 Waste Reuse and Recycling (STA 5-B4.3)

In line with current regulations the reuse and recycling of materials from operational waste and decommissioning waste is encouraged where it reduces the amount of waste for disposal and/or recovers valuable materials; with net lifecycle benefits. Materials recovered could be reused or recycled within the nuclear industry or potentially released into the non-nuclear market. This relies on the availability of technologies to:

- decontaminate materials to the state where they can be reused/recycled
- assay technologies to verify the composition of materials prior to future use

The establishment of potential reuse/recycle routes for materials should be established and where practical, decontamination and assay technologies developed to achieve this goal of reducing waste.

6.2.2.5.4 Waste Treatment (STA 5-B4.4)

The processing of nuclear material generates waste. Such waste requires treatment to limit its impact on the environment, with wasteforms produced being suitable for ongoing storage and ultimate disposal. In the nuclear industry vitrification has typically been used for the immobilisation of high active liquors with cementation and other ambient temperature technologies for immobilisation of lower level wastes. Such technologies may no longer be optimal for all today's and future waste arisings.

Demonstration of best practice and optimisation of wastes treatment requires that alternative or novel wasteform matrices and their associated processing routes are investigated. Such methods may include alternative cementation or other ambient temperature processing routes.

Thermal treatment routes which result in the passivation of active metals, the destruction of organics and the immobilisation of radionuclides in robust matrices should be explored. Thermal treatment of waste high metallic fractions should also be appraised with the aim of recycling and thus limiting the

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amount of material sent for repository disposal. New waste immobilisation technologies need to be assessed and demonstrated against a range of wastes arisings.

In addition to operational wastes, decontamination and decommissioning activities will result in significant mixed wastes and effluents containing organics bringing treatment challenges.

These will include wastes from decontamination and decommissioning activities as well as operational wastes. Reactor decommissioning will give rise to ion exchangers, tritiated and graphite containing wastes requiring solutions. Research and development is needed to develop processes for specific wastes.

6.2.2.5.5 Waste Storage (STA 5-B4.5)

Storage is a necessary step in the overall management of radioactive waste.

Some specific functions can be defined for storage facilities:

- Surveillance, maintenance of the facility and of the stored objects;
- Management of the stored objects (traceability, radiological inventory, storage mapping, etc.);
- Retrieval at any time, under conditions planned, of stored radioactive objects (packages) and possibility of reconditioning or repackaging.

In recent years, mainly because of the unavailability of permanent disposal facilities, storage facilities have had their lifetimes extended and serious consideration has been given, in some countries, to the use of storage as a long term management option.

Different types of storage are discussed hereafter: interim storage, long term storage and decay storage.

6.2.2.5.5.1 Interim Storage

Interim storage is a temporary storage whilst onward handling is being arranged. This might include the conditioned wastes waiting for disposal.

A great number of storage facility design solutions have been developed taking into account the variety of “stored materials and objects” such as activated solids, vitrified waste, drums or containers containing technological waste…

Many waste interim storage facilities are approaching the end of their initial licensing periods. The lessons learned from relicensing need to be examined in order to identify knowledge gaps and technical underpinning required for future licensing reviews or extensions to longer storage periods.

An important challenge is to improve the passiveness of this type of nuclear facility; development could be made in that way: cooling by natural convection system, I&C system as simple as possible.

In any case, the storage facility and its monitoring system should be designed so as to be able to detect any degradation of stored packages and identify involved packages, in order to initiate corrective actions. A surveillance programme has to be implemented in this objective. Non-destructive methods should be developed in order to improve the surveillance as well as to minimize

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waste generated by the storage operation. Different phenomena have to be monitored: corrosion, internal pressure or temperature, container integrity etc. Also, evolution forecast of the waste and packages characteristics (degradation by radiolysis or thermolysis, corrosion…) could be the subject of R&D work.

Record keeping of the waste data has to be organized. Different kinds of information support could be used and their development has to be aimed at ensuring their availability for very long period of time.

6.2.2.5.5.2 Long-term storage

The order of magnitude of the "long term" is considerably longer when compared to a few decades used in the designing of the existing interim storages. The codes and standards used for the facility design are often valid only for a period of several decades (e.g. 50 years). Beyond this period, codes and standards may need to be developed.

Given the storage lifetime and the interest of limiting human interventions, the design should combine simplicity and robustness, incorporating sufficient margins to take into account technical or regulatory uncertainties related to the reference time scale. Natural environmental evolutions are difficult to forecast over extended periods of time.

Moreover, the notion of "long term duration" should also lead to question about the sustainability of the facility management by an operator and its control by the safety authority, over a period of time when a decline or disappearance of nuclear industrial activity could be possible. This question and scenarios of loss of control of the storage may be considered. In addition, the scenario of final forgotten of the facility should be analyzed.

By the nature of long-term storage facility, ageing is fundamental for this type of facility. Defects can impact components by changes in material properties because of ageing degradation by irradiation, by thermal effects or by normal wear. If ageing phenomenon cannot be avoided, it is necessary to predict its magnitude, in order to define acceptable margins during the storage lifetime. R&D works could be carried out on this topic.

In any case, periodic safety review has to be performed for long term storage solutions. Studies could be performed in order to improve this safety process, in particular the control of the facility conformity to the safety case should be considered.

6.2.2.5.5.3 Decay storage

Some types of radioactive waste may be stored for periods specifically to allow the radioactivity of the waste to decay to levels that permit its final disposal or authorized discharge or removal from regulatory control (i.e. clearance). Storage for decay is particularly important for the clearance of radioactive waste containing short lived radioisotopes. Practical experience shows that storage for decay is suitable for waste contaminated by radionuclides with a half-life of less than about 100 days. Representative measurements should be carried out on samples taken and analyzed prior to the removal from control.

6.2.2.5.5.4 Condition monitoring

Monitoring throughout the storage period should be performed both by monitoring the facility and its environment (monitoring of containment barriers, radiation monitoring and environmental surveillance) and by monitoring the stored objects. This should include the periodic retrieval of some

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adequately selected “control” packages in accordance with a defined monitoring plan to estimate the effect of ageing and to anticipate generalized damages on packages which can lead to their necessary reconditioning.

This “active” monitoring requires:

- to have adequate access to stored packages,
- to have an examination cell for stored packages.

In some cases, samples can be pre-positioned in the facility in order to follow evolution rates.

Different phenomena have to be monitored including, corrosion, radiolysis phenomena, thermal phenomena and concrete alteration. Different parameters could be monitored including barrier thickness, pressure and temperature.

In order to minimize human intervention and waste generation, non-destructive monitoring methods should be developed

6.2.2.5.6 Waste Disposability (STA 5-B4.6)

The products of waste treatment are required to be suitable for interim storage and ultimate disposal in line with the particular national requirements. Any treatment process aimed at passivation of the wastes cannot be sanctioned until an assessment has been carried out of the behaviour of the product. Since wasteform performance will be linked to the environment in which the product is disposed, techniques need to be applied to predict long term behaviour feeding data back into the formulation of such wasteforms. Research in this area needs to be linked to the IGD-TP platform to enable the wasteform/package performance to be assessed against the appropriate disposal environment inclusive of backfill type, pH and composition of any groundwater.

6.2.2.5.7 Asset management and care (STA 5-B4.7)

Ongoing monitoring of equipment and facilities maintenance are obviously important activities during throughout the plant lifetime. This is increasingly important for long term storage facilities, ageing infrastructure and plants awaiting decommissioning. Technologies and techniques that are deployable to remotely monitor for equipment/facility degradation such as non-destructive analysis and in situ monitoring can give early warning of deterioration and allow action to be taken promptly.

6.2.2.6 Decommissioning and Dismantling (STA 5-B5)

6.2.2.6.1 Pre-planning and Decommissioning Strategies (STA 5-B5.1)

The pre-planning of the decommissioning of a nuclear facility should optimally start when designing the plant. For the Generation II reactors the decommissioning of such plants was only considered to a limited extent during design. Detailed decommissioning planning should start several years prior to the final shut down date of a plant. For adequate planning, the following items should be addressed as early as possible:

- Strategy: direct dismantling vs. safe enclosure; waste management concept (this decision could be subject to country-specific legislation)
- Licensing: licensing strategy and preparation of application for decommissioning approval

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- Organization: different or modified responsibilities, processes and work flows
- Change Management
- Characterization: radiological characterization of the whole facility (please refer to item 5.5.3)
- Fuel: planning and realization of fuel and debris removal from spent fuel pool
- Plant optimization: shut down of installations only necessary for operation (please refer to item 5.5.32)
- Documentation: modification of specifications for safety relevant installations and operational documentation (organizational manual)
- Risk modelling: internal (i.e. technical) and external (e.g. change in regulatory framework) risks
- Safety: reduction of radioactive and conventional hazard inventory within the plant

Prior to or during the pre-planning of a decommissioning project the decision, whether to directly dismantle or to safe enclose the respective plant has to be taken. Mixed concepts are of course possible, but depend on country specific regulations and laws. The end state of decommissioning could be brownfield or a greenfield: i.e. end of decommissioning could mean both the release from the country specific atomic law (brown field – buildings not dismantled yet) or complete site restoration to green field conditions.

Risk modelling is an important task that is certainly already addressed during design, construction and operation of NPP. The methodologies applied in these stages of the NPP's life should be adapted to the special requirements during a D&D project and the fact that the radiological hazard is less than during operation and is decreasing with the ongoing D&D project.

6.2.2.6.2 Post-operational phase (STA 5-B5.2)

The IAEA Technical Report No. 420 "Transition from Operation to Decommissioning of Nuclear Installations" explains that the "transition period from final shutdown to the start of dismantling (decommissioning) is termed the 'post-operational phase' in Germany". The post-operational phase (still governed by the operating licence) could last a period of years depending on the licence approval process for and the adequate pre-planning of the D&D project. The following tasks could be (partly) carried out within the post-operational phase:

- Unloading of the core and evacuation of the fuel elements to dry or wet storage facilities
- Post operational clean out
- Shut down of non-safety related systems
- Replacement of systems designed for NPP in operation
- Changes in the operational regime
- Changes in the maintenance cycles etc.
- Electricity and heat supply

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The tasks performed during the post-operational phase depend obviously on the decision for a direct D&D or safe storage of the NPP. R&D to support these activities post operational activities could reduce lifetime costs.

6.2.2.6.3 Plant characterisation (STA 5-B5.3)

The plant characterisation is an important task that has to be performed after final shut-down of the NPP in the post-operational phase, since it serves as a basis for an adequate planning of each D&D project. Such a plant characterisation aims to generate a comprehensive overview on relevant technical and radiological plant data such as:

- Room data (spatial - walls, volumes, surfaces)
- Components / inventory (weights, materials, numbers)
- Dose rate and contamination maps

Such a plant characterisation could be based on the existing construction or as-built drawings, but could also start with the preparation of as-built-data by means of e.g. laser scanning technology. After the as-built-conditions have been clarified, the structured compilation of all technical plant and radiological data has to be performed. Based on this inter alia the following analyses and visualisation tools would be helpful:

- 3D CAD model of the NPP with different layers such as contamination, dose rate or even activation
- Masses of different materials (concrete, steel, etc.) – (non-)activated or contaminated
- Surface area (that has to be decontaminated)
- Hazardous substances

The abovementioned analyses could support the following activities and tasks:

- ✓ **Evaluation of the waste and residuals production during the respective D&D project: the packing planning for containers or casks for final disposal and LLW and ILW could be performed at a very early stage (or even prior to start) of the D&D project; the load optimization could be an iterative process during the D&D project itself**

- ✓ **Waste management and tracking strategies: with the knowledge respectively the estimation of the amount of different wastes the management and the tracking of those could be adequately planned. Software for the management of waste is available and also under development/ improvement. On the hardware side state-of-the-art-technologies such as Barcodes are currently applied at NPP for tracking of waste in e.g. iron-barred boxes. RFID tags could possibly replace the barcodes, but have never been applied in the nuclear environment for such purpose.**

- ✓ **Dismantling and retrieval strategies: based on the 3D model of the plant and the technical and radiological data a pre-planning of selected dismantling tasks could be performed with different goals such as minimization of radiation exposure of plants personnel, reduction of waste, etc.**

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The development of one approach and tool(s) for a comprehensive plant characterization would support the pre-planning of D&D projects and result in increased efficiency, reduced radiation exposure to plants personnel and reduced decommissioning costs.

6.2.2.6.4 Decontamination and dismantling techniques (STA 5-B5.4)

Several reports on decontamination and dismantling technologies from associations including the IAEA[15] and the OECD-NEA[16] are available, but those were published already more than a decade ago. This reveals the demand that on the one hand the best-practise of decontamination and dismantling techniques applied in current D&D projects needs to be evaluated. On the other hand the technologies applied in the conventional industry should be analysed towards their potential use within a nuclear D&D project.

Materials to be dealt with vary in among others material (e.g. concrete or steel), dimensions (thin-walled piping vs. thick-walled vessels) and radioactive characteristics (contamination vs. activation). This requires a detailed investigation of the different tasks and the boundary conditions when choosing the most reasonable decontamination and dismantling technique.

6.2.2.6.4.1 Cutting optimization planning and techniques

Dismantling of nuclear facilities involves a wide range of cutting methods. Both mechanical and thermal cutting techniques are applied within D&D projects. While concrete is mainly cut by mechanical techniques (e.g. diamond wire saw), for steel also thermal processes for segmentation of the RPV were already applied in former D&D projects (refer to Stade NPP). While obviously improved methods e.g. relying on, for example laser technologies could result in more efficient dismantling of major plant components, the pre-planning of such tasks could address the minimization of waste for the final disposal. The pre-planning could involve the simulation of the cutting steps of e.g. the RPV and result in a final work step plan incl. the detailed container planning for the waste disposal.

6.2.2.6.4.2 Decontamination

The decontamination technologies could be divided into the following categories: chemical, mechanical, thermal and others. Which technology will be applied depends on different factors like location (in-situ treatment of walls vs. partly-segmented components in special workshops), material (concrete, metal) or degree of radioactive contamination/ activation. Many decontamination processes applied in current D&D projects rely on use of relatively large volumes of additives to remove contaminated material from surfaces. For steel components chemical or mechanical techniques like shot-blasting generate a huge amount of secondary waste that has to be treated and partly prepared for final disposal. Concrete surfaces are mainly treated with mechanical technologies to avoid changes in the nuclide vector due to thermal impact. The future development work should address among others the following issues:

[15] IAEA, TECHNICAL REPORTS SERIES No. 395: "State of the Art Technology for Decontamination and Dismantling of Nuclear Facilities, Vienna, 1999

[16] NEA, Task Group on Decontamination: "Decontamination Techniques Used in Decommissioning Activities", 1999

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- ✓ **Reduction of secondary waste (please refer also to •)**
- ✓ **Efficiency (decontamination factor, required manpower)**
- ✓ **Capacity: decontaminated surface per time period (m^2/h)**
- ✓ **Work load for personnel (radiation exposure, physical stress, etc.)**

6.2.2.6.4.3 Remote operations

In recent years significant advances have been made in remote operations and robotics in other industrial sectors, such as manufacture, aerospace and medicine. The sophistication and increasing miniaturisation of ROV's and remote tooling is likely to open up opportunities for their increased use in both routine maintenance operations and decommissioning activities. The nuclear industry is unlikely to lead this technology development but should take advantage of and exploit the significant advances in this field that have and are likely to take place in the near future. Especially manpower intensive tasks like decontamination of concrete surfaces should be subject to investigations of automation possibilities, or even autonomous approaches.

6.2.2.6.4.4 Reduction of secondary waste

As already described in section 4.2 decontamination technologies that do not generate secondary waste should be subject to future investigations. Furthermore, a considerable volume of low and very low level waste arises from PPE and other laundry waste associated with normal operation and decommissioning of reactors and other associated nuclear facilities. Ways in which these volumes can be reduced will have major benefits. Examples of such technologies are the use of material coveralls which can be dissolved in hot detergent solutions, resulting in liquid waste streams that can be cleaned up and safely discharged to the environment. A less obvious area relates to the further development of simulation software to optimise maintenance scheduling and the logistics of operations leading to a reduction in the number of man hours spent working in active environments. This in turn can reduce the volume and activity levels of the required PPE and tooling.

6.2.2.6.5 In-situ waste and effluent treatment (STA 5-B5.5)

The activities associated with decommissioning facilities and sites require treatment of effluents and waste arising, at a time when the site infrastructure is being dismantled. Modular, mobile plant that can be deployed in situ to support decommissioning are a potential solution to these needs; allowing projects to proceed without the need to build new site infrastructure and without impact on other site operations. Typical plant could include, decontamination units, effluent treatment modules, off gas treatment and immobilisation plant. R&D should be carried out on intensified processes for local waste and effluent treatment.

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6.2.2.6.6 Innovative Technology Development and Transfer (STA 5-B5.6)

New and innovative approaches to decommissioning and waste management offer potential step changes in performance; for example autonomous systems, in situ waste characterisation, thermal waste treatment etc. However, the introduction of new technologies requires both nuclearisation of technical approaches and demonstration. This demonstration builds confidence in technologies for the operators, decommissioners and regulators, in parallel with increasing the Technology Readiness Level.

This is often where innovation stalls. It is proposed that an international collaborative approach to support technology development and demonstration would help to overcome the barriers. This would involve:

- TRL demonstration facilities
- Technology evaluation/assessment
- Nuclearisation radiation tolerance
- Regulatory issues

6.2.2.6.7 Land remediation (STA 5-B5.7)

The management of contaminated land is a key issue during both plant operation and decommissioning. Site management strategies must be developed to effectively manage hazards associated any ground contamination. There are opportunities to identify best practice through the sharing of international practice at both a national and a site level. Furthermore, advances in the following fields could all contribute to enhancing site management strategies:

- ✓ **in situ monitoring technologies,**
- ✓ **large-scale measurement technologies,**
- ✓ **sampling strategies/statistical analysis,**
- ✓ **environmental modelling,**
- ✓ **risk based approaches to decision making,**
- ✓ **intervention land remediation technologies, and**
- ✓ **in-situ immobilisation.**

6.3 References

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7 TECHNICAL AREA 6 – Innovative LWR Design and Technology (TA6)

Technical area leader: Marylise Caron-Charles (AREVA-NPP)

7.1 Executive Summary

7.1.1 Scope

Nuclear energy currently provides around 30% of electricity in Europe, and Light Water Reactors (LWRs) are major contributors. The preparation of advanced LWRs benefiting from innovative technology will assess their position as key sources for electricity production, and this could valuably make the bridge throughout the 21st century between the ageing nuclear installations currently in operation and/or advanced Generation III ones, now under construction, and the fourth generation reactors proposed by ESNII.

The program proposed in the Technical Area 6 (TA6) aims at combining innovative technology and progress in the fields of safety and commissioning, operability, sustainability, economics and public acceptance. This encompasses currently operated reactors, and the development of new light water reactor concepts as well, that could feature improved sustainability with higher conversion ratio for a better use of uranium resources, or be sized for smaller generated thermal power while affording modular capacity.

7.1.2 Objectives

Innovation will be the key driver of the TA6 R&D projects that will address basic technology, methods, testing and computation capacities, with the objective of supporting the so called innovative light water reactor.

It is proposed to address the R&D needed work to support the design of innovative components with a view to achieving:

- Long Term Operation by design
- Safety by design
- Innovative component for reduced maintenance
- Enhanced economics

Knowing that new technology deployment at the industrial scale could be a long time duration process, the following time lines will be considered:

- Proposing evolutionary technology for mid-term application
- Developing new LWR designs such as with higher conversion ratio or small modular reactors, expected to be ready for commercial operation by 15 to 20 years
- Preparing breakthrough technology for a longer term future

7.1.3 State of the art

Synergies with all the other NUGENIA areas will be exploited for taking benefit from operating experience feedback from the current reactor fleet, prior to develop innovative technology that will

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result in LWR with improved performance. That will include requirements for safety and performance, reduced maintenance and long term operation, material ageing related issues, control and repair, and plant power upgrade, as well.

In addition to the currently operated LWRs, other concepts will be investigated. The concept of a high conversion light water reactor has went-on being studied over the 80’s in the aim at combining the advantages of LWR technology with the use of uranium – plutonium fuel, the achievement of high burn up and optimized nuclear fuel consumption. As for small modular reactors concepts, they are expected to have greater simplicity of design and reduced sitting cost compared to a current LWR. That results in a worldwide interest, and international programs lead by the IAEA and OECD / NEA, predicted them be a new part of the market segmentation several years ago. R&D programs are currently launched by the DOE in US and several vendors are already proposing concepts.

The evaluation of new concepts will seek for integrating any technology progress for making them match the objectives and challenges expressed in TA6. All the necessary stages for reactor component design and fabrication should be addressed. Safety issues will be considered at the early stage of the design. Finally, the overall performance of the component for the related reactor concept will be assessed using new methods.

7.1.4 Challenges

The roadmap is organized in five sub-areas spanning materials and component related technology, new light water reactor concepts, specific safety issues, scenario evaluation for the deployment of innovative LWR and public acceptance drivers for new build.

7.1.4.1 Innovative technology for reactor component design & construction

As a common theme for all LWR reactor concepts, this section will lead advanced and breakthrough technologies in materials processing areas such as surface engineering, nano materials, composite materials and hybrids, to obtain materials properties suitable for the design and construction of innovative nuclear components. Multi scale modelling will support the investigations. Closely related to new materials, new technologies for component fabrication will be investigated (e.g. powder metallurgy...). Subsequently, specific issues related to design and performance assessment will be addressed. It is proposed to develop a deep knowledge of local phenomena (corrosion, wear, fluid structure interaction, irradiation damage……..), using new testing methods and to improve numerical simulation methods. Finally, local phenomena simulation could be used as input to the development of engineering simulator tools which will assess the overall reactor system performance, including the evaluation of environmental impact.

R&D topics

- ✓ **Investigation into materials processing areas, surface engineering, nano-materials, composite materials and hybrids, to achieve properties suitable for high-performance nuclear components.**
- ✓ **Investigation into innovative component fabrication and assembly processes, in line with the development of new materials including multi material assembly, complex geometry, near net shape fabrication, powder metallurgy processing.**

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- ✓ **Assessment of innovative technology performance using in-depth knowledge of local phenomena leading to component degradation mechanisms: corrosion, wear, fluid structure interaction, irradiation damage…**
- ✓ **Analysis of modular construction techniques.**

7.1.4.2 Innovative LWR concepts such as: High Conversion ratio LWR, Small modular reactor

The purpose is to undertake all the necessary R&D work for preparing new Light Water Reactor concepts that could be ready for commercial operation within 15 to 20 years.

For example, high conversion ratio reactors feature improved sustainability with more efficient use of uranium resources and multi recycling capabilities of fissile materials; whereas small modular reactors offer a flexible and progressive approach to nuclear capacity optimization with limited infrastructure. An approach based on concept screening from the literature review is proposed for these new reactor models, from which the necessary R&D projects (reactor physics, cooling, compact component….) will be derived. These new concepts will foster and provide guidance for the development of innovative materials and component fabrication processes, as described previously in sub area1.

R&D topics

- ✓ **Concept screening**
- ✓ **Specific R&D topics derived from the previous screening: reactor physics, core cooling, compact component.**
- ✓ **Assessment of overall reactor system performance, in operating and accidental conditions**

7.1.4.3 Innovative LWR-specific safety approach

This part will mostly focus on innovative LWR specific safety issues. Indeed, the development of innovative materials, component fabrication processes, as well as the evaluation of new reactor concepts, could require specific safety approaches which should be integrated within the early stage of design operations. They should also be consistent with existing safety requirements or new safety requirements under development in the different European countries.

Reference to EU Directives establishing a common framework for the nuclear safety of nuclear installations, likewise to WENRA statements and to IAEA safety publications, will be made as much as possible. Safety approaches and the incorporation of passive system at the early stage of the design will be considered. In addition to conventional safety analysis, the resilience of these reactors to the issues highlighted by the Fukushima Daichi accident and covered in the EU stress tests specifications will be highlighted. Finally, the development of more sophisticated instrumentation and control systems for safety applications will be addressed.

R&D topics

- ✓ **Exploitation of pre-normative research results to implement the safety requirements, including site selection and evaluation.**

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- ✓ **Development of more sophisticated instrumentation and control systems for safety applications.**
- ✓ **Integration of the safety issues highlighted after the Fukushima Daiichi accident and by the EU stress tests specifications.**

7.1.4.4 Key success factors for innovative LWR deployment

Key success factors for innovative LWR reactors deployment will be investigated with consideration to the deployment of next power generation capacities, including Gen IV systems potentially beyond 2050 for the Sodium fast reactor, and the growing contribution of renewable energy sources. Flexible fuel cycle scenarios will be evaluated using a wide range of combinations of electricity sources for a transition period: current and advanced LWR technologies, Sodium fast reactors, other Gen IV systems, and renewable energy sources.. . In a view to assess the key role of LWR as a secure and low cost option for base load electricity generations within a regional fleet of mixed electricity sources, potential change for LWRs operation modes should be investigated in depth. Renewable energy sources are intermittent and should be complemented for avoiding grid disturbance. Flexible LWRs operations could be achieved using more load follow mode. But in the same time this will set out more stringent requirements for the plant life time management (e.g. usage factor for material, including fuel, component ageing, water chemistry practices, I&C…cost economics) that will call for technology development. Lastly, innovative solutions for minimizing the environmental impact of LWRs will be evaluated.

R&D topics

- ✓ **Evaluation of flexible fuel cycle scenarios relying on a wide range of combinations of electricity sources for a postulated transition period.**
- ✓ **Search for the enhanced operability of LWRs, including potential benefit for either load following or combined mode. Identify impact on plant life time management.**
- ✓ **Evaluation of innovative solutions for minimizing the environmental impact of LWRs.**

7.1.4.5 Public acceptance drivers for new builds

Different countries in Europe make a different evaluation of the cost and risk of nuclear energy versus its benefits. So that, some countries have a clear policy to maintain or increase the present nuclear installed power, some are planning its phase out, while other countries are considering initiating their nuclear industry in the coming years. These policies might change with time or as a function of the future evolution of their economic and political situation

It is proposed to address the rationale behind nuclear energy acceptance and respectively resistance by the public, taking into account the differences in energy policies and in the public awareness which exist among the different European countries transparency of information, education and of course, information on new safety concepts.

R&D topics

- ✓ **Identification of the main drivers towards public acceptance/ resistance for new build.**

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- ✓ **Harmonisation of the communication policies, taking into account the differences in the public awareness and acceptance, which exist among the European countries.**
- ✓ **Organization of the information dissemination.**

7.2 Sub Technical Areas (STA)

7.2.1 Innovative technology for reactor component design & construction (STA 6.1)

7.2.1.1 Scope and Objectives

Innovative technology RD&D for reactor components design and construction will be a cross-cutting aspect that will apply to all Light Water Reactor: existing and new designs. This will form “innovative LWR reactors”. Synergies will be set up for taking benefit from both the requirements of specific reactor concepts that will lead new technology developments and progress made in basic technologies for anticipating new solutions for nuclear application.

As an illustration, each reactor concept opens new possibilities for materials and component technology development. Conversely new materials and manufacturing processes will provide solutions for new component designs:

- Compact steam generators with high thermal performance materials
- Materials with properties gradients for fulfilling the interface conditions between two different environments
- Reactors operated in load following mode require component materials with enhanced resistance against wear and thermal fatigue
- New pressure vessel materials with high mechanical resistance for compact geometry, limited material quantities, and/or making easy welding operation
- High Conversion LWRs will have harder neutron spectra that could give rise to swelling issues
- It is proposed to address the RD&D needed work to support the design of innovative components with a view to achieving:
- Long Term Operation by design
- Safety by design
- Innovative component for reduced maintenance
- Modular construction techniques

All the necessary stages for reactor component design and fabrication will be addressed, while aiming for innovation on materials selection, component fabrication and non-destructive examination. Safety issues will be considered at the early stage of the design. Finally, the overall performance of the component for the related reactor concept will be assessed using new methods benefiting from the recent progress in testing capacities, physical modelling and numerical simulation.

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7.2.1.2 State of the art

The main material grades which are used in nuclear reactors components consist of: low alloyed steel, stainless steel and nickel base alloys. They were selected for their high mechanical resistance and their favourable behaviour in environmental conditions during long term operations. Nevertheless, they are susceptible to degradation mechanisms, since they cannot combine all the best functionalities. These materials are mostly used as monolithic structural materials, or may be coated. Recent studies on steels [O Bouaziz & al] have shown evidence that combining different materials and manufacturing routes may result in multi-functional materials. As an illustration, initial steel based materials may reach a new combination of strength and ductility when combined to other materials and formed with new processes.

Component manufacturing process and assembly technologies are tightly bound to material metallurgical properties. Large power generation components are commonly fabricated using conventional metallurgical methods, such as: casting, rolling, drawing, forging, extrusion, welding and heat treatment. Recent studies [EPRI] have highlighted the major attributes of powder metallurgy technology, especially high isostatic pressing, which can be used for small and large scale component. This includes: inspection, near net shape capability, elimination of rework or repair of large cast component, enhanced weldability, and potential assembly technology for different material grades using solid to solid or solid to powder assembly.

Following the component manufacturing process, finishing operations is the final stage before its installation. Conventional processes are: grinding, turning or milling. As such, they contribute to the modification of the surface characteristics and may induce potential detrimental effects during reactor operations.

Performance assessment in operating, incidental and accidental conditions is strongly supported by numerical calculation. Engineering Simulators consist of the integration of different technology components in a common hardware-software platform. They feature open systems, where the user can in principle ‘change anything’ concerning the plant and its systems and components. So the tool has an extensive application ‘window’ and can be used from the early phases of concept design. The overall tool makes it possible to develop the design of the NPP and to assess the reactor system performance. Nowadays Engineering Simulators include high fidelity codes normally used for design and safety analysis (such as RELAP and CATHARE codes).

7.2.1.3 Challenges

- Innovative material and processing
- Innovative component fabrication
- Innovative LWR specific issue on Design and performance assessment
- Modular construction techniques

7.2.1.4 Innovative material and processing

It is recognized that materials performance holds the key to fundamental advances in energy production systems. The purpose here is to investigate new and innovative materials for nuclear components of:

- Existing LWR models, and with consideration to conventional degradation mechanisms, as a basis : corrosion – wear – fatigue – irradiation – thermal ageing, and

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all coupled phenomena that could result in environmental assisted cracking and fracture
- New reactor models such as Small Modular Reactor, Higher Conversion LWR, etc….which will be demanding new requirements and call new component design and manufacturing process.

It is proposed to conduct investigations into the formulation of new materials up to proof of deployment feasibility, while considering their capability for nuclear component manufacturing.

Knowing that nuclear grade material assessment is a long time duration process, two time lines will be considered:

a. Evolutionary technology for mid-term application
b. Breakthrough technology for a longer term future

R&D topics

- ✓ **New nuclear materials with finely tuned properties**
- ✓ **Surface engineering**
- ✓ **Use of non-nuclear material with enhanced properties**
- ✓ **Innovative material processing**
- ✓ **Characterization of innovative materials properties**
- ✓ **Codes and standards**
- ✓ **Material multi scale modelling**

7.2.1.4.1 New nuclear materials with finely tuned properties

This part is mostly concerned with fabrication methods for producing innovative materials with multi functionalities. A combination between different materials to be selected for nuclear environments and the combination of different manufacturing processes will be envisioned. Specific requirements and properties of the materials to be developed: chemical composition, mechanical and metallurgy properties, resistance to corrosion, wear, thermal ageing, etc… will be selected in accordance with the component engineering requirements defined in the next sections. In this context, it is necessary to offer extensive knowledge in material and product development. The introduction of new experimental, computational techniques or data set, those help to accelerate the development of materials and processing, are considered. As such, this will be a common area for all reactor component models.

7.2.1.4.2 Surface engineering

Methods will be developed for engineering material surfaces with appropriate parameters in order to enhance its resistance against degradation mechanisms in environmental conditions

- Coating process, example:

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- electro-deposition from innovative Ionic Liquids (IL). The development of nano-structured coatings is a current research topic due to their significantly enhanced properties. In particular, electrochemistry has a special role in producing a variety of nano-structured coatings. Ionic Liquids can be used as an electrochemical medium for cost effective and environmentally friendly processing methods for the electro-deposition of multi-functional materials operating in extreme conditions against corrosion and high temperature oxidation.

- Nano-structured surface application for stress corrosion cracking resistance or heat transfer improvement, examples:
  - surface nano-structuring of steels by Plasma Electrolytic Deposition (carburizing, nitro-carburizing)
  - Mechanical attrition
- Peening process and effects
- Plasma-assisted electrolytic processes (PE) will be employed to enhance the inherent corrosion resistance and performance of the steel:
  - by achieving the desirable fine crystalline microstructure of steel;
  - by modifying its chemical composition at the nanometer scale;
  - by obtaining strengthened nanophases (carbides, nitrides, carbonitrides, intermetallics)

7.2.1.4.3 Use of non-nuclear material with enhanced properties

Review of materials already used in other industrial application which could fit NPP environment

- Non-nuclear material scoring high mechanical resistance for pressure vessels of new models (reactor, steam generator, pressuring system)
- wear resistant hard facing material without cobalt

7.2.1.4.4 Innovative material processing

Starting from conventional materials which are used in nuclear reactors, RD&D work will seek to design new materials with relevant combined and functional properties in addition to high mechanical resistance. This will proceed through:

- Selection of materials composition
- Volume fraction and spatial distribution of alloyed materials
- Structure of the final product e.g. new microstructure versus new macrostructure
- Combination of forming processes; e.g. use of powder metallurgy, bounding, deposition techniques, etc…

Development of new material processing methods for tailored material properties which are not achievable with conventional manufacturing routes will address:

- Multi-layer material e.g. high resistance to corrosion, or crack propagation

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- Steel-based multi-material composite e.g. combining high mechanical stress and other functionality
- Architectured material e.g. enhanced elasticity
- New alloyed material e.g. resistance against irradiation damage, swelling, creep
- Bulk nano material

7.2.1.4.5 Characterization of innovative materials properties

Basic properties for assessing innovative material performance in NPP environment could be studied in collaboration with NUGENIA areas 3 & 4, when possible. The work will focus here on innovative material specific qualification

- Interface characterization of multilayer material
- Non-destructive examination techniques
- Characterization of architectured materials
- Microstructure of new alloyed materials
- Metallurgical properties and domain for use

7.2.1.4.6 Codes and standards

The use of new materials for nuclear application will call for pre normative research, addressing their properties and characteristics in environmental conditions, in order to provide information for preparing the nuclear codes and standards update. The methodology will be set up with direct input from NUGENIA 7 on harmonization. It is worth noting European committees have been very active for several years, providing support to standardization and R&D links (e.g. CEN-CENELEC). Joint work with R&D organizations, utilities, manufacturer and vendors will be valuable for clearly establishing the orientations.

7.2.1.4.7 Material Multi scale modelling

Extend multi scale modelling tool such as PERFORM, for conducting preliminary investigation on innovative material with finely tuned properties

- Evaluation of new alloyed materials for enhanced properties under irradiation: Irradiation Assisted Stress Corrosion Cracking, swelling, creep……
- Guidance on new microstructure and /or new architecture to be developed

7.2.1.5 Innovative component fabrication

The purpose aims at investigating or updating new fabrication methods for ensuring short construction duration, assessing reduced maintenance and long term operation for the component. Innovative fabrication processes should be investigated too, for making use of innovative material at the component scale. They could result from a combination of conventional processes. Lastly, new reactor models could require new assembly technologies for their components.

Using the same approach as for innovative material, this chapter will be a common part for all reactor concepts.

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New research topics:

- ✓ **Manufacturing technology**
- ✓ **Specific non-destructive examination**
- ✓ **Numerical simulation**
- ✓ **Codes and standards**

7.2.1.5.1 Manufacturing technology

The purpose here is to provide key RD&D elements for supporting the development of new or innovative manufacturing technology. More emphasis will be put on processes appropriate for large scale fabrication of components using new materials (§ 1.1) selections and for new or complex geometry.

R&D program beyond the state-of-the art:

Component fabrication technology

- Deep investigation on conventional fabrication process and possible use for component designed with innovative materials
- Alternatives to conventional processing (forge, cast…)
- Near net shape fabrication
- Powder metallurgy and Hot Isostatic Pressing

New assembly technology for complex geometry, or for multi material assembly :

- Alternative to conventional welding processes
- Hot Isostatic Pressing
- Diffusion welding
- Laser processes for layered structures

Optimize surface finishing operations

- Deep investigation on conventional processes and possible use of components designed with innovative material
- Effect on surface microstructure properties and subsequent resistance in environmental conditions
- Impact of surface cleanness and evaluation of a cleanness index beyond which detrimental effects could occur.

7.2.1.5.2 Specific Non Destructive Examination

When necessary, update or develop new NDE methods for fabrication control of innovative materials / components (optical methods, etc…)

- Interface control for multi material – new coating processes….
- Control of new micro or macro structured materials

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7.2.1.5.3 Numerical simulation

Develop new computational capabilities for modelling the impact of the fabrication and assembly process parameters with conventional, combined or innovative processes, on the resulting material microstructure and subsequent metallurgical properties. Develop and/or use the appropriate testing facilities

- modelling the grain size of bulk materials
- assembly of new alloyed materials and assessment of interface properties
- interface behaviour for multi –material composites
- potential segregation of impurities

7.2.1.5.4 Codes and standards

Likewise for innovative materials and processing, pre-normative R&D work will be mandatory for the deployment of new nuclear manufacturing technologies. The main orientations will be defined specifically for the targeted component application, in line with NUGENIA area 7 on harmonization.

7.2.1.6 Innovative LWR –specific issue on Design and performance assessment

In continuity with the previous sections 1.1 and 1.2, that aim at developing new material and manufacturing technologies, it is proposed here to assess performance and long term operation by design of new component generations.

Innovative materials and fabrication methods for the components of current LWR or new reactor models (with Higher conversion factor – Small Modular Reactor, etc…), will first be assessed against conventional degradation mechanisms, with reference to the reactor model for which they are designed.

Emphasis will be put on a deep investigation of local phenomena, especially fluid to structure interaction. Existing methods will be updated and if this is deemed necessary, new testing facilities and computational tools will be developed. It is proposed then to integrate the results in a proper way, into an overall platform, or engineering simulator. This will assess the reactor system performance.

New research topics:

- ✓ **Assessment of innovative solutions for the reactor components**
- ✓ **Development of new testing capacities and computational methods for screening all design options through a detailed investigation of local phenomena**

7.2.1.6.1 Assessment of innovative solutions for the reactor components

Revisit conventional degradation mechanisms which result in maintenance and outage operations and validate solutions using innovative materials and components:

- ➢ Steam generator fouling
    - o new materials and or coating
- ➢ Radioactivity source term management
    - o material without cobalt

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- high performance filtration systems and/or avoid particle transport

- Preventive action during maintenance
  - Protective solutions against corrosion
- Plant start-up and shut down operations -
  - Improve component resistance in transient regime chemistry for shortening operations
- Plant operation in load following mode
  - improve component resistance against thermal fatigue and wear….enhanced usage factor
- Power plant upgrade
  - impact of higher reactivity and operating temperature on corrosion - fatigue in primary circuit
- Vibratory and seismic behaviour
  - Extend the elasticity limit of the materials and/or system

7.2.1.6.2 Development of new testing capacities and computational methods for screening all design options through a detailed investigation of local phenomena

For complementing conventional numerical simulation and approach, it is proposed here to focus on local phenomena, which may point out weakness of the design options. A deeper understanding will result in best estimate margins while assessing the performance in the full range of operating and safety conditions.

- Fluid structure interaction
  - Mixing zone of two turbulent fluid at different temperatures at the junction between two circuits: potential thermal stratification and subsequent wall thermal fatigue
  - Instantaneous T, P and V field mapping at the wall boundary conditions – innovative measurement techniques and advanced calculation code validation: LES – DNS…
  - Flow induced vibration, especially for compact reactor designs in which operating velocities are relatively high - Advanced methods and coupling thermal hydraulic to structural calculation codes.
- Flow in singularities
  - Accurate evaluation of the excitation source generated by turbulent flow in complex geometry - vibratory behaviour
  - Particle flow in complex geometry and clogging risk as in Steam Generators
- Particle to material surface interaction
  - Deep investigation of colloid particle to material interactions, e.g. characterization techniques – modelling tool
  - Validate new material and new design option for avoiding particle deposit and ensuring e.g. Steam Generator reduced maintenance

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7.2.1.7 Modular construction techniques

In collaboration with sub chapter 6.2.2 on small modular reactors, it is proposed to seek to:

New Research topics

- ✓ **Incorporate modular construction aspects: the way to do it, and to what extent: e.g. ancillary circuits, reactor building…**
- ✓ **Identify safety, construction, and economical limits ...**
- ✓ **Civil work: In addition to NUGENIA area 4, and for the purpose of modular construction technique, it is proposed to look for new concrete-related technologies**
    - o **Concrete ageing in steel-concrete-steel in case of modular construction…**
    - o **Investigate new methods for improving constructability e.g.: design and diameter of rebars , pouring of concrete**

7.2.2 Innovative LWR concepts including: High Conversion LWRs, Small Modular Reactors, etc….. (STA 6.2)

7.2.2.1 Scope and objectives

The purpose is to provide all the necessary RD&D work for preparing new Light Water Reactor models that differ from current Gen III LWRs, and that could be ready for commercial operation by 15 to 20 years. For example, they will feature improved sustainability with a better use of uranium resources and multi-recycling capabilities of fissile materials, or be sized for smaller generated thermal power and modular construction techniques, etc…….

The development of these new reactor models will foster and provide guidance for the development of innovative materials and components, as well as innovative fabrication processes, as described in the previous section §1. Conversely, having innovative materials and fabrication processes on available will make it possible to design and fabricate new components. In other words, a two way communication process needs to be established between materials and component related technologies, and the reactor concept requirements.

For each reactor model, the expected properties and characteristics of the fuel to be designed will be clearly established here, and the RD&D work will be split between NUGENIA areas 5 and 6. Namely, fuel material related issues such as behaviour in environmental conditions, fission product release, pellet to clad interaction, spent fuel management, etc… will be addressed in NUGENIA area 5. Nevertheless, all the necessary RD&D work on innovative and transverse technology will be addressed in NUGENIA area 6.

Finally, innovative LWR specific safety issues as described in section 3, will be integrated as much as possible in the preliminary design features.

7.2.2.2 State of the art

The concept of high conversion light water reactors has been studied intensively during the 80’s and was driven by the motivation to combine the advantages of LWR technology with the usage of Pu as fuel, high burn up and low nuclear fuel consumption.

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In other words, the development of High Conversion LWRs will aim at improving LWR sustainability as it offers:

- Maximization of uranium utilization
- Multi-recycling of fissile material, plutonium and reprocessed uranium (MOX)

There is revival of interest in small and simpler units for generating electricity from nuclear power, and for process heat. Modern small reactors are expected to have greater simplicity of design, economy of mass production, and reduced sitting costs Small modular reactor approach features a flexible and progressive means of nuclear capacity optimization, with limited infrastructure. As such, Small Modular Reactors are attractive for non-nuclear countries which are willing to use nuclear energy for either electricity production or process heat applications. They are attractive too for remote districts which cannot be connected to the national grids of nuclear countries.

7.2.2.3 Challenges

- High conversion LWRs
- Small modular reactor

7.2.2.4 High Conversion LWRs

New research topics:

- ✓ **Progress beyond the state of the art**
- ✓ **Concept screening**
- ✓ **Core design with high conversion ratio**
- ✓ **Reactor core cooling**
- ✓ **Investigation of various separate thermal hydraulic phenomena**
- ✓ **System integration**

7.2.2.4.1 Progress beyond the state of the art

High conversion LWR cores must be designed very carefully with respect to neutron economy through the use of a heterogeneous core structure, modification of fuel assembly geometry, optimization of the fuel composition and other innovative features. This raises technical challenges, namely: HCLWR must be positioned between a sufficiently high negative void coefficient, simultaneously providing a tight lattice core which can be cooled for obtaining a high conversion ratio.

The following major features will serve as reference point for performing the necessary R&D developments:

- LWR with conversion ratio up to CR= 0.9 and far above CR= 0.6 as current LWR
- Use of U-Pu fuel in a tight fuel assembly lattice for reducing neutron moderation

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- Epi-thermal neutron energy spectrum
- Flat reactor core with high power density
- High burn up fuel

As a first evolutionary development step, and for mid-term industrialization (around 15 years), the core should fit into existing or closed to LWR reactor pressure vessels. With consideration to longer fuel cycle, lower moderation rate and harder neutron spectrum, while keeping high safety standards, conceptual mechanical design of the reactor core will be proposed, as well as the necessary optimization of the structural components and subsequent innovation routes.

7.2.2.4.2 Concept screening

First, a state of the art review will serve for screening and evaluating high conversion light water reactor concepts which have been studied worldwide. Then, alternative solutions will be proposed.

- Consideration of flexible fuel cycles including thorium use and option for minor actinides recycling
- Establish main requirements, advanced and innovative properties for the material and structural component design, e.g.:
    - Flow induced vibration
    - Corrosion and radiation damage
    - High burn up

7.2.2.4.3 Core design with high conversion ratio

The progress beyond the state of the art can be found in the fields of neutron physics and thermal hydraulics calculation codes. Extensive of NURESIM tools will widen their spectrum of applicability.

- Code validation and verification for reactor physics calculation tool
- Robust nuclear data and advanced reactor physics codes
- Safety concepts and challenging cases
- Optimization of the pressure drop over the core

7.2.2.4.4 Reactor core cooling

The tight lattice core of high conversion core, that differs from current LWR, makes the cooling very specific and raises new challenges in thermal hydraulic evaluation

- Analysis of existing testing facilities and construction of new facilities with updated instrumentation

7.2.2.4.5 Investigation of various separate thermal hydraulic phenomena

In addition to core design related features, other thermal hydraulic phenomena will play a key role. Detailed investigations in different geometry scale and configuration than for conventional LWRs need to be carried out.

- Two phase flow and Critical Heat Flux in tight lattice under normal and accident conditions

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- Transverse mixing and non-uniformity of heat transfer in tight lattice FA
- Effect of new spacers
- Flow induced vibrations in tight lattice rod bundles as a result of flow pulsation.

7.2.2.4.6 System integration

It is proposed here to assess the overall reactor system performance, setting up all the components and screening their design options against operating and accidental conditions:

- Performance of heat removal system
- Extended use of passive systems and/or self-operated safety systems
- Reactor core, internals, and pressure vessel
- Steam generator
- Pumps and piping
- Specific water chemistry

7.2.2.5 Small modular reactors

The purpose here is to address RD&D work on small modular reactors concepts using light water reactor technology, and for mid-term commercial operation (near 15 years). Small compact reactors may be sized from 45 MWe up to near 300 MWe. These elementary modules are designed along with a modular construction approach for making it possible to combine them and incrementally extend the power capacity of the overall plant. Their compact and modular design means they could be made in factories and transported to the generation sites, offering economy of scales and reducing both capital costs and construction time.

New research topics:

- ✓ **Progress beyond the state of the art**
- ✓ **Concept screening**
- ✓ **Core design for long cycle life and inherent safety features**
- ✓ **Investigation of various separate thermal hydraulic phenomena**
- ✓ **Component and system integration**
- ✓ **Modular construction techniques**

7.2.2.5.1 Progress beyond the state of the art:

Among various concepts, integral design of self-contained pressurized water reactors and steam generator sets, will offer the possibility of modularity since each unit could be factory-made, and then delivered on the site. This means small size and compact components, compared to large power LWR, need to be designed, for building an integrated reactor system in which all primary components could be located inside the reactor pressure vessel.

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Small modular reactors will call for new challenges in technology development. For example, a plate steam generator, or helically coiled, rather than a conventional tube concept, could be a considerable advantage for compactness. Associated to this technology is the welding process using e.g. diffusion bonding process, and of course the material selection. These processes are not yet used in nuclear application and this will require new RD&D work.

SMRs can incorporate a high level of passive or inherent safety in the event of malfunction. As an illustration, integral design could result in eliminating large break.

Regarding the fuel cycle technology, it is envisioned to look for possible long fuel cycle and high burn-up using high enriched fuel, and the possibility for out-site refuelling.

7.2.2.5.2 Concept screening

First, a state of the art review will serve for screening and evaluating Light water cooled small modular reactor concepts which are being studied worldwide. Then, alternative solutions will be proposed.

- Establish main requirements, advanced and innovative properties for the material and structural component design, for reaching:
  - Compactness
  - Integral reactor design
  - Long fuel cycle
  - Re-fuelling on-site versus off-site

7.2.2.5.3 Core design for long cycle life and inherent safety features

Beyond the state of the art progress will be achieved through numerical simulation with updated tools: mainly in thermal hydraulics, reactor physics and neutron physics data, for application to small LWR.

- Code validation and verification for reactor physics calculation tool
- Irradiation behaviour of cladding (specification only)
- Numerical codes for description of natural convection phenomena including large pools

7.2.2.5.4 Investigation of various separate thermal hydraulic phenomena

In addition to core design related features, other thermal hydraulic phenomena will play a key role. Detailed investigations in different geometry scales and configurations than for conventional LWRs need to be carried out.

- Natural and mixed convection conditions
- Two phase flow and CHF
- Flow stability
- Flow and heat transfer in large pools
- Boiling heat transfer of steam generator tubes or plates
- Flow induced vibration in compact SMR designs

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7.2.2.5.5 Components and System integration

It is proposed here to assess the overall reactor system performance, setting up all the components and screening their design options against operating and accidental conditions:

- Design and performance of compact steam generators
- Performance of Heat removal systems
- Validation and confirmation experiments for components and systems
- Economic evaluation

7.2.2.5.6 Modular construction techniques

The possibility of extending the installed capacity through the combination of elementary reactors is one of the basic principles of "small modular reactors"

- Identify safety, construction and economic limits
- Civil work
- Operability of the whole set of modules

7.2.3 Innovative LWR specific safety issues (STA 6.3)

7.2.3.1 Scope and objectives

Knowing that safety approaches and methods are mainly addressed in the NUGENIA area 1 on plant safety and risk, this part will mostly focus on innovative LWR specific safety issues. Indeed, the development of innovative materials, component fabrication processes, as well as the evaluation of new reactor concepts, may require specific safety approaches which should be integrated within the early stage of design operations, and should be consistent with safety requirements existing or under development in the different European countries. The work will be carried out in accordance with EU Directive 2009/71/EURATOM of June 25, 2009 which establishes a Community framework for the nuclear safety of the nuclear installations.

7.2.3.2 State of the art

As expressed within NUGENIA area 7 on harmonization, participation of European organizations such as ENSREG, WENRA, and ETSON could be valuable for making a set of recommendations for the new European nuclear safety architecture. Development of innovative LWR design, whatever innovation feature, could be a good opportunity for applying harmonized methodologies related to licensing process and safety assessment requirements.

Lastly, several references on current activities and existing international working groups are listed in Appendix

7.2.3.3 Challenges

- New safety approaches & systems at the design stage
- Development of safety requirements, criteria and rules for passive systems safety assessment
- Development of more sophisticated instrumentation and control systems for safety applications

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7.2.3.4 New safety approaches & systems at the design stage

New research topics:

- ✓ **New safety approaches & systems at the design stage**

In terms of safety, at the design stage, the innovative LWR designs should present enhanced independence of Defence in Depth levels and demonstrate extended resilience compared with actual Gen II&III reactors for different types of hazards and especially for the one covered in the "EU Stress Tests specifications".

For HCR and SMR LWRs, the means to maintain the three fundamental safety functions (control of reactivity, fuel cooling, and confinement of radioactivity) and support functions (power supply, cooling through ultimate heat sink) should be reviewed.

For short and mid-term, the focus of the studies has to be placed on the resilience of these reactors to the following issues in light of the Fukushima event and of the "EU Stress Tests specifications" as mentioned below:

- Initiating events
  - Earthquake
  - Flooding
- Consequence of loss of safety functions from any initiating event conceivable at the plant site
  - Loss of electrical power, including station black out (SBO)
  - Loss of the ultimate heat sink (UHS)
  - Combination of both
- Severe accident management issues
  - Means to protect from and to manage loss of core cooling function
  - Means to protect from and to manage loss of cooling function in the fuel storage pool
  - Means to protect from and to manage loss of containment integrity

Issues as stated in b) and c) should not be limited to earthquake and tsunami as in Fukushima: flooding should also be included regardless of its origin. Furthermore, external hazards should be added too.

The approach to assess the resilience to these hazards should be essentially deterministic: when analysing an extreme scenario, a progressive approach shall be followed, in which protective measures are sequentially assumed to be defeated. However Probabilistic and Risk informed approaches should be also used to complement the deterministic safety approach.

For the innovative LWR design, the extended use of passive systems and/or off-operated systems, and optimization of safety systems should especially show benefits in the field of Severe Accident Management and Emergency Plan Preparedness and demonstrate extended grace period for example in case of station black out.

All these points would need to be developed in close collaboration with NUGENIA Areas 1 and 7 which look at Risk Assessment and Harmonization at the level of:

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- Defence in Depth approach
- Deterministic safety demonstration
- Probabilistic Safety Assessment
- Risk informed approach
- Safety Margins

7.2.3.5 Development of safety requirements, criteria and rules for passive systems safety assessment

New research topics:

- ✓ **Development of safety requirements criteria and rules for passive systems safety assessment**
- ✓ **Innovative LWRs designs present an opportunity to develop a new generation of power plants with enhanced safety performance. This should be extended not only with simpler components, fewer dependencies, and less stringent operation/maintenance requirements. Some designs incorporate inherent safety features such as higher thermal inertia. In some cases, fast moving accidents such as Loss of Coolant Accidents (LOCAs) have been eliminated, and transient response is more benign. Some designs present less of a challenge in the severe accident arena and have favourable source term characteristics. These differences can ease the burden on operating staff and create opportunities for more effective accident management and should therefore result in a more efficient licensing process than that used for current LWR designs.**
- ✓ **As an example the safety requirements and safety assessment principles associated with the following points should by covered by collaborative or R&D projects on a mid-term or long term basis:**
  - **Credibility of passive systems activation and load-up to required capacity**
  - **Safety and reliability assessment of the capability of passive system to perform the assigned function,**
  - **Dependence from external energy sources for initialisation and execution of the assigned function.**
  - **Assessment of different phenomena that could lead to the loss of assigned function,**
  - **Uncertainties and Safety margins associated with passive systems**
  - **Methodology for the reliability evaluation of passive systems and its integration into PSA**
  - **Specific requirements for Severe Accident Management and Emergency Plan Preparedness**

Another aspect to be taken into account would be to review the safety requirements linked to the modular construction of innovative light water reactors (especially SMRs) and the related safety

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requirements linked to the qualification of equipment in a factory instead of doing it on site. Lastly, harmonization on site licensing methods should be evaluated for anticipating the possible extension of nuclear capacity to be installed.

7.2.3.6 Development of more sophisticated instrumentation and control systems for safety applications

New research topics:

- ✓ **Development of more sophisticated instrumentation and control systems for safety applications**

The most challenging regulatory issues for a new reactor design tend to centre on core and reactor coolant design, new materials applications, new system configuration, accident analysis, and containment. In addition, the conduct of Probabilistic Safety Assessment (PSA) and severe accident analysis can present new challenges.

In particular for innovative LWRs designs development the usage of more sophisticated instrumentation and control systems could represent a challenge both in term of safety application and during the licensing process. The following point would deserve to be further analysed through collaborative R&D projects on a mid-term or long term basis:

- ✓ **Electronic and programmable sensors,**
- ✓ **Data transmission for safety: optic fibre, network, wireless technologies,**

7.2.4 Key success factors for innovative LWR reactor deployment (STA 6.4)

7.2.4.1 Scope and objectives

Electricity European demand is expected to increase between 25% and 35% by 2050, depending on the scenario. Nuclear energy is one of the SET plan highlighted technologies, and fulfils the main EU energy challenges as expressed by the energy roadmap 2050. As such, it will remain a key contributor to electricity generation.

It is proposed here to investigate key success factors for innovative LWR deployment, with consideration to the deployment of the next power generation capacities, including Gen IV, and the growing contribution of renewable energy sources (wind, solar….). The rationale behind the national policies on nuclear energy that differ among the European countries will not be addressed here, but they will be considered in the scenario evaluation.

7.2.4.2 State of the art

For the LWR models: current Gen II, under deployment Gen III and future generation of Gen III will likely be the main technology of this century, since they ensure.

- ❖ Low carbon technology
- ❖ Security of energy supply : Available and mature technology for large scale electricity generation which is capable of meeting the demands of a growing market without any importation

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- ❖ Competitiveness: low and stable cost thanks to a low impact of the fuel cycle on electricity production cost, and affordable energy resource

The Gen IV generation is a more sustainable concept with a closed fuel cycle which could be deployed by around 2050. Core designs with fast neutron spectrum enable a better use of fissile materials, fuel material multi-recycling, and possible actinide burning capacity. This will result in the minimization of natural uranium consumption and minimization of waste, especially long life minor actinides.

Lastly, renewable resources will be increasingly deployed. For the time being they are facing intermittency issues and new technologies such as for energy storage still need to be developed.

7.2.4.3 Challenges

- Flexible fuel cycle
- Enhanced operability
- Environmental impact

7.2.4.4 6.4.2 Flexible fuel cycle

Within such an evolving context, and considering different technologies for electricity generation will be progressively available, and possibly jointly operated, scenario evaluation is a powerful tool for assessing the deployment strategy of a flexible fuel cycle.

The European Nuclear Energy Forum (ENEF) is already committed and prospecting on scenario evaluation dealing with the energy demand evolution. They will be used as input. The purpose here will focus on the nuclear fuel cycle options which could be envisioned for a successful integration of innovative LWR design, and the assessment of their capabilities to make the bridge between the current LWR fleet and a mix of energy sources for electricity generation.

New research topics:

- ✓ **Evaluation of European fuel cycle scenario for the transition period from current LWR to energy mix**

7.2.4.4.1 Evaluation of European fuel cycle scenario for the transition period from current LWR to energy mix

Fuel cycle scenario will be evaluated using a wide range of combinations of mixed energy sources. Consideration will be given to the national policies that may differ among the European countries, notably regarding nuclear waste management, recycling options, and the stake of renewable energy sources.

LWR operation and innovative design deployable in about 15 years:

- ❖ Current LWR technology
- ❖ Long fuel cycle, High Burn Up with more than 5% enrichment fuel, plutonium mono recycling in MOX

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- ❖ High Conversion LWR and capacity for plutonium multi recycling
- ❖ Small modular LW reactor with adaptable electricity generation capacity
- ❖ Possible use of Thorium in LWR fuel cycle as complement to further improve the U/Pu fuel cycle

Gen IV Systems [ESNII]

- ❖ Sodium fast cooled reactor as the reference technology with an expected deployment beyond 2050
- ❖ Plutonium availability for SFR deployment
- ❖ Multi-recycling capacity of Plutonium and possibly of minor actinides
- ❖ Alternative Gen IV systems
- ❖ European scenarios accounting for the share of recycling facilities (feedback from PATEROS and REDIMPACT projects will be used)

Renewable energy sources

- ❖ Wind , solar, other alternative systems
- ❖ Intermittency, necessary complementarities and electricity generation capabilities

The scenario outcome will provide key parameters for nuclear energy generation notably: mass inventory, fissile material balance depending on reactor fleet deployment, evaluation of ultimate waste arising, impact on repository design and foot print.

Emphasis will be put on high conversion LWR capability in plutonium multi-recycling, as well as on the benefit offered by small modular reactors for extra or incremental power capacity.

Technical and economic evaluations will be defined by the participants coming from various European countries.

Such studies will provide inputs for a realistic representation of a possible European nuclear fleet evolution, of the associated waste generation and resources consumption, as well as the potential spent fuel liabilities that may arise from evolution and perspective in different European countries.

7.2.4.5 Enhanced operability

Enhanced operability will be considered along with two objectives that could be separate or complementary: i) for the conventional plant operation and ii) for facilitating LWR integration within a regional fleet of mixed electricity sources.

New research topics:

- ✓ **Potential for flexible operation**
- ✓ **Innovative plant operations**

7.2.4.5.1 Potential for flexible operation

In a wide range of scenarios, nuclear energy is currently recognized as the least cost option for base-load centralized generation, even in low price scenario [ref: ENEF competitiveness SWOT report].

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Nevertheless, given the progressive introduction of renewable energy sources, which are intermittent, potential change for electricity production modes should be investigated in depth, including in load following mode, or other combined mode. Technical and economic issues will be addressed in section 4.1.

The outputs from the fuel cycle scenario evaluation (section 4.1) will be used as input for assessing the potential benefit of improving LWR load follow capabilities.

All the necessary technologies for variable and/or combined modes operations will be developed:

- Ramp up or ramp down rate and Set up or down with reference to the nominal power I&C update

As a common issues,, the impact of variable modes operation on the usage factor of LWR plant will be clearly evaluated:

- Specification for materials and component design including fuel assembly
- Impact on fuel management (in collaboration with TA5)
- Routes for innovation

7.2.4.5.2 Innovative Plant operation

For simplification of the plant operations: start-up, shut-down, power and safety, and other issues generated by variable mode operation, innovative technologies will be developed and implemented for plant control, monitoring system and I&C.

RD&D projects will be undertaken in close collaboration with NUGENIA area 3 on core and reactor performance and for specific innovative LWR related operations,

- Upgrading Human system interface (in collaboration with MOTION project)
- Digital I&C
- Wireless technology
- Advanced signal transmission and analysis
- Nano sensors
- Radiation resistant electronics

7.2.4.6 Environmental impact

New research topic:

- ✓ **Reduce the environmental impact of water cooled reactors**

To more effectively reduce the environmental impact of water cooled reactor, different routes will be investigated: water resources, radiological release, waste generation, etc…. Feedback experience from current LWR operation will be integrated in the early design stage of innovative LWR concepts,

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such as high conversion and small modular light water reactors. Geological and climate features of the sites could lead specific technology development

Input from NUGENIA area 3 on core and reactor performance will pave the way towards new solution for:

- Development of advanced cooling technologies
    - reducing water consumption removed from the sources ; e.g. use recycled water with appropriate chemistry
    - minimizing temperature modification at the source
- Develop the concept of “operating dry” (atmospheric or heat sink)
- Assessment of ALARA radioactive release in the vicinity of the plant
- Develop solutions for decreasing liquid waste generation.
- Consideration to nuclear power plant dismantling operations that will result in waste production too (in collaboration with NUGENIA area 5).
    - Integration of dismantling constraints at the early stage of the design.

7.2.5 Public acceptance drivers for new build (STA 6.5)

7.2.5.1 Scope and objectives

Nuclear energy characteristics perfectly match the requirements of reduction of greenhouse gas emissions, security of supply and competitiveness. However, different countries in Europe make a different evaluation of the cost and risk of nuclear energy versus its benefits, and we can find in Europe countries that have a clear policy to maintain or increase the present nuclear installed power, others planning to reduce or phase out the electricity generation from nuclear fission, and countries considering to initiate in the coming years their nuclear industry. In addition, theses policies might in some countries change with time or as a function of the future evolution of its economical and political environment. Independently of the political decision of continuation or phase out nuclear energy, all countries using nuclear energy to generate electricity are facing the question of the final management of its spent fuel and other high level radioactive wastes that is a key issue for public acceptance.

It is proposed here to address the rationales behind nuclear energy acceptation by the public, notably for new builds deployment.

7.2.5.2 Challenges

- Main drivers
- Weighting the drivers versus European country specificity
- Vector of information
- Communication on new safety concepts

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7.2.5.3 Main drivers

New research topics:

- ✓ **Identifying the main drivers towards public acceptance for new build**
- ✓ **Checking the drivers versus European country specificity**

7.2.5.3.1 Identify the main drivers towards public acceptance for new build

- ✓ **public perception of the energy strategy**
- ✓ **Security of supply - cost - stability**
- ✓ **transparency of safety authorities**
- ✓ **correctness of government policy and coherence of decisions and actions**
- ✓ **involvement of (and dialogue with) local populations interested in the plant sitting**
- ✓ **public sensitivity toward environmental aspects**
- ✓ **influence of specific industrial interests on the public**
- ✓ **influence of international political pressure on the public**

7.2.5.3.2 Weighting the drivers versus European country specificity

According to the national policies that may differ among the European countries, the drivers towards public acceptance for new build need to be weighted. European policy will provide key elements for guidance, in presenting nuclear power as a fundamental part of the energy mix.

7.2.5.4 Dissemination of information

New research topics

- ✓ **Aahrus convention application to nuclear**
- ✓ **Correct information dissemination through schools and media**
- ✓ **Communication on new safety concepts**

7.2.5.4.1 Aahrus convention application to nuclear

- Transparency of information
- Consultation of the public
- Possibility of litigation

7.2.5.4.2 Correct information dissemination through schools and media

- manage popularization by influential experts and institutions
    - o manage ad-hoc courses for school teachers

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- create familiarity with fundamental energy concepts, engineering approach and environmental aspects of energetic issues
- present comparisons among the different power generation technologies in terms of:
    - o impact of emissions along the complete cycle
    - o immediate accident consequences in the cycles of construction materials and fuel transport
    - o immediate accident consequences in a generation plant
    - o environmental impact of the waste
    - o low dose effect

7.2.5.4.3 Communication on new safety concepts

New safety concepts should be publicly claimed

- the accident at Fukushima is seen as a realistic possibility in a modern and advanced countries
- the complexity of large nuclear power plants frightens people
- need to make people more confident with nuclear technology and its actual risks
- overcome the concept of “low-probability risk”
- need to present the innovative LWR designs as safe, even if considering catastrophic external events like Fukushima, terroristic attacks or personnel negligence

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8 TECHNICAL AREA 7 – Harmonization (TA7)

Technical area leader: Edouard Scott de Martinville (IRSN)

8.1 Executive Summary

8.1.1 Scope

Harmonization in the civil nuclear domain is a cross-cutting topic aiming at settling and sharing best practices, codes and standards. It is aimed at reducing any substantial difference within a group of countries in nuclear safety requirements and objectives, fabrication, verification and operation of systems and components, safety assessment procedures and practices, as well. It involves the search for a long-term convergence which guarantees respect of agreed general objectives and principles - and the shared way to achieve them -.

That claims for an organized and structured ascending abstraction process relying upon best practices (modus operandi) jointly developed and shared by stakeholders. This process needs also suitable data, the collection and selection of which is to be achieved through pre-normative research.

The SNETP - promotes safety related research and harmonization for current and future generation of nuclear fission technologies, at European level. In this respect, an effective strategy seeking for harmonization is to be proposed within NUGENIA.

Due to its intrinsic cross-cutting nature, topics for Harmonisation are spread wherever in NUGENIA's activity. Accordingly, the NUGENIA TA7 - Harmonisation - was originally structured to supporting projects in Harmonisation, considering the following main distinct - but complementary - fields of endeavour:

- STA 7.1 Pre-normative research - PNR - for new system and component design and operating conditions, but also for suitable definition of limits and engineering criteria and establishment of practices,
- STA 7.2 Safety-oriented design and inherent assessment methodologies and practices,
- STA 7.3 Shared codes and standards,
- STA 7.4 Strategy providing with smooth and efficient methods to enlarge progressively the field of consensus among stakeholders in Europe and worldwide.

While the harmonisation is widely acknowledged to be prime importance by the whole NUGENIA community, nevertheless, the TA7 turned-out scarcely attractive to NUGENIA's people and its activity showed-up extremely poor during the first three years of the Association's life. Accordingly, after wide and open discussion within the NUGENIA's community, and mainly among the TALs, and upon suggestion by the Technical Coordinator during the 2014 Forum, it was decided to search for ways to improve the effectiveness of the process of harmonisation placing its engine closer to the actual R&D activity, so as to support the bottom-up overall move to generality.

After decision in the 27 NUGENIA's EXCOM meeting held in Otaniemi, the following improvement actions were undertaken:

- ❖ the content of STA 7.1 was moved to TA 4, watching to establishing close connection to TA6 and TA8 ,

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- the content of STA7.2 was fully integrated in TA 1,
- the overall TA7 scope was then restricted to the content of previous STA7.3 and 7.4, now renamed 7.1 and 7.2 respectively,
- The content of STA7.4 has been complemented adding topics on education and training.

Nevertheless the general scope of harmonisation in NUGENIA still holds, and the general objectives listed here below remain unchanged; only the actual implementation of activities have been moved to the above mentioned TAs.

8.1.2 Objectives

The objectives of harmonization in Europe are meant to bring improvement in three different fields of endeavour:

- Improving the safety level of the nuclear installation through shared design approaches and licensing processes,
- Supporting the deployment of nuclear energy within the European market setting-up the basis for an effective standardization of reactor components assessment,
- Benefiting in the public acceptance and cost reduction.

Accordingly, a comprehensive programme for pre-normative research should include, at least, the following main steps:

- Collecting, assimilating and structuring the already existing knowledge originating from safety-oriented programmes and demonstration activities,
- Cross-cutting this knowledge with the outcomes from the on-going research, should it be basic or pre-normative,
- Defining new programmes targeting first priority objectives, e.g. on fuel, material and system reliability and performance.

The objective for codes and standards development - while keeping coherence with the already existing efforts - should be oriented towards:

- Supporting to the development of EU standardization and regulatory frameworks for the safe operation of the nuclear installations,
- Developing handbooks and codes of best practice relying upon European Standard Organizations.

Extended operating experience sharing among stakeholders is to favour the adoption of common best practices.

The process of harmonization in the nuclear industry at European - and international - level is supported by several industrial organisations, and the European regulatory authorities have defined safety objectives that are referred to in the EU nuclear safety directive.

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Considering the goal to promote the safe and efficient operation of nuclear installations, and the participation of the main stakeholders, NUGENIA is to provide with scientific and technical basis for efficiently and effectively harmonizing criteria, methodologies and practices in the fission nuclear field and proposing guides for their implementation.

8.1.3 State of the art

Pre-normative research should focus on contributing to enrich information, practices, procedures, codes and standards; its outcomes could participate in the definition of safety criteria.

As an example, advances have been achieved in PNR in the field of structural mechanics and were recorded in the eurocodes since 1989; they are completed for concrete structures by the ETCC (in France) and other national codes. On the same structures, ISO is carrying out specific work dedicated to the fire effects and the IMPACT and IRIS projects are dedicated to the behaviour of structures under shock loading. As for metallic structures, ASME and RCC- M codes are used and, concerning the ageing an international operator consortium has launched the Material Ageing Institute.

As far as safety-PNR is concerned, large efforts were made in the US at the beginning of civil nuclear era, the development of criteria in the 70s widely relied upon. Since then, complementary studies have been carried-out on the Equivalent Cladding Reacted (ECR) and the Peak Cladding Temperature (PCT) criteria to address the growing concern on LOCA transient. Beyond these studies, it is acknowledged that the core cooling criteria are to be continuously reviewed, as well as the effect of the evolution of safety assessment methodologies.

Beyond the robust and simple methods for safety assessment - widely relying on the enforcement of suitable margins on the onset variables - that have been adopted for decades, the risk-informed methodology based on PSA studies was developed. Other methods, e.g. based on the risk space definition, are still under development.

The development of codes and standards in Europe, overviewed at the EU level by the European Standardization System, are being pursued along specialised technical domains in the CENELEC for the electro-technical domain, and by CEN for the other domains relevant to nuclear industry. On the safety side, the WENRA reference levels for existing plants were adopted in the 2009/71/Euratom safety directive, and WENRA proposed safety directives for the new power plants in 2010. At the worldwide level, IAEA, IEC and ISO have produced, with the participation of all stakeholders, including Europeans.

The harmonization in the nuclear industry at European and international level and the standardization of the safety assessment of nuclear power plants are supported by several industrial organisations as well as by the regulatory authorities. Pushing forward these efforts doesn't need so much additional legal obligations but mainly a large cooperation among all stakeholders in order to produce common proposals to practically reach reliable assessments.

8.1.4 Challenges

The main challenges for the NUGENIA in the field of Harmonisation are:

8.1.4.1 In the pre-normative research

NUGENIA is a privileged field to implement pre-normative research activities, because the Association's partners enjoy a wide portfolio of ongoing and future projects which propose and

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develop new technologies, new materials, new approaches and innovative technological solutions. To achieve their R&D objectives they must develop and adopt their own procedures and internal rules which are translated into practical guidelines within their field of endeavour. These practical guidelines - matter-of-fact based practices - have to be firstly assessed, accepted and validated within the project to allow it be conducted efficiently and effectively. They could be later on proposed for enlarged end generalized application through a process for generalisation and standardisation.

This bottom up process of generalisation and standardisation should be the main engine for harmonisation through the pre-normative research within NUGENIA.

Several topics suitable for harmonisation have already been clearly identified. They mainly concern system design and safety practices, design, fabrication, test and operation of system and components, methodologies and practices for testing and validation. The standardization should start with sharing the state of knowledge among NUGENIA areas.

The safety margins evaluation approach is the subject of a common challenge. The actual activity in the field will be carried-out in TA1 (STA1.1). Systems materials and components, including ageing, harmonisation issues are addressed mainly in TA4 (STA4.3) and TA3 (STA3.2 - I&C -, and 3.4 - water chemistry -). Connections are established with TA6 for consequences on new design and TA8 for testing and validation.

Moreover, NUGENIA's partners are directly or indirectly owners of a large amount of experimental results which can be effectively and proficiently analyzed for pre-normative purposes to seek and propose new and more effective limitation and thresholds to onset parameters in the design. The topic is addressed inTA1 (STA1.3).

8.1.4.2 In the plant performance assessment

In the design of new plant or systems, the adoption of advanced methodologies, combining defence-in-depth, risk-informed and safety margin approaches underline the need for new research efforts.

The objective for codes and standards development - while keeping coherence with the already existing efforts - should be oriented towards:

- Support to the development of EU standardization and regulatory frameworks for the safe operation of the nuclear installations,
- Development of handbooks and codes of best practice relying upon European Standard Organizations.

Extended sharing of operating experience among stakeholders has to favour the adoption of common best practices

8.1.4.3 In the codes and standards development

Establishing liaisons with IAEA, CEN, and ISO TC 85,

Gathering and synthesizing the already existing knowledge to single-out the methodologies and practices that need to be harmonised

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- Exploiting any NUGENIA launched project to reference the standardization relevant results, preparing formatted data for standard activities and promoting the standardization process in the projects
- Developing NUGENIA handbooks and codes of best practices.

8.1.4.4 In the harmonization strategy

The harmonization strategy is not aimed at issuing new safety directives. It should rely upon a systematic and continuous dialogue among the stakeholders.
NUGENIA enjoys gathering those stakeholders that adopt the regulations and rules for design, operation, safety-survey and safety-assessment. Thus it appears as a favoured place to create harmonization through scientifically found consensus.
The TA7 objectives are to be strongly considered in all NUGENIA areas to ensure a systematic and comprehensive sharing of knowledge and R&D results.

8.2 Sub Technical Areas (STA)

8.2.1 Forewords

8.2.1.1 Pre-normative research

NUGENIA is a privileged field to implement pre-normative research activities, because the Association's partners enjoy a wide portfolio of ongoing and future projects which propose and develop new technologies, new materials, new approaches and innovative technological solutions.

To achieve their R&D objectives they must develop and adopt their own procedures and internal rules which are translated into practical guidelines within their field of endeavour. These practical guidelines - matter-of-fact based practices - have to be firstly assessed, accepted and validated within the project to allow it be conducted efficiently and effectively. They could be later on proposed for enlarged end generalized application through a process for generalisation and standardisation.

This bottom up process of generalisation and standardisation should be the main engine for harmonisation through the pre-normative research within NUGENIA.

The activity on Harmonisation through pre-normative research is to be conducted close to the actual ongoing projects it has to rely upon; thus the description of task is provided in TA3, TA4 , with tight connections to TA6 and TA8. Please refer to them for any useful information.

8.2.1.2 Towards an improved and generalized safety culture: Design and assessment methodologies and practices

It has been widely acknowledged from the very beginnings of the definition of the NUGENIA R&D Roadmaps that the implementation and improvement of the Safety Culture through harmonisation which was located in TA7 had to be conducted in tight connection with TA1. According to the decisions taken in the NUGENIA Forum and upon agreement of the TALs[17] , the whole R&D activity in

[17] As said in the Introduction, the proposal of the Forum concerning TA7 has been agreed by the TALs during the third NUGENIA TALS meetingheld in Paris, June 2014. Discussions are going on the way to put the decision in practice Two ways are allowed : either spreading the whole R&D content of the TA7 all aver NUGENIA or

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the field has been moved to TA1. The description of the objectives and the approach is provided in TA1. Please refer to it for any useful information.

8.2.2 Codes and Standards (STA 7.1)

8.2.2.1 Scope

[18]Codes and standards can be considered as the final product resulting from the Pre-normative research subtask.

Working within the European framework it is worth mainly focusing on European standards even if some organizations or standards originating from outside Europe are to be mentioned for information sake.

In this paragraph the definitions adopted for the terms “standard” and “European standard” are to be found in CEN/CENELEC IR part 3. These definitions originating from ISO/IEC were used to prepare definitions in official European texts (e.g. 93/38/EEC, 83/189/EEC or 2004/17/EC)

Concerning the term “Code” it generally comes with the term “Rules”. Obligations linked to administrative or political decisions to be fulfilled are generally associated to the “Rules”. Usually the “Rules” do not indicate how the obligations can be fulfilled. The role of the “Codes” produced generally by the industry sector is to explain how a particular technical solution abiding by these “Codes” is fulfilling the “Rules”. The term “Codes” has no formal definition and so we will focus in this document mainly on “Standards” and “European standards”, even if “Codes” can be considered sometime as a pre-normative entry data for the development of a “Standard” or another pre normative document produced by standard organizations (Technical Report, Publically Available Specification, Workshop Agreement …),, e.g. RCC-MRX considered today in a frame of a CEN workshop, that activity being supported in the frame of the SNETP.

8.2.2.2 Objectives

One final product resulting from the pre-normative research subtask is entry data for the standard organization experts to be incorporated in the development of standards or in development of one of the pre-normative documents produced by standard organizations.

As a consequence our two objectives should be:

- Developing EU standardization and regulatory framework for the safe operation of the nuclear installations,
- And when the development of a European standard is not possible, developing handbooks and codes of best practice using the different type of pre-normative documents which can be developed in European Standard Organizations,

These two objectives call some comments:

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limiting to transfer to TA1 the R&D activity on the improvement of the safety culture. This second option has been retainded in the present document. Further evolution and changes are likely. They will be accounted for in an up-dated release when fully aknowledged and implemented.

[18]The whole scope of the STA7.1, Codes and Standards could be in the future completely removed from the NUGENIA Roadmaps and the inherents objectives assigned as a specific mission to to the NUGENIA’s Reperesentaive in the European Group for Standardization in the field.

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- ❖ Nuclear safety is not only an European matter, it is an international one; the consequences of major accidents are not regionally limited,
- ❖ So, harmonization and standardization in an European frame, not only need to be established taking into consideration of improvements made at international level, but also Europe should have a major role to foster excellence in safety objectives and standards established in international frames (IAEA, IEC, ISO),
- ❖ The development of European safety objectives and standards should be carried out with the attention for a "legibility" at international level, in terms of compatibility, differences… comparing to other standards as, for instance, those of the American nuclear industry and regulations. This point seems crucial for the success of a European nuclear industry.

8.2.2.3 State of the art

When generally speaking of standards in Europe we have to note that European standards inherited a legal nature from the European legal texts on which they are grafted. So it comes as no surprise that describing the context relevant for the standardization issue we cannot avoid speaking of the legal European context.

8.2.2.4 Europe and standardization

Concerning European orientations relevant for standardization, the main recent document covering that issue is the EXPRESS report (EXpert Panel for the Review of the European Standardization System) titled « Standardization for a competitive an innovative Europe: a vision for 2020 » published in 2010 and available on the EC website "ec.europa.eu". This consensually based and strongly supported report, prepared by a 30 standardization expert panel, presents a review of the ESS (European Standardization System) and a set of 10 recommendations. It concludes that building on the current arrangements the ESS can meet the considerable challenge that lies ahead to 2020 and beyond.

Out of the 10 recommendations formulated, at least the following ones can have a specific resonance in the nuclear sector:

- ❖ *"Engagement with policymakers"*
- ❖ *"Meeting the needs of society"*
- ❖ *"Engaging SMEs, engaging all stakeholders"*
- ❖ *"Building relationships with researchers and innovators"*

As a reminder, the ESS (European Standardization System) comprises 3 ESOs (European Standardization Organizations): CEN ("Comité Européen de Normalisation" - the European Committee for Standardisation), CENELEC ("Comité Européen de Normalisation du domaine ELECtrotechnique" - The European committee for the standardisation of the electrotechnical domain) and ETSI (European Telecommunication Standard Institute). This system is one of the main mechanisms which allowed during the two last decades the development of the European unique market, in particular thanks to the use of the standards to suppress non-tariff barriers representing obstacle to trade; this system which had for political frame the "New approach" is now integrated in the New Legislative Framework.

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8.2.2.5 European standardization and the nuclear sector

The first text to consider is the 1957 Rome Treaty which made no reference to safety of NPPs but which covered workers radiation protection. As a consequence nuclear safety of NPPs was situated outside the European competences, on the contrary of radiation protection for workers.

From the early 60's till end of the 90's, the European Standardization System was de facto aligned on that legal situation and no European standards were published.

In 2000, the situation changed, some projects were launched at the Commission level, WENRA (Western European Nuclear Regulator Association) started its activities. In IEC/SC45A (International Electrotechnical Commission, Sub-Committee 45A (Instrumentation and control of nuclear facilities), a debate was launched between the representatives of the European National Committees to envisage the possibility to produce European standards related to safety of NPPs. The debate concluded that development of European standards related to safety of NPPs was premature, but it opened the door for the development of standards related to radioprotection for workers using published IEC/SC45B (International Electrotechnical Commission, Sub-Committee 45B (Radioprotection instrumentation) standards.

As a consequence, from 2003-2007, a task force of the CENELEC prepared the endorsement of IEC/SC45B standards on a case by case basis, developing for each standard to be endorsed CMs (Common Modifications) and 13 EN standards based on IEC/SC45B standards were published.

In 2007, WENRA published its Safety Reference Levels, and the publication of that document launched again in IEC/SC45A, the debate between the representatives of the European National Committees on the possibility of development of European standards related to safety of NPPs. A survey was ordered by the CENELEC on the use and penetration of IEC/SC45A standards in Europe and on the basis of that report CENELEC BT decided in 2007 to activate CLC/TC45AX (CENELEC Technical Committee 45AX (Instrumentation and control of nuclear facilities)) and CLC/TC45B (CENELEC Technical Committee 45B (Radioprotection instrumentation)) to produce EN standards on their respective domains.

In 2009, in the Council Directives 2009/71/EURATOM, the IAEA and WENRA documents were recognized.

In 2009/2010, WENRA issued documents concerning "Safety Objectives for New Power Plants".

In 2012, CLC/TC45AX already published 11 EN standards which correspond to non modified IEC/SC45A standards; those standards being consistent and coherent with the IAEA safety principles and terminology. In 2012 CLC/TC45B published 17 standards based on IEC/SC45B standards integrating common modifications; this last figure includes the 13 standards published prior to CLC/TC45B activation.

November 2011, CEN/CENELEC decided to launch a focus group to produce a report on the issue of the European standardization for the nuclear sector. Spring 2012 a questionnaire will be circulated to the main stakeholders of the sector to evaluate the situation and prepare a report to be presented to CEN/CENELEC BT on the situation and decide of orientations to follow concerning the European standardization for the nuclear sector.

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8.2.2.6 International organizations and standards

At international level the organizations we are considering are:

- The IAEA (International Atomic Energy Agency) which is part of the United Nation family,
- The international standard organizations mirrored by the ESS, namely the ISO for the CEN, the IEC for the CENELEC and the UTI for the ETSI. For the nuclear sector only ISO and IEC are relevant. One point to note is the fact that those organizations are recognized in the annex III of the (TBT) Trade Barrier Treaty of the (WTO) World Trade Organization.

8.2.2.6.1 IAEA (International Atomic Energy Agency)

IAEA covers globally the nuclear sector for peaceful purposes. In particular, IAEA develops nuclear safety standards to promote the achievement and maintenance of high levels of safety in applications of nuclear energy, as well as the protection of human health and the environment against ionizing radiation. The IAEA documents are publically available and free; more information on the IAEA document series is to be found on the IAEA web site:

- Safety Fundamentals;
- General safety requirements;
- Safety Guides.

8.2.2.6.2 IEC (International Electrotechnical Commission)

In IEC, the technical committee related to the nuclear sector is TC45 (Nuclear Instrumentation) with its two subcommittees SC45A (Instrumentation and control of nuclear facilities) and SC45B (Radioprotection instrumentation).

- IEC/TC45 prepares international standards relating to electrical and electronic equipment and systems for instrumentation specific to nuclear applications.
- IEC/SC45A prepares standards applicable to the electronic and electrical functions and associated systems and equipment used in the instrumentation and control systems (I&C) of nuclear energy generation facilities (NPPss, fuel handling and processing facilities, interim and final repositories for spent fuel and nuclear waste) to improve the efficiency and safety of nuclear energy generation. Those standards cover the entire lifecycle of these I&C systems, from conception, through design, manufacture, test, installation, commissioning, operation, maintenance, aging management, modernization and decommissioning. In this context, one of the IEC/SC45A strategic tasks is to review and comment on drafts of IAEA safety code in order to maintain consistency between IAEA and IEC documents and identify detailed technical aspects for which IEC standard developments are appropriate and responsive to the market needs.
- IEC/SC45B prepares standards covering all the fields of radiation protection instrumentation. That is, for the measurement, under normal and accident conditions, of external and internal individual exposure, the workplace, the environment (including foodstuffs).

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- IEC/TC45, IEC/SC45A and IEC/SC45B portfolios comprise respectively 37, 68 and 51 published standards. More information on this committee; those sub committees and the documents they published are to found on the IEC website.

8.2.2.6.3 ISO (International Standard Organisation)

In ISO, the technical committee related to the nuclear sector is ISO/TC85 (Nuclear energy, nuclear technologies, and radiological protection).

- ISO/TC85 prepares standards in the field of peaceful applications of nuclear energy and of the protection of individuals against all sources of ionising radiations. ISO/TC85 comprises 3 sub committees: SC2 (radiological protection), SC5 (nuclear fuel cycle) and SC6 (reactor technology).
- ISO/TC85/SC2 (radiation protection) prepares standards answering the needs of medical activities, with a focus on radiation protection of patients.
- ISO/TC85/SC5 (nuclear fuel cycle) prepares standards in the field of waste management and decommissioning activities (e.g.: characterization (waste, fuel, MOX pellets…), standardization related to criticality safety, transport).
- ISO/TC85/SC6 (reactor technology) prepares standards about analysis and measurements dedicated to the safe and efficient operation of nuclear power reactor, standards concerning the safe and efficient operation of research reactors (RRs) and the irradiation services from RRs, standards including reliability data for NPPs and RRs, and data related to NPPs life extension and decommissioning planning.
- ISO/TC85, ISO/TC85/SC2, ISO/TC85/SC5, ISO/TC85/SC6 portfolios comprise respectively 27, 66, 65, 6 published standards. More information on this committee, those sub committees and the documents they published are to found on the IEC website.

8.2.2.7 Nuclear sector coverage by ESOs and nuclear EN standards

As indicated for the time being the only European standards for the nuclear sector are 11 ones published by CLC/TC45AX and 17 ones published by CLC/TC45B.

CEN/CENELEC launched recently the Focus Group on Nuclear Standardization. This Group intends to prepare an overview for the nuclear energy stakeholder community on suitable standards already publicly available (from the International Standards Organizations ISO and IEC, from CEN/CENELEC or other sources) or in preparation, to meet specific needs for products and services in the nuclear sector. Where no suitable standards exist, the Group will define best ways to provide them in future, in preference internationally but if necessary in Europe, and make recommendations accordingly. The activities of this Focus Group on Nuclear Energy Standards help improve synergies between innovation/research projects and civil standardization for nuclear energy, noting the various proposals in the Commission Action Plan for European Standardization."

It is suggested to wait for the results of the circulation of the CEN/CENELEC focus group questionnaire which will be circulated to the main stakeholders of the nuclear sector to evaluate the current situation and the subsequent report to be prepared on the issue of the European standardization for the nuclear sector to have a better idea of the particular technical topics for

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which standards are needed and then be able to evaluate the need of pre-normative research to develop those standards or pre-normative documents.

8.2.2.8 Other "standard" organizations and documents

We cannot avoid speaking of national standard organizations acting at international level; this is in particular the case for US organizations as ASME, IEEE or ISA... Those organizations which are very powerful are listed in the list of the 280 Standard accredited developers of the ANSI (American National Standard Institute). The most important point which is to be taken into account is that concerning the nuclear sector their basic frame reference is the US regulatory framework which is not the reference in Europe. Moreover concerning the issue of penetration of the US standard organizations see the EXPRESS report. Moreover, we can also list many organizations (e.g. EPRI in the US (Life prediction of NPPs structural components guidelines, NPPs water chemistry guidelines, practices for maintenance and decommissioning), JRC in Europe) preparing technical reports which can reflect different level of consensus and could be used as basis for developing pre-normative documents or even standards in some particular cases.

The French AFCEN is worth noting with codes RCC-M (and presently its development with the RCC-MRX), largely used for the French PWRs of generation II and III (EPR).

Harmonization topics

A great number of the existing standards and guidelines are not always applied. There is first an effort to be made in promotion, information dissemination and acceptance. In the framework of NUGENIA, that could be achieved through:

- ✓ **The systematic reference in the research projects to domain relevant existing European or International (IAEA, IEC or ISO) standards from the early steps of the project,**
- ✓ **The introduction in the final report of the research projects of a paragraph on production and identification of pre-normative entry data formatted to be easily integrated in the standard activities,**
- ✓ **The participation in the research projects of experts having an experience of standard development in particular to prepare the formatted pre-normative entry data,**
- ✓ **The organisation of training periods and workshops for the research people to know the world of standardization,**
- ✓ **The review of the standardization organization needs for pre-normative data.**

To be proactive, European partners should rely upon the already operating CEN-CENELEC, and particularly on its Focus Group on Nuclear Standardization. FORATOM-ENISS should also be associated.

As for the ENSREG/WENRA/ETSON, it should be useful to have proposals ("route sheet") of CEN-CENELEC for harmonization of codes and standards (I&C, mechanics). Europe should foster such an initiative, and a representative of NUGENIA Topic 7 group should be useful at appropriate level of

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CEN-CENELEC instances. Recommendations from the ENSREG/WENRA/ETSON (see above a)) could be taken into account as "input data", among others.

Collection of data – identification of areas to be addressed should be organized:

- ✓ **Establishing liaisons with IAEA, CEN, and ISO TC 85**
- ✓ **Gathering and synthesizing the existing knowledge to single-out the methodologies and practices that need to be harmonised.**

8.2.3 Harmonization strategy (STA 7.2)

8.2.3.1 Scope (including specific objectives)

The process of harmonization in the design of nuclear power plants at European - and international - level and the standardization of nuclear power plants are supported by several industrial organisations. The regulatory authorities have cooperated together in order to established common safety objectives that were published in 2010. The harmonization process has now to fit the design of the NPPs with these new objectives while defining standardised reactor designs.

As part of NUGENIA's goal to become a community to promote the safe and efficient operation of nuclear installations, particularly the NPPs, NUGENIA shall provide, in a transparent and visible way, a scientific and technical basis by initiating and supporting European, and international as appropriate, R&D projects and programmes on harmonization of design and of safety standards and facilitate implementation and dissemination of R&D results.

In this respect, an effective strategy towards harmonization must cover the whole range of activities: pre-normative research, plant design and operation methodologies and practices, codes and standards. Similarly, a strong cross-cutting link with the other scope technical areas of NUGENIA is also necessary to ensure a systematic approach in the direction of harmonization.

8.2.3.1.1 European harmonization of safety objectives and safety assessment practices

To guarantee a common level of safety of current reactors across Europe and to promote the construction of new reactors by bringing predictability to the licensing process, efforts are needed from the European MS in order to harmonize the safety objectives and methodologies related to licensing process and safety assessment requirements (design evaluation and safety assessment, subcontractor certification, introduction of new fabrication and assembly technologies, transport of irradiated fuel / materials…). A first major milestone has been achieved by the WENRA association with the issuing of "Reference levels" for generation II NPPs and safety objectives for new projects. Harmonized safety approaches and criteria will facilitate design licensing of standardised reactors and ensure more predictable reactor construction and operation timescales, while improving experience feedback.

Presently, harmonization of safety practices are under development in the different working groups of the European Technical Safety Organizations Network (ETSON). This harmonization consists in elaborating common safety assessment guidelines based on the common experience of design and operation assessment. This bottom up approach does not require further development of the European nuclear safety directives, but makes efforts to translate into practical approaches the design and operation according to safety objectives

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Under current Euratom and national legislation, the prime responsibility for nuclear safety relies on the licensee (the operator of the nuclear facility or installation). National regulators require license holders to make technical improvements to their facilities as a follow up to safety assessments. National regulators have to ensure that the required measures are implemented correctly. The license holders must also demonstrate the regulators that, once they have been defined, the new regulations are implemented. A necessary way towards harmonization consists in all the operators to harmonize their implementation, considering all nuclear installations across Europe. Nevertheless, at present, even if some common technical safety standards exist, there is no unified guidance for nuclear power plants design in the EU.

The preliminary results of the stress tests show that there is no consistency in the handling of safety margins across nuclear power plants in Europe. Depending on the final results of the EU stress tests process, as well as on lessons learnt from the Fukushima nuclear accident, an EU-wide set of basic principles and requirements could be envisaged, together with associated minimum technical criteria in the areas of siting, design & construction and operation of nuclear power plants; these criteria would ultimately be implemented by plant operators. Nevertheless, this work shall be difficult: for instance, it can be mentioned that the WENRA association concluded to the difficulty to define shared safety objectives in terms of radiological doses.

An EU wide set of criteria for the definition of site characteristics, licensing requirements and operational checks would require plant operators to converge towards best practices for new nuclear power plants that are to be built in the EU. Such requirements already exist in international and EU practice. For example, international peer reviews, at present limited to the national legal and regulatory framework, could be broadened to include design safety and operational safety of NPPs.

A range of actors should be involved in finalising the set of recommendations for the new European nuclear safety architecture, including the national regulators with ENSREG and WENRA, the Technical Safety Organizations (ETSON), together with the nuclear industry representatives.

8.2.3.1.2 European harmonization of codes and standards

Standards are at the basis of many technical solutions developed for the nuclear power plants. Due to the development at national level of codes and standards previous to or in parallel to development of equivalent standards at international or regional level in the relevant bodies (IAEA, IEC, ISO, CEN, CENELEC…), these codes and national standards are only applicable nationally or in other countries that accept them. This status impedes the opening to international markets of services and products that could be more widely available, therefore, improving competition in a high-technological market. Aware of this status, nuclear community is working together to propose codes and standards in various domains that are internationally accepted, such as mechanical codes or instrumentation and control (I&C) standards. Several joint collaboration projects have been put in place among the organizations producing standards. Some harmonization studies have been launched, for example, to compare mechanical codes of the ASME and AFCEN's RCC-M. At the same time, agreements have been reached between organizations to develop common standards for the nuclear industry (e.g. IEEE and IEC agreements on electrical standards for nuclear applications). Nuclear community has already launched several activities aiming at setting common documents to define utility requirements for new reactor designs like the Utility Requirements Document (URD) in the US, produced by the Electric Power Research Institute (EPRI), or the European Utility Requirements (EUR) in Europe. The convergence of codes and standards would be a further motive

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for regulators to accept regional or international codes and standards if they are in compliance with national safety regulations and represent good modern practices.

A first help to foster the development of harmonized codes and standards can come from the survey of application of best practices in the normative research. Either the pre-normative research (PNR), to support future trends and standards or the co-normative research (CNR) in direct interaction with ongoing and/or planned standardization activities should be conducted according to best practices that allow a more integrated and harmonized approach.

In the past, EC RTD Framework Programmes (FP) have had specific routes for funding normative research. Normative research (remind!!! Not technical research) still remains at the centre of free market policy as shown by the Europe 2020 Innovation Union Flagship Initiative (2010) that indicates how “standards play an important role for innovation by codifying information on the state of the art of a particular technology thereby enabling dissemination of knowledge, interoperability between new products and services and provide a platform for further innovation. The rapid shortening of innovation cycles and the convergence of technologies across the boundaries of the three European standardisation organisations are a particular challenge.”

8.2.3.2 State of the art

The legal framework for the harmonization of nuclear safety regulatory requirements, criteria and review practices is relatively recent.

For historical and political reasons, nuclear energy and nuclear safety are regulated somewhat differently from other sectors under EU law. The Treaty establishing the European Atomic Energy Community (the “Euratom Treaty”) was adopted in 1957 with the purpose to create among the Member States conditions for the establishment and growth of the nuclear industry but the safety of nuclear installations is not a responsibility granted to the Community by the Euratom Treaty.

The Euratom Treaty was considered as giving Community institutions competence to adopt directives and recommendations in the field of radiation protection and waste management only. Given the more-or-less inevitably cross-border nature of any major nuclear accident, and given the aim of the Euratom Treaty of creating a Common Market for nuclear energy, the omission of any harmonization provisions for nuclear reactor safety does seem quite remarkable. In 2002, the European Court of Justice (ECJ) clarified in Case 29/99 that the Community shares competences with Member States in respect of nuclear safety as well as radiation protection.

The adoption of common EU nuclear safety standards has been under discussion for a number of years. In 2007, the High Level Group on Nuclear Safety and Waste Management – the European Nuclear Safety Regulators Group ENSREG considered whether there was a need for a legally binding instrument on nuclear safety at the EU level.

In November 2008, the Commission tabled a revised directive on nuclear safety. Subsequent revision of the proposal paved the way for the adoption of the Directive on 25 June 2009.

The Directive incorporates the provisions of the Convention as well as some of the IAEA’s safety standards and principles set out in the Fundamental Safety Principles. It is important to note that the Directive does not prevent Member States from adopting safety measures that are more stringent than those covered by the Directive, provided this is done in compliance with Community law. Rather than prescribing a legislative and regulatory framework for nuclear safety, the Directive ensures a

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minimum level of nuclear safety within the EU. The Directive sets the basis for harmonized methodologies concerning nuclear safety including the design, the operational and the decommissioning of NPPs and fuel cycle activates.

Two approaches can be followed to improve the nuclear safety framework: the use of legislative amendments, to reinforce the existing Community nuclear safety legislative framework and improvements in the implementation of existing mechanisms, as well as enhanced coordination between the Member States:

- ❖ In the first approach, the nuclear safety framework will be reviewed both at the Community and at the national level. Within international institutions, the Commission and the Member States will have to act together to ensure that developments of the international nuclear safety framework are consistent with Community and national legislation. The EURATOM legislative revision process will also need to reflect the international developments to ensure coherence. A possible harmonization of design licensing of nuclear power plants is closely linked to this process, as it also aims to improve nuclear safety
- ❖ The role of NUGENIA can be crucial in defining the priorities that allow a rapid and consistent development towards harmonization; this would represent an important part in the second approach, the one which does not implies directives and regulations.

Two important initiatives aiming at design standardisation and harmonization of national regulatory regimes have been launched in recent years:

1) Representing the industry, the World Nuclear Association (WNA) has set up in January 2007 the expert working group CORDEL (Cooperation in Reactor Design Evaluation and Licensing). The working group seeks to identify and demonstrate the real benefits that could be realized from internationally accepted standard licensing requirements for reactor designs and in 2010 came up with a roadmap on “International Standardisation of Nuclear Reactor Design”. The roadmap proposes a three-phase stepwise approach that culminates in an international design certification and provides a useful conceptual framework for the practicalities of bringing about standardization and harmonization. It should also be recognized that there is already some movement in this direction. Efforts are in hand to identify differences and develop aligned international codes and standards in various domains such as mechanical codes and instrumentation and control (I&C) through such organizations as ASME and AFCEN, and IEEE and IEC.

Overarching utility requirements for new reactor designs have been developed by EPRI-URD in the US and EUR in Europe. The European Utilities Requirements (EUR) is a product of major European electricity producers and associations and focuses on common requirements for future LWRs to be built in Europe. It is intended to be a tool for promoting harmonization, in particular of main safety objectives and safety requirements.

2) The European Standardisation Bodies and the European Commission co-operate since a number of years towards stronger links between design standardisation and research. Input of publicly funded research to European standard activities is primarily based upon the RTD

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framework of the European Union. The methodology to address the research-standards interface has been developed by the CEN/CENELEC BT STAIR (STAndards, Innovation and Research) Working Group. This methodology is known as the "Integrated Approach"; the vision is that standardisation should not be an afterthought of research and development and innovation (R&D&I) projects but instead be considered right from the start. [The term standardisation is used here for the full range of CEN or CENELEC deliverables, including European Standards (Ens), Technical Specifications (TSs), Technical Reports (TRs) and CEN or CENELEC Workshop Agreements (CWAs)]. In order to realize the "Integrated Approach" CEN and CENELEC members would participate in projects consortia to advice on available standards and, where considered timely, to ensure a standardisation activity is progressed under the dissemination actions of the project. In CEN, the collaboration between the project and the standardisation world is facilitated through Project Liaisons which allow research projects to participate in CEN Technical Committees and relevant Working Group meetings as observer. CENELEC has a similar (albeit slightly different) approach: the Technical Liaison Partnership.

CEN-CENELEC launched recently a short term Focus Group on Nuclear Standardization. This Group intends to prepare by the end of 2012 an overview for the nuclear energy stakeholder community on suitable standards already publicly available (from the International Standards Organizations ISO and IEC, from CEN-CENELEC or other sources) or in preparation, to meet specific needs for products and services in the nuclear sector.

The European network of excellence NULIFE (Nuclear plant life prediction) was launched under the sixth Euratom Framework Programme with a clear focus on integrating research on materials, structures and systems and exploiting the results of this integration through the production of harmonized lifetime assessment methods. NULIFE helps provide a better common understanding of the factors affecting the lifetime of nuclear power plants which, together with associated management methods, help to facilitate safe and economic long term operation of existing nuclear power plants.

On the regulatory side and their technical safety organisations (TSO), national authorities from ten OECD countries participate in the Multinational Design Evaluation Programme (MDEP) with the purpose of co-operating on safety design reviews of new reactors and identifying opportunities to harmonize and converge on safety licensing review practices and requirements. As individual regulators review new nuclear reactor designs, MDEP aims to enhance cooperation among them through sharing resources and knowledge, thus improving efficiency and effectiveness of the licensing process. The MDEP was established in 2008 by the nuclear regulators of France, Finland and the USA in order to assist them in the exchange of technical data during certification of the European Pressurized Reactor (EPR). MDEP currently supports and promotes the convergence of codes, standards and safety goals in participating countries, namely Canada, China, Finland, France, Japan, South Africa, South Korea, Russia, the UK and the USA. The IAEA also participates in MDEP meetings. At a later stage it is expected that the lessons learned will be used to facilitate licensing of new reactors, including Generation IV designs.

Regional initiatives have also been taken by regulators and utilities (such as the Western European Regulators' Association (WENRA) and the European Nuclear Installations Safety Standards (ENISS) initiative in Europe). WENRA is a network of Chief Regulators of EU countries with nuclear power

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plants and Switzerland as well as of other interested European countries with observer status. As mentioned above (§ 7.3.2.2), in 2007, WENRA has developed Reactor Safety Reference Levels as an instrument to develop a common approach on the harmonization of nuclear safety and its regulation in the EU countries. In 2010, WENRA adopted safety objectives for new nuclear power plants on the basis of the IAEA Fundamental Safety Principles.

The main mission of ENISS, created in 2005 within FORATOM, is to bring together decision-makers, operators and specialists from the nuclear industry with regulators at European level in order to identify and possibly agree upon the scope and substance of harmonized safety standards recognising that harmonizing regulations is the best way of ensuring that the industry can evolve within a stable legal framework.

The IAEA's Integrated Regulatory Review Service (IRRS) provides reviews of national regulatory systems to identify and spread best practices in licensing and oversight. The IAEA Safety Standards specify safety requirements and guides representing best/good practices, which are increasingly used as reference for review of national safety standards and as a benchmark for harmonization in all countries utilizing nuclear energy for peaceful purposes.

Finally, at European level, the European Technical Safety Organizations Network (ETSON) has engaged working groups related to safety approaches/assessment and technical topics (physics and thermal-hydraulics, severe accidents, human and organisational factors…) in order to share and harmonize practices.

8.2.3.3 Challenges

Specific areas that need concerted international co-operative efforts include:

- Probabilistic safety assessment (PSA) methods and the role of PSA in safety decision making, including the appropriate balance among PSA, deterministic methods, and engineering judgement;
- Safety assessment procedures within the licensing process from country to country, including technical documentation requirements; consideration also should be given to harmonization steps that ease the complications inherent in licensing a plant designed to the codes and standards of a different country

8.2.3.4 Harmonization topics

To implement harmonization within European countries is worth:

- Identifying key interlocutors working on harmonization (IAEA, CEN-CENELEC, WENRA, ETSON, OECD-NEA…) and gathering them regularly (One meeting/2 yrs)
- Dispatching one NUGENIA representative in the CEN-CENELEC Focus Group on Nuclear Standardization
- Editing of an overview report on the work done by international organizations on design and safety harmonization

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8.2.3.4.1 Safety assessment and regulatory process

As said, the engine for harmonisation process is to be found in TA 1 to TA 3 and TA 4, with connection to TA6 and TA8.

The main actor in harmonization of safety assessment and regulatory process is identified to be TA1.

Nevertheless, on these concerns, Europe must primary rely on:

- ENSREG and WENRA associations,
- ETSON.
- FORATOM-ENISS should be associated and JRC can bring a support, to be defined.
- WANO can be associated or consulted, as appropriate.

Europe should incite ENSREG/WENRA/ETSON to carry on and foster their activities on harmonization; "route sheets" could be useful.

They could be invited to do "state of the art" reports of the present practices in their members' countries, emphasizing discrepancies, best practices and needs for convergences.

They should also be invited to engage a reflection and give their propositions concerning:

- Harmonized licensing process (at design stage, then during the lifetime of the facilities)
- European design certification
- International design certification.

Obviously, these reflections could be made in association with the IAEA.

It should also be emphasized that, in the framework of the "state of the art" reports, ENSREG/WENRA/ETSON carry out a review of the codes and standards proposed and used by the designers and operators for nuclear facilities in the different members countries, and assessments, positions and possible recommendations made by the regulators on the uses of these codes and standards (and possibly on these codes and standards themselves). The results of this review could be an important "input data" in developing harmonized codes and standards.

In addition, at European level, harmonization and sharing of national supports in "emergency preparedness" should have priority. ENSREG/WENRA/ETSON, with IAEA, should be invited to take opportunity of the post-Fukushima learning to engage reflections, knowing that the sharing of supports must not impede the national organisations. This capacity in supporting implies the crisis management to be sufficiently harmonized among the different countries in terms of organisation, technical communication, management methodology and scientific calculation tools.

Another important topic is ageing and LTO (Long Term Operation – extension of operation). There is a safety guide of the IAEA on "Ageing Management", but ageing and LTO seem to deserve a harmonization of strategy and assessment practices,

Other topics really need harmonization, such as:

- **combination of hazards, between themselves and with internal "initiating events",**

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- **events not considered for the design and safety demonstration ("exclusion", "practical elimination"…),**
- **the use of "leak before break" approaches for the safety demonstration,**

TSO members of NUGENIA Topic groups could ensure a link with ENSREG/WENRA/ETSON reflections

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9 TECHNICAL AREA 8 – In Service Inspection, Inspection Qualification and NDE Evaluation (TA8)

Technical Area Leader: Etienne Martin (EDF)

9.1 Executive summary

The European Network for Inspection and Qualification (ENIQ) is a network driven by the nuclear utilities of the European Union member countries and Switzerland and has been existing since 1992.

ENIQ is active on the field of in-service inspection (ISI) of nuclear power plants (NPP) by non-destructive testing (NDT), and is working mainly in the areas of qualification of NDT systems and risk-informed in-service inspection (RI-ISI). The technical work is performed by three task groups (TG): TG Qualification, TG Risk TG Independent Qualification Body. Both task groups are made up of experts from Europe with a few experts from overseas countries (USA and Canada).

The main areas on qualification for future work are:

- Ensure a robust link between inspection and inspection qualification and the input processes such as code requirements, safety cases.
- Provide guidance on inspection qualification issues to address concerns and queries raised by members.
- Ensure that inspection results are used effectively to inform the safety assessments made by others.
- Condition monitoring by NDT in order to improve maintenance of plants.
- Support extended use of computer modelling and simulations.
- Extended application of NDT qualification to other NDT methods.

The main areas on risk for future work are:

- Improve the link between RI-ISI and inspection qualification (e.g. explains how risk reduction criteria could set ISI requirements, quantify the output of a qualified inspection, etc.).
- The role of RI-ISI in defence-in depth (pressure vessels, qualification approach, risk, etc.).
- RI-ISI for new build (including pre-service inspections).
- RI-ISI and qualification guidance for non-pressure boundary items.
- RI-ISI guidance to optimise ISI intervals.
- Develop more detailed guidance for the use of expert knowledge in RI-ISI and qualification.
- Guidance on the use of alternative methods (to ISI) for managing risk.
- Probabilistic safety assessment (PSA) (technical adequacy for RI-ISI, refinement for RI-ISI programmes).
- Guidance on expert judgement.

Additionally, it has been proposed to create a third TG as a forum for international qualification bodies to exchange experience and develop good practices and guidelines.

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In 2010, the ENIQ Steering Committee recognised that the European nuclear industry was entering a period of significant change and thus initiated an internal discussion to determine its vision and objectives regarding ENIQ's future role and activities. This exercise resulted in the issuing of a strategy document, entitled the "ENIQ 2020 Roadmap" (January 2011). As final result from this discussion process the ENIQ voting members agreed to integrate ENIQ into NUGENIA, making ENIQ the 8thTechnical Area of NUGENIA.

ENIQ will continue to be recognised as a highly respected and efficient entity supporting safe and reliable long term operation of European nuclear facilities through the effective application of NDT techniques, including qualification and RI-ISI.

9.1.1 Scope

The European Network for Inspection and Qualification (ENIQ) is a network driven by the nuclear utilities of the European Union member countries and Switzerland and has been existing since 1992.

ENIQ is active on the field of in-service inspection (ISI) of nuclear power plants (NPP) by non-destructive testing (NDT), and is working mainly in the areas of qualification of NDT systems and risk-informed in-service inspection (RI-ISI). The technical work is performed by two task groups, the Task Group on Qualification (TGQ) and the Task Group on Risk (TGR).Both task groups are made up of experts from Europe with additional experts from Canada and the USA.

ENIQ has a steering committee (SC), which is the decision making body of ENIQ. The SC determines the development of ENIQ and approves all documents issued by the two task groups. The SC has twelve voting members, one for each EU member country and Switzerland. They are the SC members who take decisions and they all come entirely from European utilities. Thus voting members come from Belgium, Czech Republic, Finland, France, Germany, Hungary, the Netherlands, Slovakia, Spain, Sweden, Switzerland and United Kingdom. Beside the voting members the SC has non-voting members (observers) from Canada, USA, the chairmen of the three task groups and additional representatives of European utilities, vendors or research organisations. In 2010, the ENIQ SC recognised that the European nuclear industry was entering a period of significant change and thus initiated an internal discussion to determine its vision and objectives regarding ENIQ's future role and activities. This exercise resulted in the issuing of a strategy document, entitled the "ENIQ 2020 Roadmap" (January 2011). As final result from this discussion process the ENIQ voting members agreed to integrate ENIQ into NUGENIA, making ENIQ the 8thTechnical Area of NUGENIA.

ENIQ will continue to be recognised as a highly respected and efficient entity supporting the safe and reliable long term operation of European nuclear facilities through the effective application of NDT techniques, including qualification and risk-informed ISI.ENIQ will continue to foster and maintain collaboration with the global nuclear industry in advancing and harmonising best practices.

9.1.2 Objectives

The European Network for Inspection and Qualification (ENIQ) is a network driven by the nuclear utilities of the European Union member countries and Switzerland and has been existing since 1992.

ENIQ is active on the field of in-service inspection (ISI) of nuclear power plants (NPP) by non-destructive testing (NDT), and is working mainly in the areas of qualification of NDT systems and risk-informed in-service inspection (RI-ISI). The technical work is performed by three task groups, the Task Group on Qualification (TGQ), the Task Group on Risk (TGR) and Task Group for Inspection

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Qualification Bodies (TGIQB). Their members come from utilities, ISI vendors, IQBs or research organisations in Europe with additional members from Canada and the USA

ENIQ has a steering committee (SC), which is the decision making body of ENIQ. The SC determines the development of ENIQ and approves all documents issued by the three task groups. The SC has twelve voting members, one for each EU member country and Switzerland. They are the SC members who take decisions and they all come entirely from European utilities. Thus voting members come from Belgium, Czech Republic, Finland, France, Germany, Hungary, the Netherlands, Slovakia, Spain, Sweden, Switzerland and United Kingdom. Beside the voting members the SC has non-voting members (observers) from Canada, USA, the chairmen of the three task groups and additional representatives of European utilities, vendors or research organisations. In 2010, the ENIQ SC recognised that the European nuclear industry was entering a period of significant change and thus initiated an internal discussion to determine its vision and objectives regarding ENIQ's future role and activities. This exercise resulted in the issuing of a strategy document, entitled the "ENIQ 2020 Roadmap" (January 2011). As final result from this discussion process the ENIQ voting members agreed to integrate ENIQ into NUGENIA, making ENIQ the 8thTechnical Area of NUGENIA.

ENIQ will continue to be recognised as a highly respected and efficient entity supporting the safe and reliable long term operation of European nuclear facilities through the effective application of NDT techniques, including qualification and risk-informed ISI.ENIQ will continue to foster and maintain collaboration with the global nuclear industry in advancing and harmonising best practices.

9.1.3 State of the art

9.1.3.1 NDT Qualification

ENIQ can be recognized as one of the main contributors to today's global qualification standard along with other initiatives like the Performance Demonstration Initiative (PDI) of the United States of America. Both inspection qualification methodologies (ENIQ and PDI) goal is to probe NDT inspection techniques procedures capabilities to detect defects on nuclear power plants components, but from two different approaches, mainly practical (mock-ups and real test) from the PDI point of view and more theoretical (physical reasoning, past inspection experience, numerical simulation and technical justification) from the ENIQ perspective. As such, ENIQ has always been playing a first order role on qualification state of the art, creating the first qualification methodology based on technical justifications, issuing recommended practices (RP) and carrying out pilot studies.

At present, the main challenges for qualification are mutual recognition of qualification approaches between countries and qualification of new NDT technique procedures based on phased array ultrasonic testing, time of flight diffraction ultrasonic testing, computed radiography, etc. These topics represent the main areas of interest for the near future of TGQ.

9.1.3.2 Risk Informed Inspection

ENIQ has been a very active contributor to risk informed programmes expansion, providing guidelines about the definition of RI-ISI programmes and participating in the RISMET project of OECD-NEA to benchmark different RI-ISI methodologies [2] to investigate to what extent they deliver different results. Another project investigated the link between RI-ISI and inspection qualification. The present challenges on the RI field are RI on pre-service inspection (PSI) for new build, risk reduction quantification and optimization of ISI intervals. TGR is currently working on these issues or will do so in the near future.

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9.1.4 Challenges

Apart from the above technical challenges ENIQ, in particular its SC, has to ensure that ENIQ documents (methodology documents and RPs) remain valid and relevant, enabling adequate use of the documents.

To meet this requirement ENIQ SC has to develop a formal procedure for periodic review of each document, even if there are no changes required (the review process should reflect this).The actual revision of the documents i

The main pending challenges are:

On qualification:

- Ensure a robust link between inspection and inspection qualification and the input processes such as code requirements, safety cases,
- Provide guidance on inspection qualification,
- Ensure that inspection results are used effectively to inform the safety assessments,
- Precise the role of Condition monitoring by NDT to improve maintenance of plants.
- Support extended application of computer modelling and simulations.
- Extend application of NDT qualification to other NDT methods.

On risk :

- Improve the link between RI-ISI and Inspection Qualification,
- Precise the role of RI-ISI in Defence-in Depth (pressure vessels, qualification approach, risk, etc.).
- Define the RI-ISI for new build (including pre-service inspections).
- Implement the RI-ISI and qualification guidance for non-pressure boundary items.
- Adopt RI-ISI guidance to optimise ISI intervals.
- Develop more detailed guidance for the use of expert knowledge in RI-ISI and qualification.
- Assess guidance on the use of alternative methods (to ISI) for managing risk.
- Implement Probabilistic Safety Assessment (PSA) (technical adequacy for RI-ISI, refinement for RI-ISI programmes).
- Implement and adopt guidance on Expert Judgement

9.2 Sub Technical Areas (STA)

9.2.1 Task Group on qualification (STA 7.1)

This is the task group responsible for having developed the inspection qualification methodology that is now being used as a basis for all European methodologies and the one for CANDU type reactors. The ENIQ inspection qualification methodology is also accepted by the IAEA as recommended practice to be followed for nuclear inspection qualification all over the world.

After publishing the third issue of the European Qualification Methodology Document in 2007, TGQ has recently issued a RP on personnel qualification and a document giving an overview of inspection qualification for the non-specialist. Currently TGQ is involved in a number of projects.

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9.2.1.1 Computed and Digital Radiography Project

Computed radiography (CR) and digital radiography (DR) are two contenders as replacement technologies for traditional film-based industrial radiography. One aspect of the CR/DR method that makes it attractive is its linear detection characteristics and the consequential ability to reduce the radiation exposure (which in turn reduces inspection times and the potential radiological hazard).However, the transition from traditional silver film to CR/DR requires an in-depth understanding of the relative performances of CR/DR and film based radiography. As detection, processing and interpretation of the two methods are significantly different there are many technical aspects that need to be explored.

Whilst some studies have explored the relative performances of the radiographic methods there has not been a comprehensive assessment of CR /DR in relation to inspection qualification. Thus the aim of the study is to identify the essential parameters that affect the performance of CR and DR and thereby providing a consistent approach to inspection design and the production of technical justifications

The project would begin with a review of the current state of the art of CR and identify the key aspects of these systems that determine the relative performance. This would comprise:

- ✓ **contributions from technical experts within TGQ**
- ✓ **new theoretical and practical studies**

The output of the project would be an ENIQ Recommended Practice that would be used by inspection designers and qualification bodies.

This is a short term project (up to 2 years).

9.2.1.2 Phased Array Project

Ultrasonic phased arrays provide significant improvements over conventional ultrasonic probes for a wide range of applications. These benefits range from the ability to generate several fixed beam angles from within a single probe, beam focussing and the provision of detailed defect characterisation through signal processing.

The current approach to qualifying inspections using phased arrays is to apply the same principles and processes as for conventional methods on a case by case basis. With the increasing use of phased array probes there is substantial benefit to be gained through a more formalised approach.

The essential parameters for phased array probes are more numerous and complex than for conventional, fixed angle probes and an extensive study of these is required to ensure that the claimed inspection reliability is assured in practice.

The project would begin with a review of the current state of the art on phased array probes and identify the key aspects that determine performance. This would comprise:

- ✓ **contributions from technical experts within TGQ**
- ✓ **new theoretical and practical studies**

The output from the project would be a new Recommended Practice on the application of phased arrays for high reliability inspections and their associated qualification.

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This is a short term project (up to 2 years).

9.2.1.3 Guided Waves Assessment

The goal of this project is an independent assessment of the accuracy of GWUT for NDT inspections. The results of this evaluation will be used by ENIQ members to make an informed decision regarding the relevance of the deployment of this technique.

The project would begin with a review of the current state of the art on GWUT identifying key aspects to determine its performance. This would comprise:

- ✓ **contributions from technical experts within TGQ**
- ✓ **new theoretical and practical studies**

The output from the project would be a new Recommended Practice on the application of Guided Wave for high reliability inspections and their associated qualification.

This is a short term project (up to 2 years).

9.2.1.4 Pilot Study on Mutual Recognition

The European Methodology for inspection qualification was initially designed as a framework that would enable each country to establish its own detailed practices that matched the specific national requirements (regulatory, plant type, resources etc.). Whilst being extremely successful in achieving this objective, the inevitable specific nature of the qualification processes has introduced significant obstacles in the ability to transport qualifications between countries. For example, ISI (In-Service Inspection) vendors that have qualified their inspection systems in one country are then required to repeat the qualifications in another even if the plant is similar. Consequently, one of the major tasks faced by ENIQ is to understand the technical barriers that preclude the transport of qualifications and to overcome these.

Due to its importance for licensees and nuclear safety this activity is being progressed both within TGQ and by TGIQB (see below). An important benefit of the project would be to provide evidence that will be noted by regulatory bodies which may lead to changes in regulatory approaches. It is expected, as a minimum, that this initiative taken by licensees may influence regulators in different countries to increase their co-operation.

The first stage of this process is to understand the key aspects relating to the conduct of inspection qualification as it is performed in each of the countries participating in ENIQ. To this end, ENIQ TGQ has undertaken a survey relating to key aspects relating to the inspection qualification requirements and processes. The next stage is to conduct a pilot study to explore in detail how differences in approach can affect the different aspects of the qualification process:

The project contents are:

- ✓ **Choose a case (real or fictitious) to be sent to every single ENIQ member to carry out a qualification according to their national rules, guidelines and methodologies.**
- ✓ **Compare the results from the different rules, guidelines and methodologies.**

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- ✓ **Generate a report pointing out differences between qualification requirements in different countries and with guidelines/recommendations giving instructions on how to extend the NDT procedure qualification of one country so that is accepted by another country.**

This is a medium term project (up to 5 years).

9.2.1.5 Simulation Software for NDT Inspection Qualification

Simulation software is used all over the world as first step for inspection procedures development and sometimes as final step to verify inspection capabilities of a technique to detect a defect , because of their relatively small costs compared with mock-ups.

An independent assessment to verify simulation software for NDT inspections accuracy is needed. The results of such an evaluation will be used by ENIQ members to make an informed decision regarding the relevance of the modelling software output to their particular applications (Qualification of NDT Systems).

The project contents are:

- ✓ **List NDT inspection simulation software used in the nuclear industry**
- ✓ **Compare results from the different software on similar configuration, to see if all of them are acceptable**
- ✓ **Prepare a pilot study with all the accepted software to compare their results with real results obtained while carrying out an inspection on a mock-up**

This is a medium term project (up to 5 years).

9.2.1.6 High Density Polyethylene (HDPE) NDT Technologies

High Density Polyethylene (HDPE) has showed itself as a promising alternative to cast iron in tertiary cooling water systems of NPPs, since HDPE avoids most of corrosion and erosion problems discovered among cast iron piping. The greatest challenge concerning HDPE is its accurate and reliable inspection.

The project will focus on benchmarking and focusing at the promising NDT System and on investigating and evaluating new NDT technologies.

The project contents are:

- ✓ **Generate a report on actual NDT techniques used for HDPE inspection and the strengths and weaknesses**
- ✓ **Analyse other NDT techniques suitable to assess HDPE component accurately**
- ✓ **Prepare a pilot study with HDPE flawed mock-ups**

This is a medium term project (up to 5 years).

9.2.1.7 NDT Capabilities in Concrete

As nuclear power plants long term operation is becoming a reality, concerns about concrete structures start to arise because of their difficulty to be inspected as metallic components.

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The project is to develop a process for evaluating the reliability of commercially available inspection techniques and how to implement a qualification process for those techniques on concrete components.

The project contents are:

- ✓ **Generate a report on actual NDT techniques used for concrete inspection and the strengths and weaknesses**
- ✓ **Analyse other NDT techniques suitable to assess concrete component accurately, taking into account degradation mechanism and expected defects**
- ✓ **Prepare a pilot study with concrete flawed mock-ups**

This is a medium to long term project (up to 7-8 years).

9.2.1.8 Future Projects

The main working areas on qualification for the future are:

- ✓ **Ensure a robust link between inspection and inspection qualification and the input processes such as code requirements, safety cases.**
- ✓ **Provide guidance on inspection qualification issues to address concerns and queries raised by members.**
- ✓ **Ensure that inspection results are used effectively to inform the safety assessments made by others.**
- ✓ **The role of Condition monitoring by NDT in order to improve maintenance of plants.**
- ✓ **Support extended application of computer modelling and simulations.**
- ✓ **Extended application of NDT qualification to other NDT methods.**

9.2.2 Task Group on Risk (STA 7.2)

In 2005, TGR published the "European Framework Document on RI-ISI", and has since been working at producing more detailed Recommended Practices (RPs) and discussion documents on several RI-ISI related issues. Amongst them is RPs on the verification and validation of structural reliability models and guidance on the use of expert panels together with discussion documents on the application of RI-ISI to the inspection of the reactor pressure vessel and updating of RI-ISI programmes.

9.2.2.1 RI-PSI Project

In view of the on-going and planned new build of NPPs in Europe ENIQ members suggested to look more closely at RI-PSI for new build. This supports the development of RI-ISI and qualification guidance for new build in general. The contents of this project are:

- ✓ **Review development in ASME and during Olkiluoto construction.**
- ✓ **Produce ENIQ discussion document and recommend practice.**

It is a medium term project (up to 5 years) with its first milestone (state of the art review) being a short term (up to 2 years).

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9.2.2.2 PoD Methodology Project

The use of probability of detection (PoD) curves to quantify the reliability of NDT systems is slowly finding its way into the nuclear industry. The need to quantify the output of inspection qualification has become more important, especially as structural reliability modelling and quantitative RI-ISI methodologies become more common and the PoD provides a metric for quantifying ISI reliability. Despite recent progress in the development of PoD methodologies adequate guidance on the use of PoD for practitioners is still missing. The project contents are:

- ✓ **Produce specifications for quantitative PoD.**
- ✓ **Review and assess approaches for producing PoD.**
- ✓ **Pilot study on application of Monte-Carlo approach for producing PoD.**

This is a medium term project.

9.2.2.3 Justification of Risk-Reduction through ISI Project

Despite RI-ISI being well established, an analysis on its role for defence-in-depth and an assessment of the achievable level of risk reduction with RI-ISI are still missing. The aim of the study is to quantify the benefit from ISI and to indicate the level of risk reduction that can be expected from ISI. The evaluation of the risk reduction will be performed by using structural reliability models and probabilistic analyses. The influence of uncertainty level of loadings, material properties, damage growth, PoD curves and inspection intervals will be assessed. The project contents are:

- ✓ **Produce discussion document on the role and expectations of ISI.**
- ✓ **Determine level of risk reduction that is achievable through ISI.**
- ✓ **Role of quantitative PoD in risk reduction.**

This is a medium term project (up to 5 years).

9.2.2.4 Future Project

The main areas on Risk for future work are:

- ✓ **Improve the link between RI-ISI and Inspection Qualification (e.g. to establish how risk reduction criteria could set ISI requirements, quantify the output of a qualified inspection, etc.).**
- ✓ **The Role of RI-ISI in Defence-in Depth (pressure vessels, qualification approach, risk, etc.).**
- ✓ **RI-ISI for new build (including pre-service inspections).**
- ✓ **RI-ISI and qualification guidance for non-pressure boundary items.**
- ✓ **RI-ISI guidance to optimise ISI intervals.**
- ✓ **Develop more detailed guidance for the use of expert knowledge in RI-ISI and qualification.**
- ✓ **Guidance on the use of alternative methods (to ISI) for managing risk.**
- ✓ **Probabilistic safety assessment (PSA) (technical adequacy for RI-ISI, refinement for RI-ISI programmes).**

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- ✓ **Guidance on Expert Judgement.**

9.2.3 Task Group on Qualification Bodies (STA 7.3)

Structural Integrity claims for safety critical nuclear plant often depend upon having a high level of confidence in NDT results, performed either during manufacture or in-service. The inspection qualification process that provides this confidence requires the function of a qualification body that acts as a third party independent assessment organization. As such the qualification bodies that are established by each country are in the best place to provide advice to licensees on the processes and application of inspection qualification. Consequently, the licensees represented in ENIQ have decided to set up a separate task group for qualification bodies to improve qualification practice and to provide a consensus view to licensees.

The role of the Task Group for Inspection Qualification Bodies (TGIQB) is to provide a forum for the free exchange of information between qualification bodies and to identify and conduct R&D activities that are targeted at improving the efficiency and effectiveness of approaches for establishing confidence in NDT.

TGIQB will have its inaugural meeting in March 2013 and this will include a discussion on technical tasks that need to be performed, a prioritization of these tasks and the development of a work programme. Below are some of the tasks that the TGIQB may consider.

9.2.3.1 Mutual Recognition Qualified Inspections

The European Methodology for inspection qualification was initially designed as a framework that would enable each country to establish its own detailed practices that matched the specific national requirements (regulatory, plant type, resources etc.). Whilst being extremely successful in achieving this objective, the inevitable specific nature of the qualification processes has introduced significant obstacles in the ability to transport qualifications between countries. For example, ISI (In-Service Inspection) vendors that have qualified their inspection systems in one country are then required to repeat the qualifications in another even if the plant is similar. Consequently, one of the major tasks faced by ENIQ is to understand the technical barriers that preclude the transport of qualifications and to overcome these.

Due to its importance for licensees and nuclear safety this task is being progressed both within TGQ (see above) and by TGIQB. The aim of the TGIQB aspect of this project is to first understand, through an open forum, how qualification bodies undertake there specific tasks, including:

- ✓ **The development of "generic" inspection specifications. Here the inspection specification is the document that provides the details of the component to be inspected, the defect types that are sought by the inspection and the performance requirements for the inspection. As such, the inspection specification is the foundation document for the inspection designers and the document against which the qualification body assesses inspection performance. Presently, there is no consistency for inspection specifications for different countries, even in cases where the plant design is the same. A project to improve the consistency in the inspection specifications, including the format, type of content and technical details would provide a great benefit to licensees.**

- ✓ **Developing a consensus for the design of practical trials and test piece production for procedure and personnel qualification. Currently, there are different approaches for**

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simulating the deliberately implanted defects in test pieces ranging from machining slots through to fully representative cracks. Individual qualification bodies have undertaken specific development activities to assess the viability of various defect implantation techniques. It is clear however, that a wider and more coordinated R&D activity would be beneficial in arriving at techniques that are acceptable to a wider range of qualification bodies. This would include an assessment of what defect features must be simulated in certain cases and developing techniques for inserting defects.

- ✓ **Formulating methods for the assessment of NDT modelling. The modelling of NDT is becoming increasingly used for inspection design and to justify the NDT performance. Consequently there is a strong need to assess the validation status of such models. Whilst specific exercises have been undertaken to validate certain aspects of models, a substantial project that considers wider aspects of model validation would provide substantial benefits to easing the transport of qualified inspections.**

9.2.3.2 Improved Inspection Qualification Practice

TGIQB will provide a vital opportunity for open discussions between qualification bodies that will identify where improvements can be made in the qualification process and identify how these improvements can be realised.

The ENIQ Methodology for inspection qualification is a relatively mature process and whilst experience shows that improvements can be made, any such change will need to be based on sound technical evidence. Pilot studies, undertaken during the early stages of ENIQ's development, were important in developing the ENIQ Methodology and associated Recommended Practices. For example one of the pilot studies informed a Recommended Practice on how modelling should be used in inspection qualification.

9.3 References

[TA8-1] ENIQ 2020 Roadmap. ENIQ Report Nº 43. EUR 24803 – 2011

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10 Outcomes from the FP7-NUGENIA+ Project

Coordinator: E.K. Puska (VTT)

The objective of the NUGENIA+ project is to support the NUGENIA Association in its role to coordinate and integrate European research on safety of the Gen II and III nuclear installations in order to better ensure their safe long term operation, integrating private and public efforts, and initiating international collaboration that will create added value in its activity fields.

The project consists of two parts, the first part being a **Coordination and Support Action** and the second part **a Collaborative Project**. The aim of the first part, **the Coordination and Support Action,** is to establish an efficient, transparent and high quality management structure to carry out the planning and management of R&D including project calls, proposal evaluation, project follow-up dissemination and valorisation of R&D results in the area of safety of existing Gen II and future Gen III nuclear installations.

The preparatory work will encompass governance, organizational, legal and financial work, as well as the establishment of annual work plans, with the aim to structure public-public and/or private-public joint programming. The management structure will build on the existing organisation of the NUGENIA Association, grouping nuclear organisations from research and industry (utilities, vendors and small and medium enterprises) active in R&D.

In the second part, the **Collaborative project**, one thematic call for research proposals has been organized and took place one year after the start of the project. The call implemented the priorities recognised in the NUGENIA Roadmap, in line with the Sustainable Nuclear Energy Technology Platform (SNETP) and International Atomic Energy Agency (IAEA) strategies. The research call was open to all eligible organisations.

10.1 NUGENIA+ Pilot Call Contents

There are two outcomes from NUGENIA+ EC FP7 project on research topics and priorities. The first outcome is the definition of the subject areas for the open public NUGENIA+ research project proposal call that was opened on September 22, 2014 and had the deadline of November 28, 2014[19].

The content of the call addressed eight topics, which are in accordance with the high level objectives listed in pages 53 and 54 of the NUGENIA-Roadmap published in 2013. They are identified as cross-cutting through the various technical areas of NUGENIA. The main intention was to bring researchers in different areas together in order to work on a high level R&D issue. As proposals both small projects (short term) with a defined goal and small R&D work to consolidate and draft a large project (long term) were welcomed.

[19]Twenty-six research proposals were received by the deadline. The proposals are reviewed by independent evaluators using the evaluation criteria defined by NUGENIA+ project in the NUGENIA+ Call Reference Document number 5 and accepted by the EC. After due acceptance by the NUGENIA+ project and the EC the selected proposals will become new tasks of the WP6 of the NUGENIA+ (preparing NUGENIA for Horizon 2020) project –Grant agreement N°: 604965. The expected funding for each selected proposal will be up to 200 k€ from the total of 2.6 M€ of EC funding reserved for this call in the NUGENIA+ project budget. The duration of the NUGENIA+ funded projects can vary, but only a maximum of 18 months will be funded. All NUGENIA+ funded activities have to be completed by 31 August 2016. There must be at least one deliverable and one milestone at the completion of the NUGENIA+ funded activities, which is the final month 36 of NUGENIA+ project.

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The eight cross-cutting items with their high level and short term objectives were defined as:

10.1.1 Improve safety in operation and by design

The high level objective is to identify preventive and protective measures against all sources of external or internal events and identify the way to efficiently and effectively implement them in current and future reactors. The short term objective is to comply with the conclusions of the recently conducted Nuclear Power Plants (NPPs) stress tests and will be a precursor to further work that will be required in order to meet the long term objective.

10.1.2 High reliability of components

The high level objective is to ensure the safe operation of components in Gen II and III NPPs through high reliability by technological development in the fabrication process for structural and fuel components resulting in improvements in subsequent reliable maintenance and affordable inspectability. The short term objective (relating to this call) is to undertake a preliminary project based on the approach given below as a pre-curser to further work that will be required in order to meet the long term objective.

10.1.3 High reliability and optimized functionality of systems

The high level objective is to ensure safe operation of systems in Gen II and III NPPs through high reliability and optimized functionality by producing unified European wide guidance for nuclear energy stakeholders. The short term objective (relating to this call) is to undertake a preliminary project based on the approach given below as a pre-curser to further work that will be required in order to meet the long term objective.

10.1.4 Improve modelling of phenomena in NPPs

The high level long term objective is to demonstrate the reliability and predictability of the advanced simulation codes based on the interaction and coupling of different physical processes and providing them with the opportune and extended validation for design needs and safety-assessment use relying upon existing data base from mock-up experiments and operation feedback. The short term objective is to provide benchmarking cases (inter-comparison between codes on theoretical cases of calculations or application to a given experiment) to validate advanced simulation codes and tools for specific application to simulate real plant operation, performance, ageing phenomena and safety situations using the approach given below as a pre-curser to further work that will be required in order to meet the long term objective.

10.1.5 Increase public awareness

The high level objective is to address the rationale behind nuclear energy acceptance and public resistance, taking into account the differences in energy policies and in the public awareness which exist among the different European countries. The short term objective is the identification of the main drivers towards public acceptance and resistance using the approach given below as a pre-curser to further work that will be required in order to meet the long term objective.

The outcome of the project will be the production of guidelines for efficient management of educational material in schools, informative papers and conferences and the development of strategies to involve the public in the political decision making process in relation to energy choices involving nuclear power.

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Emphasis should typically be placed on:

- ✓ dissemination and transparency of information,
- ✓ consultation of the public,
- ✓ increase openness to the public on safety,
- ✓ increase public involvement on research relating to nuclear energy systems,
- ✓ contribution to public acceptance.

10.1.6 Efficient integration of NPPs in the energy mix

The long term objective is to use the energy sources in the most efficient way. NPPs are basically designed to operate under constant load without large load cycles. To cope with the unstable grid caused by the energy mix, innovative operation of the LWRs has to be designed which considers the power manoeuvrability of NPPs without disturbance of NPPs operability. The short term objective is to identify the consequences of the new operation mode specifically on the performance of components and systems.

10.1.7 Prepare the future to avoid technology obsolescence

The long term objective is to accurately identify key components or systems where obsolescence needs to be avoided because of the impact on NPP safety and availability. Obsolescence mitigation procedures and recommendations need to be developed. The short term objective (relating to this call) is to identify the key components and systems in NPPs that cannot readily be replaced (e.g. because of requirements from codes and standards) based on current operation experience.

10.1.8 Performance and ageing of NPPs for long term operation

The long term objective is to obtain enhanced understanding of the ageing degradation mechanisms and make available approaches and tools for effective monitoring and mitigation to guarantee that the ageing effects are properly managed or analysed (e.g. by time limiting ageing analysis (TLAA)). The short term objective (relating to this call) is to undertake a preliminary project on how to extrapolate the ageing related data and to assess the applicability of methodologies to cover 60+ years in order to meet the long term objective.

The full call document (NUGENIA+ project deliverable D1.1 and NUGENIA+ Call reference document number 1 at the public page at www.nugenia.org) provided further advice on the desired scope and approach of the proposals in these eight subject areas.

10.2 NUGENIA+ Framework plan for year 2

The second outcome of the NUGENIA+ project on research priorities and topics is the NUGENIA+ Framework plan for year 2 of NUGENIA+ project. The document provided in full as the deliverable D1.2 of the NUGENIA+ project describes the state of the art or the research in the eight technical areas of NUGENIA and describes the identified short, mid and long-term research needs in each respective area including the subareas. The prioritization of the research needs is also indicated as well as the existing research proposals responding to these needs and priorities in the NOIP or as already on-going research projects.

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The full deliverable D1.2 is found under the link /my proposal is to add it as a link, not as copy[20].

[20] For the updated information the reader is urged to check the NOIP on those high priority areas that do not have any proposed or on-going research projects and that may have specified request for proposals.

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11 Contribution to the SNETP Deployment Strategy

NUGENIA program has been constructed as a set of Technical Areas [TAs] to cover all R&D Gen II- III issues, while seeking for improved safety and performance. Installations in operation, under construction and current innovative designs have been considered within the European fleet, mostly relaying upon LWR technology.

As a complementary approach to TA specific topics and challenges widely addressed in the present document, cross cutting high level objectives of NUGENIA program have also been identified and were the subject of the FP7-NUGENIA+ project. For the deployement strategy document, being drafted, NUGENIA has made a substantial effort in identifying the milestones associated with these objects as well as the time frame needed to accomplish them as listed in the table below:

High level objective	Technical objective	(TA specific) challenge	Expected major milestone (T0 + X y)
Improve safety in operation & by design	Minimize the impact of internal and external loads and hazards on the safety functions	Improve methodologies to assess impact on barriers, structures, systems and components considering single and multiple events,	T0 + 5 - 10y
	Eliminate accidental sequences that could yield in very important consequences	Developing methods to better assess the probability of rare events and their consequences	T0 + 5 - 10 y
	Develop advanced safety assessment methodologies	Integrating deterministic and probabilistic safety assessments in order to better quantify safety margins with best estimate methods	T0 + 10 y
		Implementation of stress test in Europe	T0 + 5 - 10 y
	New systems for mitigation of consequences of severe accidents	-Identification -Validation & qualification	T0 + 5 -10 y
High reliability & optimized functionality of systems	Operational excellence	Innovative asset management approaches, sharing of best practices	T0 + 5 - 10 y
	Reliability and security of digital systems	Maintaining the necessary cybersecurity level by continuous improvement	T0 + 5 - 10 y
	Reliability of NPPs as complex socio-	Development of system resiliency concept	

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	technical systems	(interaction of safety-human organization capabilities– I&C systems)	T0 + 5- 10 y
High reliability of fuel	Accident tolerant fuel	Identification of candidate materials for fuel & cladding & component	T0 +5 y
		Advanced material assembly in test reactor	T0+ 10 y
		Production of lead test assembly on accident tolerant fuel type	T0+ 20 y
High reliability of structural component	Increased resistance of materials under severe and/or more stringent conditions	- Advanced surface engineering technology -Advanced capabilities for in depth characterization and long life time assessment -Advanced /innovative material with multi functions	T0 + 5 – 10 y T0 + 5 – 10 y T0 + 10 – 15y
	Increase integrity of components	Develop improved methods for assessing integrity of systems, structures and components	T0 + 5 – 10 y
	Advanced and new process for easy fabrication - manufacturing and assembly - mitigation solution	-As low as possible defect in component fabrication -Master surface treatment -Qualify Powder metallurgy process for nuclear application	T0 + 5 – 10 y
	Equipment qualification & control	-Advanced NDE -Instrumented component from design & fabrication to installation -Advanced methods for on site surveillance	T0 + 5-10 y continuous
	Fully validated codes for severe accidents	-Improved modelling of severe accidents phenomenology and management -System code and CFD codes validation - support existing facilities and build new ones when necessary	Continuous T0 + 5 – 10y and continuous

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Improve modelling phenomena in NPPs	Develop predictive software platform based on multi physics and multi scale modelling	Advanced capabilities & methods in material behavior – neutron physic – fluid dynamics- chemistry -Coupling between different phenomena - Provide accurate test results for validation and qualification -	T0 + 5-10y and continuous evolution
Efficient integration of NPPs in the energy mix	Define NPPs role in a country-specific generation mix	Assessment of functions for stabilization of transmission grid	T0 + 5 y
	Identify consequences of higher flexible operations on NPP management and cost	Impacts of dynamic loading on material ageing Improvement of core and fuel management Impacts on performance characteristics and development of economic strategies	T0+ 5-10 y
	Implementation of flexible operations on existing plants – Flexibility by design for new build	Continuous plant modification : I&C – component management – fuel cycle Implement measures allowing minimizing grid instability risks	T0 + 5 – 15 y T0 + 10 – 15 y
Performance and ageing of NPPS for long term operation	Demonstrate structural integrity of NPPs components at regular intervals throughout life time	- Reliable design curves valid for environmental conditions -Plant data to underpin the safety case sin structural integrity by surveillance programs -Advanced capabilities for load evaluation: fluid to structure interaction – Pressurized thermal shock….. -Advanced capabilities and methods for accurately predicting material ageing with best estimate margin (chemistry – irradiation – thermal ageing – fatigue –	T0+ 5-10 y T0+5-10y T0+ 5 – 10y

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		crack initiation…)	
	Develop on line /on site monitoring & diagnostic - NDE	- Support PLIM- PLEX in implementing structural health monitoring	T0 + 5-10y
		Instrumented component from design & fabrication to installation	T0 + 10 - 15y
		- Crack detection beyond 60 years	T0+10 -15y
		Update In core measurement - SPND	T0 + 5 - 10 y
Prepare the future to avoid technology obsolescence	Continuously update technology and practices Foster harmonization	-Ensure technology transfer and dissemination -Incorporate innovative technology -Update codes & standards through pre normative research	continuous
Increase public awareness	Dissemination and transparency of the information, especially on safety	New information channels and use of social media to increase public awareness	T0 + 5 y continuous
		New participatory approaches in decision making	T0 + 5 – 10 y continuous

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12 Overall Conclusion

The present NUGENIA global vision Document - a living one - is aimed at providing the NUGENIA community and, more wider, the whole GEN II GEN III R&D community worldwide with a quite detailed description of the technical and scientific content of the NUGENIA's 8 Technical Areas that cover the current GEN II &GEN III research needs in a quite comprehensive way, without scaling[21] and prioritizing, while recalling their general scope and state of the art, addressing their main R&D objectives, underlining their main challenges in the medium and long term and identifying the most important cross-cutting and interface / edge problems. It also outlines and sketches a preliminary tentative prioritization of actions in tight relationship with the EC FP7 NUGENIA+ project, now underway.

The document has been elaborated and written by the NUGENIA's Technical Area Leaders (the TALs) under the coordination of the Technical Coordinator (TA), with the valuable contribution from the members of the NUGENIA Secretariat and the Executive Committee (ExCom), as well as the Sub-Technical Area leaders (STALs) and some volunteer participants in the TA activity. The main references, abbreviations and contributors to the tasks are indicated in the document, too.

The NUGENIA Roadmap Working Document is to be issued by March 2015, approved by the NUGENIA General Assembly and disseminated within the NUGENIA's partners and outside at the NUGENIA's 2015 Forum.

It has been agreed among NUGENIA's Partners that the document should undergo a periodic revision and up-dating process on three to five-year basis to allow its content accounting for the progress in the research and matching with the new R&D challenges to provide a revision of the document

Accordingly next revision is expected, upon the ExCom decision, by 2018-2020. In the mean-time a revision of the NUGENIA Roadmap 2013 should also be prepared and issued so as to establish and impose coherence along these two complementary, fundamental and structuring documents of NUGENIA's R&D activity.

[21] The scaling and prioritization work is to be done adopting the present docuyment as an extensive and comprehensive information source. The actual work is actualkly within the NUGENIA+ projet now underway

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Abbreviations

AFCEN	French Association for standarisation
AGR	Advanced gas-cooled reactors
ASME	American Society of Mechanical Engineers
ATF	Accident Tolerant Fuel
BDBA	Beyond Design Basis Accident
BPD	Best practice documents
BWR	Boiling Water reactor
CANDU	(CANada Deuterium Uranium-type reactor
CBM	Condition based maintenance
CCF	Common cause failures
CEB	Comité Européen du Béton
CEN	European Committee for Standardization -
CENELEC	European Committee for Electro-technical Standardization
CFD	Computational Fluid Dynamics
GCR	Gas-cooled reactors
CHF	Critical heat flux
CR	Conversion ratio
CRg	Computed Radiography
DB	Design Basis
DBA	Design Basis Accident
DBE	Design basis emergency
DCF	Dual-cooled fuels
DCH	Direct Containment Heating
DEC	Design Extension Conditions
DiD	Defence-in-depth
ECC	Emergency core cooling
ECR	Equivalent Cladding Reacted
EERA JPNM	European Energy Research Alliance Joint Program for Nuclear Material
EIA	Environmental Impact Assessment
ENEF	European nuclear energy forum
ENIQ	European Network for Inspection and Qualification
ENISS	European Nuclear Installations Safety Standards
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
ESNII	European Sustainable Nuclear Industrial Initiative
ESS	European Standardization System
ETC	embrittlement trend curve
ETSON	European Technical Safety Organisation Network
EU	European union
FAC	Flow assisted corrosion
FBR	Fast breeder reactor
FCI	Fuel Coolant Interaction

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FGR	Fission gas retention
FORATOM	European Atomic Forum
FP	Fission Product
FP6, 7	6th, 7th Framework Programme of the European Commission
FTA	Failure Tolerance Analyses
GenII	Generation II reactor
GWUT	Guided Wave Ultrasonic
HCFLWR	High conversion factor light water reactor
HIP	Hot isostatic pressing
HOF	Human and Organizational Factors
HRA	Human reliability assessment
HRP	Halden Reactor Project
HWR	Heavy Water Reactor
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IAGE	Integrity assessment and ageing
ICRP	International Commission on Radiological Protection
IDPSA	Integrated Deterministic-Probabilistic Safety Analysis
IGD-TP	Implementing Geological Disposal of radioactive waste Technology Platform
IGSCC	Inter granular stress corrosion cracking
IMF	Inert Matrix Fuel
INPO	Institute of Nuclear Power Operations
IQB	International Qualification Bodies
ISA	Internal sulphate attack
ISI	In Service Inspection
ISP	International Standard Problem
ISTC	International Science Technology Centre
ISTP	International Source Term Programme
J	J-Integral
K	Stress intensity factor
LBB	Leak before Break
LERF	large early release frequency
LLW	low level waste
LOCA	Loss of Coolant Accident
LP	Lumped Parameter code
LRF	large release frequency
LTAs	lead test assemblies
LTO	Long Term Operation
LWR	light water reactor
MA	Minor Actinides
MCC	Motor control centre
MCCI	Molten Core Concrete Interaction
MOX	Mixed-oxide fuels
MSLB	Main steam line break

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NDE	Non Destructive Examination
NDT	Non Destructive Testing
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
ODS	Oxide dispersion strengthened steels
OECD	Organisation for Economic Co-operation and Development
PA	Phased Array
PAR	Passive Autocatalytic Recombiners
PCI	Pellet clad interaction
PCT	Peak Cladding Temperature
PIE	Post irradiation Examination
PLEX	Plant life extension
PSA	Probabilistic Safety Assessment
PSI	Pre-Service Inspection
PTS	Pressurize thermal shock
PWR	Pressurised Water Reactor
PWSCC	Primary water stress corrosion cracking
R&D	Research and development
RAFM	Reduced activation ferritic/martensitic steels
RAG	Alkali aggregate reaction
RCC-M	The RCC-M French design code
RCS	Reactor Cooling System
RHR	Residual heat removal
RI	Risk Informed
RIA	Reactivity Initiated Accident
RIDM	Risk Informed Decision Management
RP	Radiation protection
RcP	Recommended Practice
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SA	Severe accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SARNET	Severe Accident Research NETwork of Excellence
SC	Steering Committee
SCC	stress corrosion cracking
SFP	Spent Fuel Pool
SMA	Safety management assessment
SMR	Small modular reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SPS	Spark Plasma Sintering
SSC	System, Structure, Component
STA	Sub-technical area
TA	Technical area

TG	Task Group
TGIQB	Task Group IQB
TGQ	Task Group Qualification
TGR	Task Group Risk
TMI2	Three-Mile-Island unit 2
TOFD	Time of Flight Diffraction
TSO	Technical Safety Organisation
VHTR	Very high temperature reactor
VLLW	Very Low Level Waste
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
WNA	World Nuclear Association
WWER = VVER	Water- water energy reactor

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NUGENIA is an international association under Belgian law established in 2011 to promote R&D on Gen II & III nuclear reactors. NUGENIA primarily serves the needs of European nuclear industry and has more than 100 member organisations from industry, research, safety organisations and academia. The activities of NUGENIA cover plant safety & risk assessment, severe accidents, reactor operation, integrity assessment and ageing of systems, structures & components, development of fuel, waste & spent fuel management & reactor decommissioning, innovative light water reactor design & technologies, harmonization and in-service inspection & their qualification.

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