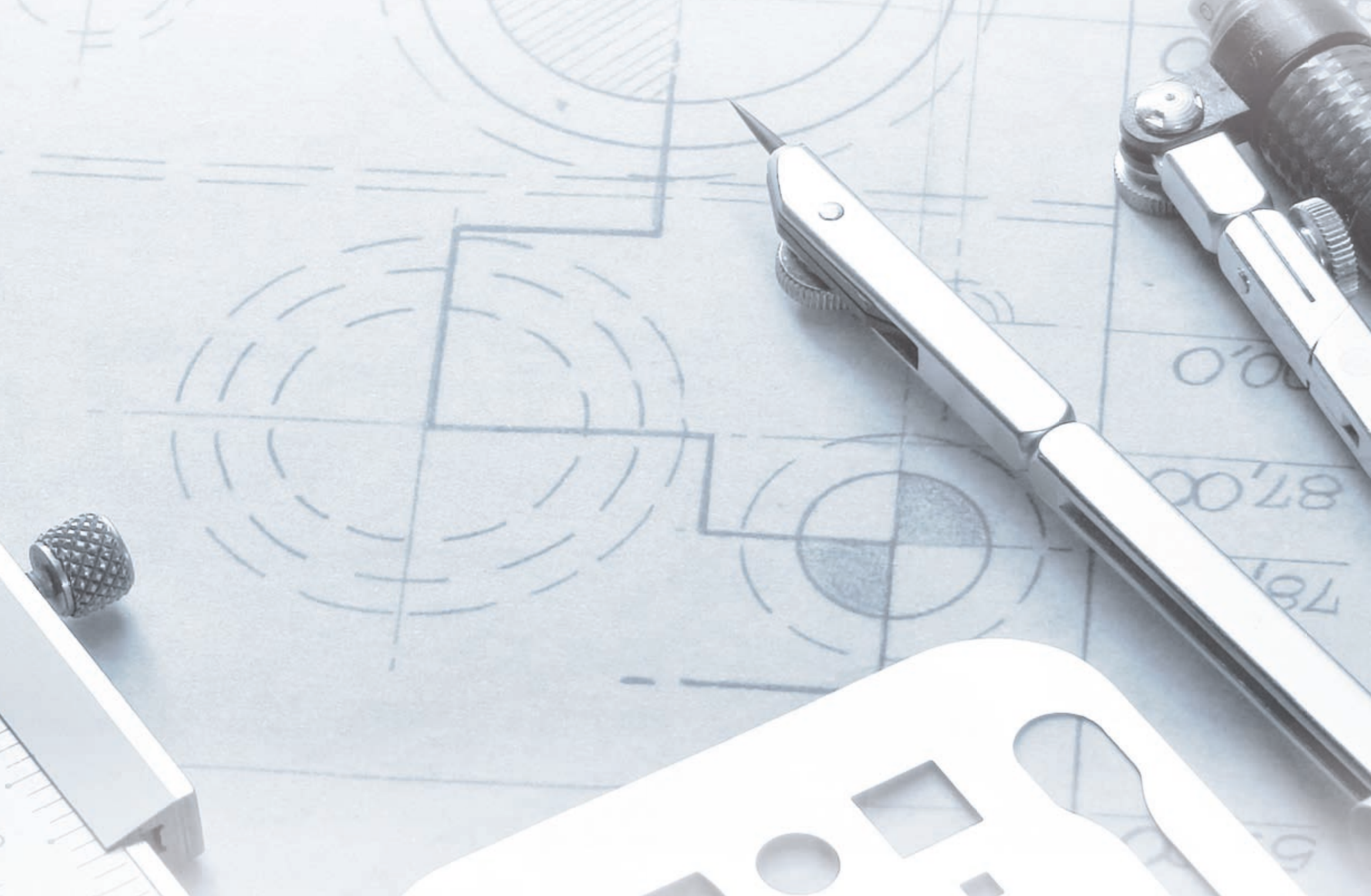
Topical Issues in Nuclear Installation Safety

Safety Demonstration of Advanced Water Cooled Nuclear Power Plants

Proceedings of an International Conference
6–9 June 2017, Vienna, Austria

Vol. 1



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TOPICAL ISSUES IN NUCLEAR INSTALLATION SAFETY

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The Agency’s Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is “to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world’’.

PROCEEDINGS SERIES

TOPICAL ISSUES IN NUCLEAR INSTALLATION SAFETY

SAFETY DEMONSTRATION OF ADVANCED WATER COOLED NUCLEAR POWER PLANTS

PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 6–9 JUNE 2017

In two volumes

VOLUME 1

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2018

Marketing and Sales Unit, Publishing Section
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1400 Vienna, Austria
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tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/books

Printed by the IAEA in Austria
August 2018
STI/PUB/1829

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Topical issues in nuclear installation safety / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2018. | Series: Proceedings series (International Atomic Energy Agency), ISSN 0074–1884 | Includes bibliographical references.
Identifiers: IAEAL 18-01171 | ISBN 978–92–0–104618–5 (paperback : alk. paper)
Subjects: LCSH: Nuclear reactors — Safety measures — Congresses. | Nuclear reactors — Design and construction. | Reliability (Engineering)
Classification: UDC 621.039.58 | STI/PUB/1829.

FOREWORD

The IAEA held the International Conference on the Safety of Nuclear Power: Strategy for the Future in Vienna in 1991. Recommendations from this conference prompted actions in subsequent years that advanced the safety of nuclear installations worldwide and included the establishment of the Convention on Nuclear Safety, which entered into force in 1996. In 1998, the IAEA organized the first of a series of international conferences on topical issues in nuclear safety. Subsequent conferences in the series have taken place in Vienna (2001 and 2013), Beijing (2004) and Mumbai (2008). These conferences have contributed significantly to the exchange of information and experience on the latest advances in the field of nuclear installation safety.

The sixth IAEA International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants was held in Vienna, 6–9 June 2017. Its purpose was to foster the exchange of information on the latest approaches, advances and challenges in the demonstration of the safety of nuclear power plants, in particular those using water cooled reactors, including small and medium sized or modular reactors. The conference focused on the safety demonstration of nuclear power plants that have been or soon will be licensed and constructed, which includes, among other aspects, the establishment of, and compliance with, comprehensive and rigorous requirements for siting, design and operation; the demonstration of adequate safety margins against external hazards; and a robust and reliable design to prevent early radioactive releases or radioactive releases large enough to require long term protective measures and actions.

The introduction of passive safety systems, digital instrumentation and a number of innovative safety features in the designs, as well as the inclusion of severe accidents in the design envelope of the new plants, are some of the developments that pose crucial challenges to the safety demonstration and licensing of new reactors. All these aspects are of central interest to design organizations, nuclear regulators, plant operators and technical support organizations in Member States.

More than 300 participants from 48 Member States and 5 international organizations attended the conference, and its programme included 100 paper presentations and 18 posters. There were several side events and two high level plenaries — one on the Vienna Declaration and one on insights gained from the design, construction and commissioning of advanced water cooled reactors. The number of contributions reflects the strong interest in the topic. Of particular relevance was the participants' frank and open exchange of views and experiences that will benefit the further enhancement of nuclear safety. The key insights and recommendations obtained, as summarized by the Conference President, will also shape future work on nuclear installation safety.

This publication, organized in two volumes, provides the executive summary of the conference including the key outcomes and recommendations, together with the papers presented. The IAEA officers responsible for this publication were C. Spitzer of the Division of Nuclear Installation Safety and S. Monti of the Division of Nuclear Power.

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EXECUTIVE SUMMARY

Outline

Over the years, the IAEA has organized a series of international conferences on topical issues in nuclear installation safety. The conferences have yielded recommendations and led to activities that have served to increase international cooperation and to promote the exchange of vital information to enhance nuclear safety.

The sixth *IAEA International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants* took place in Vienna, Austria, 6 – 9 June 2017.

The purpose of the conference was to foster the exchange of information on the latest approaches, advances and challenges in the demonstration of the safety of nuclear power plants that are planned to be licensed and constructed in the near future, in particular those using water cooled reactors, including small and medium sized or modular reactors. This conference in the series was focused on the safety demonstration of the nuclear power plants (NPPs) that have been and will be licensed and constructed in the near future, which includes, among other aspects, the establishment of, and compliance with, comprehensive and rigorous requirements for siting, design and operation; the demonstration of adequate safety margins against external hazards; and a robust and reliable design to prevent early radioactive releases or radioactive releases large enough to require long term protective measures and actions.

The introduction of passive safety systems, digital instrumentation and a number of innovative safety features in the designs, as well as the inclusion of severe accidents in the design envelope of the new plants, are some of the developments that pose crucial challenges to the safety demonstration and licensing of new reactors.

The conference aimed to provide a platform for the interchange of experiences that can provide valuable insights into how the topics covered by the conference are currently addressed in different countries for various types of stakeholder organizations in Member States. Accordingly, the conference intended to contribute to the harmonization of approaches and methods applied for the safety demonstration of nuclear power plants worldwide.

The conference covered safety assessment of advanced reactor designs, design safety principles, licensing of advanced reactor designs and safety reinforcement of existing installations. The content of the technical programme is summarised in detail below. It was comprised of presentations accompanied by discussions on the four topical areas, several side events and two plenaries - one on the Vienna Declaration and one on Insights gained from Design, Construction and Commissioning of Advanced Water Cooled Reactors.

The high-level plenary titled *Vienna Declaration on Nuclear Safety: Objectives, Challenges and Prospects* featured discussions on how to implement in practice the principles in the 2015 declaration, which aims to strengthen work to prevent accidents with radiological consequences and mitigate such consequences should they occur.

The conference participants' recommendation that the IAEA facilitate the application of the new design safety requirements by Member States would "support the harmonisation of approaches and methods applied to nuclear power plants that are planned to be licensed and constructed in the near future," said Conference President Petteri Tiippana, Director General of the Finnish Radiation and Nuclear Safety Authority.

"We now need Member States' to adopt a bottom-up approach and share their experiences and practical approaches also regarding safety improvements for existing nuclear power plants in the context of the Vienna Declaration," Mr. Tiippana continued.

The conference recommended that IAEA collect both positive and negative regulatory and industry experiences and lessons for nuclear power programmes, including new-build projects, he said adding that participants encouraged the Agency to continue to provide fora for discussions among Member States to strengthen international cooperation and knowledge transfer.

The IAEA Deputy Director General Juan Carlos Lentijo, Head of the Department of Nuclear Safety and Security, emphasized the participants' frank and open-minded exchange of views and experiences benefited the further enhancement of nuclear safety.

"The conference provided valuable insights on challenges and progress related to technical and scientific matters on topics such as innovative design features. It also enhanced the understanding of how to meet new safety requirements, for example the practical elimination of early or large radioactive releases and the need for design for potential core-melt scenarios," he said.

A final synthesis concluded that the wide variety of topics discussed at the Conference demonstrated the broad interest of the global nuclear safety community, and the strong need for discussions such as those held during the Conference.

Plenaries

Opening Plenary

The opening plenary, introduced by Ms. Cornelia Spitzer (IAEA) included the opening remark by the IAEA Director General Mr. Y. Amano and the opening statement by Mr. P. Tiippana, Director General of the Finnish Radiation and Nuclear Safety Authority and Conference President.

The speakers welcomed the conference participants and expressed their expectation on fruitful discussions among the large audience on the latest approach and challenges in demonstrating safety at nuclear power plants on topics such as safety assessment and licensing of advanced reactor designs, design safety principles and safety reinforcement of existing installations. The discussions had a particular focus on nuclear power plants using water cooled reactors, including small and medium sized or modular reactors.

Several considerations on the role of nuclear power in the context of increasing safety and minimizing accidents were addressed in the opening statements. A common element which can be highlighted from the opening speeches is the general belief that international collaboration, peer and safety reviews and sharing of operating experience make the nuclear infrastructure stronger as a whole. It was emphasized that the IAEA can contribute to strengthen this collaboration through the development of the IAEA safety standards by integrating current technology and best practices, and providing for their application to achieve a high level of safety.

The IAEA Director General Y. Amano delivered several important messages to the audience at the conference opening remarks as follows:

> *"Good morning, Ladies and Gentlemen, Mr. Tiippana, Dear Colleagues,*
>
> *I am pleased to welcome you to the sixth IAEA International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants. This Conference is part of a series on nuclear safety which the IAEA has organized since 1998.*
>
> *Nuclear installation safety is of global importance as nuclear accidents can have effects across borders. This makes licensing and supervision of nuclear power plants a concern not only for operating nations, but also for countries near and far.*
>
> *This Conference provides a platform for discussions on issues such as safety assessment and licensing of advanced reactor designs, design safety principles, and safety reinforcement of existing installations.*
>
> *Nuclear power plants are designed with the goal of minimising the likelihood of accidents and ensuring that – if an accident should occur – its consequences can be mitigated. A comprehensive safety assessment is essential to ensure the protection of workers, the public and the environment.*
>
> *Over the coming days, you will consider the latest advances and challenges in demonstrating the safety of nuclear power plants that are expected to be licensed and built in the near future*
>
> *There will be a particular focus on plants using water cooled reactors, including small and medium sized or modular reactors. Another important topic will be the implications of the Vienna Declaration on Nuclear Safety for operating nuclear power plants.*
>
> *Ladies and Gentlemen,*
>
> *Nuclear power makes a significant contribution to reducing greenhouse gas emissions and improving energy security, while delivering energy in the growing quantities needed for development.*
>
> *Global use of nuclear power continues to grow, despite the Fukushima Daiichi accident in 2011. At present, 30 countries are using nuclear power. About 30 others are considering building their first nuclear power plant, or have started doing so. Most of these possible newcomers are developing nations.*
>
> *IAEA safety standards establish fundamental principles, requirements and recommendations for ensuring nuclear safety. They serve as a global reference for protecting people and the environment.*
>
> *We have revised requirements on safety assessment and design safety, and a revision of associated safety guides is underway.*
>
> *I encourage all countries to make full use of the many services offered by the IAEA in nuclear safety.*

Our education and training programmes help to strengthen Member States' capacities in nuclear safety, including in design safety.

We offer expert peer reviews on topics such as generic and plant-specific designs, national design requirements, safety assessments and periodic safety review programmes.

Ladies and Gentlemen,

I am pleased to see so many participants here today.

We look forward to your insights and recommendations, which will help to shape our future work on nuclear installation safety.

I wish you every success with your discussions and I look forward to learning about the outcome.

Thank you."

The Director General of the Finnish Radiation and Nuclear Safety Authority and Conference President Mr. P. Tiippana highlighted some rationale for this conference, and in particular why the safety demonstration of a nuclear installation remains a topical issue for the nuclear safety:

"Demonstrating nuclear installation safety is no simple task, and it is becoming more and more difficult and challenging due to new technologies and updated safety requirements.

We need to make sure safety standards are applied, and to identify both good practices and areas of improvement in conducting safety demonstrations.

With the ever-evolving digitalization of almost everything in people's daily lives, the nuclear industry is also considering the use of novel measures to enhance operational performance of nuclear installations.

The industry has started exploring the possibility of using big data to enhance the reliability and safety of their nuclear installations. The nuclear community has been very good at collecting data, and if we look into the possibilities and challenges, we might be able to discover nuanced approaches in the engineering and operations of nuclear power plants to make them safer in a way we could not imagine before."

The Vienna Declaration on Nuclear Safety: Objectives, Challenges and Prospects

The first plenary, chaired by Mr. P. Tiippana, Director General of the Finnish Radiation and Nuclear Safety Authority and Conference President, focused on the objectives, challenges and prospects of the Vienna Declaration on Nuclear Safety (VDNS); moreover, it provided an opportunity to present and discuss open issues, challenges of achieving the objectives of the VDNS and possibly different understanding from technical point of view. The plenary was composed of the following esteemed panellists:

- D. Drabova, Czech Republic;
- J. C. Niel, France;
- A. Kawano, Japan;
- A. Lyubarskiy, Russian Federation;
- M. Johnson, United States of America.

The panellists were requested to provide statements on the following topics:

- Major objectives, challenges and possible shortcomings in the context of the IAEA safety standards as well as national requirements and regulations to implement the principles of the VDNS;
- Possible difference related to the terms "avoiding", "preventing", "practical elimination" used in the VDNS, the IAEA safety standards and/or national regulations; need for further discussion / clarification / harmonisation to better deal with / understand the consequences on the design and operation of different types of nuclear power plants (NPPs);
- Meaning of "early" and "large" releases in practice for the siting, design and operation of the NPPs; establishment of definitions / requirements / regulations in the respective country to meet the objective; consideration of site related factors, such as population distribution, evacuation routes, etc. in the safety assessments;
- Understanding and view of "reasonably practicable" and "achievable" safety improvements in the context of "to be implemented in a timely manner"; meaning in practise, in particular for existing NPPs; interpretation and implementation throughout the lifetime of a NPP, dependence on the life

cycle phase (for instance periodic safety review (PSR) after first ten years of operation vs. subsequent PSR after 20 years etc. or a decision on progressing to long term operation (LTO) phase); practical guidance for what would constitute as timely implementation.

In the subsequent discussion the content of the Vienna Declaration was largely discussed by the panellists and participants elaborating on the clarification of the objectives of the declaration, and the challenges of its implementation.

As general conclusion of this plenary, the panellists and participants agreed that the revised IAEA safety standards well reflect the objectives of the VDNS. Panellists and participants recognized that sharing information is one of the best ways to maintain permanent focus on improving safety. It was recommended that Member States pursue a proactive, bottom up approach in sharing experiences and practical approaches regarding safety improvements for existing NPPs in line with the objectives of the VDNS.

It was noted that similar conclusion was drawn from the Commission on Safety Standards already in 2015 and reiterated at the 7th Review Meeting on the Safety Convention on Nuclear Safety.

Extensive discussions on the concepts of practical elimination and reasonably practicable as well as on the definition of early or large releases were held. The general consensus was that it is important for each Member State to have a process in place to perform a safety assessment and determine if those concepts have been adequately and effectively implemented.

During the discussions the safety improvements to existing nuclear power plants and what is considered sufficient to meet the objectives of the VDNS were identified as a main challenge related to the backfitting of nuclear power plants in operation.

A number of Member States requested further support and assistance from the IAEA in implementing the latest design safety requirements at nuclear power plants in operation.

Insights Gained from Design, Construction and Commissioning of Advanced Water Cooled Reactors

Newcomer countries and countries with established nuclear programmes are currently engaging in a number of construction projects at various stages of completion globally. These projects have encountered a diverse range of challenges, including difficulties in keeping first-of-a-kind (FOAK) realization on budget and on schedule. As such, the second plenary, chaired by Mr. A. Bychkov, Russian Federation, focused on sharing insights from the design, manufacturing, construction and commissioning of advanced water cooled reactors. The plenary was composed of the following esteemed panellists:

- M. Zheng, China;
- T. E. Jin, Korea, Republic of;
- A. Kiryukhin, Russian Federation;
- A. Bradford, United States of America;
- G. Rzentkowski, International Atomic Energy Agency;
- P. Vincze, International Atomic Energy Agency.

The plenary was intended to present specific cases of design, manufacturing, construction and commissioning of advanced water-cooled reactors, as well as to share international experience. Each of the panellists provided short presentations focused on the following topics:

- Impact of design change and/or design finalization during construction;
- Challenges related to a suitable supply chain;
- Construction and commissioning management and related risks;
- Modern construction technologies and methods: advancements vs challenges;
- Good practices and lessons learned to overcome challenges during design, construction and commissioning of advanced NPPs.

The panellists noted that while challenges exist, they are well identified, and as such, vendors, engineering, procurement and construction (EPC) companies, operators and regulators are aware of the focal areas of contention and have found effective ways to manage these challenges accordingly. Throughout the discussion, the importance of a stable and well established regulatory framework to succeed in the completion of a nuclear project was stressed. The discussion focused on the following main challenges:

- *Complexity of designs*: A mature technology should pursue simplification rather than an increasing number of systems, sub-systems and components;

- *Supply chain*: Reliable and well qualified supply chain and appropriate oversight is a must;
- *Design changes during construction:* Sometimes managed by means of licensing amendments. This suggests the need to well define the technical specifications of the plant and finalize the design prior to starting construction;
- *Complexity of the construction phase:* Integration of thousands of people from hundreds of organizations with different organizational culture;
- Some challenges have arisen from peculiar features/advantages of the advanced water cooled reactor technology like modularization and the adoption of passive safety systems.

Keynote – ETSON: Its Role and Activities for Harmonising Safety Assessment

The Keynote, delivered by Mr. B. De Boeck, ETSON, and chaired by Mr U. Stoll, Germany, provided the participants with the opportunity to receive information on the background, role and main activities of the European Technical Safety Organisations Network (ETSON). Among the objectives defined for their future activities, ETSON aims at strengthening links with the IAEA, in particular in the Technical and Scientific Support Organizations (TSOs) Forum and the TSOs conference, as well as their participation in the development and revision of IAEA safety standards through their member organisations in Member States.

The need for independence between the nuclear power plant design developer and the reviewer of the safety demonstration in relation to the key role of the TSOs in this field was emphasized. Participants also stressed that expertise in safety assessment for newcomer countries is necessary and therefore encouraged new comer countries to establish relationships with competent and well experienced external organizations, to develop and implement education programmes and training courses in the early phase of a new nuclear power programme in order to build capacity for performing all the activities necessary to ensure safety in licensing, construction and operation.

Closing Plenary

The closing plenary was comprised of a closing remark by the IAEA Deputy Director General of the Department of Nuclear Safety and Security J. C. Lentijo and the closing statement by Mr. P. Tiippana, Director General of the Finnish Radiation and Nuclear Safety Authority and Conference President.

In his closing remarks, the President of the Conference focused on the key outputs and recommendations. Mr. Tiippana highlighted that international collaboration, peer and safety reviews and sharing of operating experience remain essential elements to improve the nuclear safety of the installations; and he emphasized the need for international collaboration specifically as it relates to new designs and new build projects. Sharing is valuable for all the stakeholders and can make the industry stronger as a whole. The President of the Conference also reminded that the IAEA technical safety review services contribute to the enhancement of nuclear safety by providing an independent review on different subject areas; he encouraged Member States to further explore and utilize these services in a systematic approach.

The key outputs delivered during the conference summary are as follows:

- A Common approach is needed to assess the reliability of safety systems relying on passive concepts;
- Fora for discussion on approaches to demonstrate safety in core melt scenarios are recommended;
- The development and verification of safety demonstration tools represent a priority for Member States;
- The use of a quality Probabilistic Safety Assessment (PSA) is recommended to enhance safety;
- Further guidance is needed to address the concept of "practical elimination" and its demonstration;
- The importance of assessing the implementation of defence-in-depth in design is highlighted;
- The Review of the applicability of the IAEA safety standards to SMR designs is requested;
- Internationally accepted methods to evaluate new design features are desired;
- Consideration for multi-unit interactions are recommended (e.g. SMRs, but not limited to).
- The sharing of experiences in the licensing of passive systems is recommended;
- Further improvements to severe accident management programmes are recommended;
- The sharing of experiences with backfitting measures implemented at nuclear power plants in operation to meet the objective of the Vienna Declaration on Nuclear Safety to the extent practicable is recommended;

- Further clarification on the terminology of the VDNS is desired.

The conference recommendations highlighted by the Conference President are summarised as follows:

- Member States should share experiences and practical approaches (e.g. related to the VDNS and the evaluation of new design features) and the IAEA should provide fora to strengthen international cooperation and knowledge transfer;
- The IAEA should facilitate the application of new design safety principles, also related to small and medium sized or modular reactors (SMRs);
- IAEA should collect regulatory and industry experience and lessons, positive and negative ones (e.g. New build Projects);
- The IAEA should continue to support the harmonization, verification and validation and common approaches.

The closing remark by Deputy Director General Lentijo remarked on the impressive participation at the conference and the excellent quality of the scientific contributions. Closing his speech, gratitude was expressed in particular to the Conference Secretariat, who provided advice on the scope, overall objectives, structure or the conference and to the Scientific Organizing Committee, who set up the detailed conference programme, identified key speakers, and last but not least, selected and peer reviewed over 100 scientific contributions. The joint efforts of the chairpersons, the Scientific Organizing Committee and the Conference Secretariat were fundamental in making the conference a success.

The IAEA Deputy Director General of the Department of Nuclear Safety and Security J. C. Lentijo closing remarks are as follows:

Good afternoon, ladies and gentlemen. On behalf of the IAEA Director General, I thank you for your participation in this sixth IAEA International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants. I am glad that more than 300 participants from 48 Member States and 5 International Organisations are here.

The conference has been intense, with 100 papers and 18 posters presented. There were several side events and two high-level plenaries - one on the Vienna Declaration and one on Insights gained from Design, Construction and Commissioning of Advanced Water Cooled Reactors. This packed programme reflects the strong interest in the topic.

The Conference covered topics, ranging from new reactor projects to nuclear power plants in operation addressing safety assessment and licensing of advanced reactor designs, design safety principles, and safety reinforcement of existing installations. We had fruitful, open and beneficial discussions on the experiences and challenges of demonstrating the safety of nuclear power plants that are planned to be licensed and constructed in the near future, in particular those using water cooled reactors, including small and medium sized or modular reactors.

Valuable insights were obtained for both, challenges and advances from technological and scientific point of view, and on the application of the new safety requirements.

The Conference also featured a useful discussion of the implications of the Vienna Declaration on nuclear safety for nuclear power plants in operation.

This wide variety of topics demonstrates the broad interests of the global nuclear safety community. The Director General in his opening remarks emphasised the global importance of nuclear installation safety along with the IAEA Safety Standards and review services available to the Member States. I invite you to take advantage of these services.

We welcome the interest in the Agency's safety standards in the area of safety assessment and design safety, and particularly in their ongoing revision as well as the development of supporting technical documentation. I encourage all countries to participate in these activities.

The key insights and recommendations obtained, as just outlined by the Conference President, will enable us to shape our future work on nuclear installation safety. We will focus on facilitating the application of the new safety requirements by building on the practical experiences from Member States in order to support the harmonisation of approaches and methods. We will also continue to provide fora for technical and scientific exchanges among Member States to strengthening international cooperation and knowledge transfer. As always, the Secretariat stands ready to assist the Member States in working to address their challenges.

My department – the IAEA Department of Nuclear Safety and Security - cooperated with the IAEA Department of Nuclear Energy to prepare this Conference. I recognize the instrumental role of the Scientific Secretaries: Ms. Cornelia Spitzer and Mr Stefano Monti. Ms. Julie Zellinger of Conference Services support was essential in organizing the Conference. Special thanks go at first to the President of *the Conference, Mr. Tiippana, as well as to the members of the Scientific Organising Committee and to all the panellist, speakers, chairpersons and poster presenters for your effort. Thanks to you, this Conference was very successful.*

Thank you for taking part in this Conference. I wish you a safe and pleasant journey home or wherever your travels may take you.

I hereby declare the Conference closed.

Side Events

Workshop on Technical Safety Review Services

The workshop was intended to share experiences from both the IAEA and Member States representatives about the Technical Safety Review (TSR) services. The IAEA presentation summarized the scope, intent and current status of the TSR services and was followed by presentations and discussion with each panellist. There was a consensus view amongst the panellists that the TSR services benefit Member States by providing tailored, independent evaluation of the safety assessment and design safety documentation and making recommendations for enhancements and improvements to nuclear safety. TSR services encompass six subject areas including design safety, generic reactor safety, safety requirements, probabilistic safety assessments, accident management and periodic safety reviews. The significance of these services was reinforced by the fact that they are based on IAEA safety standards which represent an international consensus view on an appropriate level of safety. Several panellists also emphasized that the benefit comes not just from the review itself, but from the review preparations and work undertaken to address observations and recommendations.

Questions from the audience primarily focused on the scope and documentation required for the TSR services. Both the panellists and the IAEA staff reinforced that the TSR service does not constitute any kind of design certification or licensing activity as this is not a function of the IAEA; rather, it is the responsibility of the Member States. It was pointed out that the TSR services are primarily intended to improve the quality of the documentation being considered by providing recommendations in areas were supplementing information or modifications are needed to adhere to the IAEA safety standards. An example of this discussion focused on the TSR review of design safety documentation. This TSR services does not constitute a review of the design itself, but, rather, a review of the quality of how the documentation demonstrates that the design adheres to the IAEA safety requirements which are utilized as the review criteria.

Clarification regarding the role of the IAEA safety guides and the involvement of design organizations was discussed for inclusion in the TSR Services Guideline document that is currently under preparation. In response to this discussion, the IAEA will circulate the draft TSR Services Guideline document for comments prior to finalising and publishing. Several panellists also encouraged Member States to request a TSR service to partake in the benefits such recommendations and observations provide.

Workshop on an Introduction and Further Explanation on Design Extension Conditions

The workshop introduced and discussed the application of design extension conditions (DECs) as described in IAEA Safety Standard SSR-2/1, Nuclear Power Plants: Design. During the discussion, Member States presented their approach for implementing DEC into their safety requirements. Consensus was found in regards to categorizing accident conditions caused by multiple failures or those exceeding capability of the safety systems as DECs. Member States agreed that designing dedicated safety features for DEC with less conservatism was acceptable provided justification of sufficient margin to cover uncertainties and avoid cliff edge effects exist.

Further engagement on the effectiveness of provisions credited in the safety analysis of DEC was discussed and it was determined that this may be demonstrated by the application of rules that are less penalizing than those applied to design basis accident (DBA) analyses. Further discussion concluded that a comprehensive safety demonstration of the DEC analyses must be included in the safety analysis report (SAR) regardless of whether or not the consequences are mitigated by the operation of safety systems unaffected by the DEC sequence.

To conclude, it was determined that most of the DEC without significant fuel damage are dependent on the reactor technology and design. As such, a systematic and comprehensive approach should be implemented and documented to justify the postulated DECs for the design of a nuclear power plant with the objective to reinforce the plant capabilities to prevent accident with core melting and to meet the total core damage frequency target with a reliable confidence.

Roundtable on SMR Deployment: Technical, Construction and Licensing Challenges

A roundtable was held to discuss the technical, construction and licensing challenges facing the deployment of SMRs. The IAEA provided a short presentation to introduce small modular reactor (SMR) technology and provide a comprehensive overview of the designs being pursued and Member State involvement. Following which, each of the five panellists provided an overview of their countries SMR programme, including those in the Russian Federation, United Kingdom, United States of America, People's Republic of China, France and Switzerland.

The panellists discussed the challenges and opportunities SMRs face as a first-of-a-kind technology. One such challenge lies in the manufacturing of SMRs, as SMRs tend to shift the terrain from on-site stick-built to factory built nuclear power plants. The importance of economies of scale was also discussed given the reduced power output of SMRs and the optimization of passive safety systems results in lower maintenance and staffing costs.

Challenges and opportunities facing the regulatory scope specifically related to the use of non-conventional reactor types, like modular high temperature gas and molten salt reactors, were largely. For instance, uncertainty associated with accident progression and their types was emphasized. An opportunity to streamline the harmonization of safety standards was recognized taking into account that the design safety requirements established for the nuclear power plants will need to be adapted to consider specificities of SMR designs. The need to harmonize design safety requirements on an international scale to improve economic viability was also highlighted as challenge but considered by the participants as highly desirable. The participants agreed that SMRs could be designed so that the implementation of protective actions for the people and the environment would not be necessary in accident conditions. The panel also discussed the importance of assisting newcomer countries through IAEA services and partnerships with developed nuclear power plant programmes.

Many SMR designs are proposed to be used for applications other than electricity generation like desalination, district heating, etc. As such, the need to understand and minimize the possibility of cascading effects during accidents for multi-unit plants and those in close proximity to chemical plants were discussed. As was the need to investigate the introduction of non-radioactive hazards, such as chemical and biological hazards that these applications and innovative designs may present.

In regards to floating reactors, it was emphasized by the panellists that proven reactor designs will be utilized. However, it was stated that much emphasis needs to be placed of resolving legal aspects related to transporting a reactor in international and domestic waters and the liability associated with doing so. Further accident analysis must also be considered as it relates to capsizing, terrorist attacks, etc.

Topical Areas

Safety Assessment of Advanced Reactor Designs

A wide range of aspects was covered by the papers presented, including atmospheric dispersion of radioactive materials, dynamic assessment of facilities' contamination and failure in accidental conditions, single-phase direct numerical simulation, reactor pressure vessel (RPV) neutron fluence assessment and severe accident sequences analysis. Relevant efforts are being made in advanced modelling of physical phenomena focused on closing the gaps introduced by the unavailability of modelling tools, such as those related to atmospheric dispersion of radionuclides and dynamic probabilistic safety assessment (PSA). Evidence related to international efforts to enhance codes, such as those for multiscale simulation and safety demonstration of advanced water-cooled reactors, was provided. Relevant activities are being implemented at the experimental level to provide new sets of data for severe accident codes benchmarking. The IAEA's role in contributing to identification of mechanisms for facilitating the selection and sharing of good quality data for code validation (most of them protected by proprietary rights) was emphasized. Data for severe accident analysis was one of the specific aspects highlighted, which included both plant data (e.g. Fukushima Daiichi NPP) and data from experimental installations. Relevance of the need to further develop full parallel deterministic transport codes was also specifically identified. In summary, continuing development and corresponding verification of safety demonstration tools was considered as still needed.

Safety systems relying on passive phenomena to accomplish their safety functions are being introduced with innovative design safety features, and thus, there is a high interest from Members States to develop methods to evaluate the reliability of these passive systems. During the discussions, the critical role passive systems play in ensuring nuclear safety, especially in advanced reactor designs, was noted and the need for thorough assessment of passive systems reliability was highlighted. In spite of the potential advantages of passive systems, they imply several challenges in the demonstration of their reliability. In particular, evaluation of passive systems performance could challenge computer codes by phenomena that could be beyond their validation domain; therefore, the need for validating the computer codes against phenomena involved in passive systems was underlined. It was specifically noted that the validation of computer codes needs to be done on the basis of both separate effect tests and integral effect tests considering any scaling effects. Another challenge that caused concerns among participants is related to the interactions between passive systems and necessity of their investigation. It was concluded that integral experimental facilities including different types of passive systems are required for proper investigation of interactions between passive systems. Several sessions of the conference touched upon this issue and eventually the conclusion made was that there is a need for a common approach to assess the reliability of passive systems. The participants further agreed that the use of innovative technology and components should necessitate specific qualification tests and analyses to demonstrate efficacy and reliability of those components and systems.

Another topic that raised a large interest among various Member States representatives was the in-vessel melt retention (IVMR) strategy. In general, it was noted that the robust safety demonstration of effectiveness of IVMR strategy should include margins and consider a large spectrum of influencing factors and account for uncertainties (especially for high power reactors). During the discussions, it was revealed that in addition to the national, regional and international programmes that are underway to reduce the uncertainties related in particular to IVMR, there is a need for a sound technical basis to support a robust safety demonstration of corium stabilization and cooling in case of an accident with core melting. The IAEA is supporting this effort by providing platforms for discussion and fostering information sharing and dissemination.

The safety assessment of SMRs was discussed in regards to the need to develop an understanding of how to demonstrate the safety of these designs, and moreover, how to determine an appropriate level of uncertainty where unproven methods are utilized. The need to develop validation codes for modelling was discussed in great detail. Third party validation and international collaboration was deemed vital to ensuring a high level of effectiveness in this regard. Further understanding and development regarding design basis accident and severe accident management were discussed, as well as the possibility to investigate decreasing the required emergency planning zone size.

A deterministic approach supplemented by probabilistic insights and feedback from operation remains a good practice widely used for the identification of postulated initiating events (PIEs). The safety demonstration proving the compliance of the design with the regulatory requirements is performed on the basis of a set of deterministic analyses and PSAs, they represent complementary means to provide a comprehensive view of the overall safety of the plant.

Several papers and associated intensive discussions were specifically devoted to the use of PSA to support the design process of advanced NPPs. It was noted that the approaches for PSA modelling have grown and changed in parallel with the evolution of NPP designs; however, the direction of PSA applications has not changed significantly. While the main technological solutions are provided based on the deterministic considerations, the PSA is mainly used to balance the design, reveal hidden vulnerabilities, optimize and justify Limiting Conditions for Operation (LCOs), verify compliance with system reliability targets and other applications. In general, it was mentioned that enhanced designs have been achieved by continuous consultations between design and system engineers and PSA teams, concluding that the use of PSA in design is a highly interactive process.

In the meantime, participants highlighted the critical importance of the quality of the PSA models used in the process of plant design and operation. It was specifically noted that current PSA models have to deal with new modelling challenges such as modelling of passive systems, the assessment of reliability of passive systems and digital I&C systems and others. In addition, it was underlined that high quality PSAs are achievable only through the comprehensive and independent review process. In this context, Member States were recommended to request the IAEA Technical Safety Review – Probabilistic Safety Assessment (TSR-PSA) service in order to receive independent PSA review based on the IAEA safety standards.

The need for consideration of multi-unit interactions in safety assessment, both by probabilistic and deterministic means, was specifically highlighted.

Design Safety Principles

The identification of design extension conditions (DEC), rules and criteria used in the design of safety features for DEC was addressed and discussed in several presentations. The presentations showed that the identification and assessment of DECs differ from one country to another for the same basic design.

Uncertainty and safety margin evaluation including cliff edge effects were discussed in terms of their better and common understanding and definitions with the aim of soundly including them in the best-estimate plus uncertainty analysis also recommended by the IAEA. Although this topic was discussed for power reactors, it also applies to SMRs.

The interpretation and demonstration of practical elimination of large or early releases was addressed in a specific session of the conference and the presentations led to intensive discussions. The origins of the concept and its evolution were highlighted in the presentations. The concept of practical elimination could be considered as a relevant part of the application of defence-in-depth principle in order to ensure that the likelihood of accident conditions that could lead to early or large radioactive releases is extremely low. There is a need to develop further clarification in the practical application of the concept and demonstrating for the different cases the effectiveness of the safety provisions for meeting the objective of practical elimination with a high level of confidence.

Nevertheless, the participants agreed that appropriate engineering provisions and guidance should be required to be implemented to mitigate the consequences of severe accident scenarios which might still occur due to unexpected further failures.

Related to the implementation of defence-in-depth, also specific aspects such as the safety classification or equipment qualification for equipment required for different plant states, their diversity and independence were addressed. The discussion showed that the issue is more on the application of the defence-in-depth concept rather than on harmonization of requirements and/or recommendations. Further clarification of current practices for new builds as well as nuclear power plants in operation was regarded useful.

Enhancement of defence-in-depth at operating nuclear power plants is usually investigated considering the feasibility of the installation of new systems and components for mitigating the consequences of accidents with core melting, and to improve the independence between the systems designed for different plant states. Regarding the mitigation of accidents with core melting, a majority of the safety improvements already implemented or planned aim at preventing dispersion of the molten fuel caused by the rupture of the reactor vessel at high pressure, and at ensuring the corium debris retention inside the reactor vessel by the implementation of an adequate ex vessel cooling system. In some Member States, an ex-vessel retention strategy is envisaged. Recognizing that full independence of the different levels of defence-in-depth is not practically achievable raises the question to what extent independence between the levels of defence is achievable. A comprehensive analysis to determine weaknesses in the implementation of defence-in-depth is largely regarded to be necessary.

The application of design principles is also linked to requirements developed internationally or by regulatory bodies. In this context, the IAEA safety standards, the new revision of the European Utility Requirements (EUR) and applicable regulations in some countries such as France, China, Finland or the UK were considered. A specific session was dedicated to the challenges in the implementation of design safety principles due to different regulatory frameworks.

High level nuclear safety objectives have reached a certain level of harmonization and are comparable between countries but some safety principles and aspects are not evaluated with the same criteria between countries having the same overall safety requirements. Therefore, the organization of exchanges between regulatory bodies and nuclear industry sector actors was recommended to get a better mutual understanding of the rational for their application. Efforts of harmonization between countries need to be carried on for the benefit of consistency between country regulatory requirements, the predictability of the licensing of a reactor design, and the cost to completion for new nuclear projects.

Licensing of Advanced Reactor Designs

In regards to meeting the objectives of the Vienna Declaration on Nuclear Safety a majority of the Contracting Parties had reported at the 7th Review Meeting on Convention on Nuclear safety that the objectives of the Declaration were included in the latest revision of their regulatory requirements.

Issues related to consideration of vendor country regulations, safety requirements relevant to passive safety features, legal implications of the generic design assessment process, approaches to respond to the Vienna Declaration, independent verification of a safety case, design extension conditions applied to the spent fuel pool were addressed.

The participants recognized that the safety objectives and the high level regulatory requirements reflected in the national regulations are often identical or similar, but the guidelines published by regulatory bodies may differ in the application or safety demonstration. For the industry, efforts of harmonization between countries are desirable for the benefit of consistency between country regulatory requirements, the predictability of the licensing of a reactor design, and the costs to completion for new nuclear projects.

Harmonization of technical specifications for advanced light water reactors is very important for all stakeholders. For example, the EUR includes 4500 requirements that cover many aspects such as safety, performance, and competitiveness, and can be used by the utilities for design assessment and technical reference in call for bids.

From all sides, the importance of harmonization of codes and standards was emphasized because it has impacts on the time of the project, costs, supplies of equipment and safety. There is a need to harmonize regulatory requirements (e.g., within the Multinational Design Evaluation Programme (MDEP)) and industrial standards and codes as well. The differences are being analysed and harmonisation aims at either convergence (same or similar requirements) or reconciliation (differences are accepted but justified), as designs are associated to codes and some vendors cannot always adapt to different codes.

Digital instrumentation and control (I&C) remains an important topic of discussion. It was concluded that there is a need for developing an internationally recognized method to evaluate common cause failures in digital I&C control and protection systems and support systems including emergency diesel generator using embedded digital devices that will be fundamentally structured around the defence-in-depth concept.

There are more than fifty SMR designs under various stages of development comprising a broad spectrum of reactor technologies; each having its own advantages and disadvantages. Selecting the appropriate SMR design depends on the intended application and timeline for deployment. Standardization and/or harmonization of safety standards and international collaboration are very desirable and most countries pursuing SMRs are already engaged internationally in this discussion. There is a strong interest in the industry in trying to harmonize safety and licensing requirements for SMRs because much of the plant will be built in a factory and standardization is important. According to the discussion, emergency planning zones (EPZ) for SMRs should be revisited and refined to account for advantages in SMR designs. Moreover, multi-modular issues need to be researched further to account for multiple failures or possible mass failures' effects on all units/modules.

The interests and statuses of the development of SMRs in Member States were discussed during the roundtable. In China, the national safety authority has issued a guide for an SMR review plan and started the preparation of design requirements; in the Russian Federation, three SMRs are being licensed. In other Member States, there is no real development work but there is a clear interest in international cooperation. Economics in a competitive market seem to be the most important hurtle, in a context where now-a-days renewable energies are highly subsidized. In addition, first-of-a-kind issues could be important regardless of the technology of the SMRs.

Safety Reinforcement of Existing Installations

The Vienna Declaration on Nuclear Safety (VDNS) stresses the importance of improving the nuclear safety of nuclear power plants in operation in order to prevent long term off-site contamination should an accident occur. It was recognized that implementing the VDNS will influence the nature of the provisions to be implemented and the priorities for their implementation. Additionally, the EU Nuclear Safety Directive and the reference levels established by the Western European Nuclear Regulators' Association (WENRA) must also be considered by the EU Members.

The practical implementation of the VDNS at existing nuclear power plants is still a challenge for the operating organisations and a clarification of the terminology is highly desirable.

Maintaining a high level of nuclear safety at nuclear power plants in operation throughout their entire lifetime is required by the national regulations, but meeting this requirement remains a challenge to operating organization resources beyond the obligation of performing periodic testing, inspection and maintenance. Considering Member States' practices different means to identify, design and implement safety improvements have been reported. The majority of the participants recognized that the stress tests or the safety re-assessments and the peer reviews required by the regulatory bodies and implemented in the light of the Fukushima Daiichi accident have largely identified safety improvements already implemented or that are in the process of being implemented. However, as the regulatory framework is not the same in all Member States, the self-assessments completed by Member States in the light of the Fukushima Daiichi accident have revealed differences in the objectives, priorities and implementation schedule for safety improvements due to different regulatory approaches. Other means of identifying safety improvements were discussed among them license renewal, authorizations for continued operation or the use of operating experience.

The presentations outlined a set of technology independent common areas where the operating organisations have planned or already implemented some modifications aimed at improving nuclear safety including:

- Reinforcement of the defence-in-depth concept by strengthening the independence among the systems and components designed to mitigate different plant states;
- Definition of strategies to mitigate the consequences of accidents with core melting and development of the associated severe accident management guidelines (SAMG);
- Introduction of safety features to cope with potential failures of the safety systems;
- Prevention and limitation of the effects of the external and internal hazards.

The discussions highlighted some difficulties in the application of the latest design safety requirements primarily established for new builds to operating nuclear power plants and recommended to share experiences with backfitting measures implemented at nuclear power plants in operation to meet the objective of the Vienna Declaration on Nuclear Safety to the extent practicable.

All participants agreed that sharing approaches, practices and design solutions at an international level as well as dissemination of research and development (R&D) achievements contribute to saving time and resources in the identification and implementation of the safety improvements. IAEA conferences and technical meetings, the World Association of Nuclear Operators (WANO) inspections and owner group events provide good opportunities for cooperation and sharing of good and best practices and lessons learned.

The effective implementation of severe accident management guidance remains a priority for operating reactors. Severe accident management has been strengthened following the Fukushima Daiichi nuclear power accident and most operating nuclear power plants have completed or have planned to complete the process of implementing these improvements. Questions remain regarding the proper usage and control of portable equipment which is used during accident management and the appropriate level of regulatory oversight.

Multiple failures are widely recognized as the major contributing factor to accidents with core melting. For nuclear power plants in operation a strategy widely applied to reinforce prevention of core melting relies on the installation of additional equipment in case of failure of the systems designed to respond to in such an event. Another issue discussed was that site hazards have to be considered on the basis of their causation and likelihood including a realistic set of combinations of natural hazards when natural causes for their combination exist.

The role of the off-site support services was discussed in the context of the accident management in conditions not initially considered by the design of the nuclear power plant and in the event of natural hazards of a magnitude higher than the one considered for the design of the structures and components. The participants discussed that the use of non-permanent equipment should be investigated with due account taken of the coping time available before unacceptable consequences occur.

SAFETY ASSESSMENT OF ADVANCED REACTOR DESIGN

SAFETY ASSESSMENT OF PASSIVE SAFETY SYSTEMS

Chairperson

M. COLBY
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QUALIFICATION OF THE SYSTEM CODE AC² (SUBMODULE ATHLET) FOR THE SAFETY ASSESSMENT OF PASSIVE RESIDUAL HEAT REMOVAL SYSTEMS

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Abstract

GRS has been developing the thermal hydraulics system code ATHLET (now being a submodule of the AC² code, which is a substantial part of the GRS nuclear simulation chain) over many years. Since the code is widely used in nuclear supervisory and licensing procedures, its simulation capabilities have to represent the current state of science and technology. One notable innovative design safety feature of advanced light water reactors are passive systems. Unlike active systems which work at defined operating points, passive systems operate under conditions which are set on their own and may vary during the course of a transient dependent on the boundary conditions (e.g. pressure and temperature). Moreover, the driving forces of passive systems are usually relatively small compared to those of active systems so that uncertainties in the parameters can have a larger impact. Thus, the correct prediction of the behavior of these systems still proves to be a challenge for the AC² code. The present paper is focused on the recent efforts that have been made to qualify AC² for the simulation of slightly inclined horizontal heat exchangers like the emergency condensers used in the KERENA and CAREM reactor.

1. INTRODUCTION

In order to simulate all relevant phenomena within a nuclear power plant, GRS uses various self-developed and validated methods and computer codes. These codes are forming the so called nuclear simulation chain covering phenomena of neutron kinetics, thermal hydraulics within the cooling circuit and containment as well as structural mechanics [1]. Being part of this nuclear simulation chain, the code AC² covers the simulation of all operational states, incidents, accidents and severe accidents in a nuclear power plant with the GRS modules ATHLET, ATHLET-CD, COCOSYS and ATLAS. The submodule ATHLET (Analysis of Thermal-hydraulics of Leaks and Transients) has been developed by GRS for many years. It is used for best estimate analyses of normal operation, incidents and accidents of the existing nuclear power plants. ATHLET has a substantial validation basis (e.g. LOCAs, transients, etc.) comprising OECD/CSNI validation matrices, start-up and operational transients of nuclear power plants, international standard problems (ISP) and benchmarks [2]. The currently latest release version of ATHLET within AC² is ATHLET 3.1A.

The characteristics of passive systems, such as relatively weak and continuously changing driving forces relying on basic physical laws as well as independent onsets of such systems, pose a challenge for the correct simulation with system codes. The present paper addresses the efforts that have recently been undertaken by GRS to qualify ATHLET for the simulation of slightly inclined horizontal heat exchangers like the emergency condensers used in the KERENA and CAREM reactor.

It should be noted that the paper concentrates only on pure steam condensation inside horizontal tubes; the presence of non-condensable gases is not treated here. The consideration of this very important, the heat transfer performance limiting factor in ATHLET will be investigated in further studies. Another factor — the effect of condenser tube inclination – is not discussed here because the heat transfer module of ATHLET treats all inclination angles smaller than approx. 11.5° as fully horizontal. In comparison, the inclination angle of the discussed NOKO test facility is 1.6° in the upper tube rows and 3.2° in the lower rows.

2. SIMULATIONS AND DEVELOPMENTS OF THE RECENT PAST

This chapter provides a very brief summary of the ATHLET developments and simulations of the recent past with regards to emergency condensers. An extensive description of these topics was presented in [3].

2.1. NOKO facility description and simulation results

Starting point for the recent ATHLET developments related to horizontal passive residual heat exchangers were post-test simulations of experiments conducted at the NOKO test facility (Notkondensator test facility). The – now dismantled – NOKO test facility was constructed at the Forschungszentrum Jülich GmbH (KFA, now FZJ) for experimental investigations of the effectiveness of the emergency condenser of the BWR600/1000, the predecessor of the KERENA reactor. The facility had an operating pressure of 7.2 MPa and a maximum power of 4 MW for steam production. It comprised an emergency condenser consisting of eight tubes with original geometries and materials of the BWR600/1000 emergency condenser (EC); four of these eight tubes could be isolated by plugs, thus yielding a 1:13, resp. 1:26 scaling of the EC with respect to the original component.

A thorough facility description can be found in [4]. An ATHLET nodalization of the main components of the facility is shown in Fig. 1. The pressure vessel represents the reactor pressure vessel and has a defined liquid level which can be regulated by the condensate drainage. Saturated steam is injected to the pressure vessel by a steam supply. The condenser tube bundle is connected with the pressure vessel by a feed line and a return line. It follows from this configuration that the liquid level in the pressure vessel has an influence on the liquid level inside the condenser tubes. As for the secondary side, the condenser tube bundle is located in the condenser vessel where it is completely immerged in water. While the steam and water on the primary side are saturated (except for the condensate which may be subcooled), the water on the secondary side is either subcooled or saturated, dependent on the performed experiment.

The experiments conducted at the NOKO facility are described in detail in [5]. Their objective was to determine the heat transfer power of the EC under quasi-stationary conditions depending on various parameters such as primary side pressure, secondary side temperature and pressure, or liquid level in the pressure vessel. Many of these experiments have been simulated with ATHLET. For the sake of brevity, however, the paper concentrates on the simulation of a class of experiments in which the liquid level in the pressure vessel was low enough to ensure a complete primary side exposure of the condenser tubes and in which the secondary side water was in a saturated state. The conclusions that can be drawn from the comparison of these selected simulation results with the experimental data are representative for all simulated experiments.

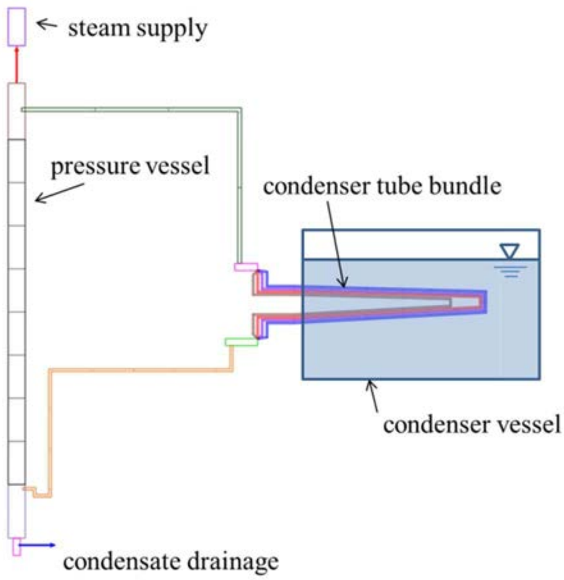

FIG. 1. ATHLET nodalization of the main components of the NOKO test facility (secondary side schematic).

The first simulations of the NOKO facility were performed with ATHLET 3.0A. The nodalization scheme for the calculations was similar to the one shown in Fig. 1 with a modelling of the secondary side that allowed for circulation processes to occur. In order to make the results comparable, this nodalization scheme was used for all simulations with all code versions presented in this paper.

As the simulation results deviated substantially from the experimental data (ATHLET 3.0A underestimated the EC power by a factor of approx. 2), condensation and boiling heat transfer models were modified and incorporated into the new code version ATHLET 3.1A.

With these code developments, the simulation results improved significantly. However, for experiments with larger temperature differences between the primary and the secondary side – and thus for larger EC powers, the agreement between the experiments and the calculations is not yet satisfying as can be seen below in Fig. 2.

2.2. Other emergency condenser related simulations

Beside the NOKO simulations, GRS conducted further simulations related to passive emergency condensers, notably post-test calculations of both stationary and transient experiments carried out at the INKA (Integral Teststand Karlstein) test facility of AREVA in Karlstein, Germany. The INKA facility is a representation of the KERENA reactor with all its volumes and passive safety systems, like the EC, the building condenser or the passive pressure pulse transmitter. The scaling in height is 1:1 while the volumetric scaling is about 1:24. The passive safety systems are in full scale, but their number is scaled down to 1:4 [6]. The concluding statement one can make related to these calculations is that ATHLET systematically under-predicted the EC power by approx. 30%.

Additional temporary modifications of the ATHLET code gave rise to minor improvements of the INKA simulations, but a significant underestimation of the EC power still remained.

3. ONGOING WORK

Since the simulation results of EC behaviour are not yet satisfying, further investigations have to be undertaken. For this purpose, GRS is currently participating in two joint research projects in the frames of which experimental work together with code development is envisaged:

The first project, EASY (Integral experimental and analytical investigations regarding the controllability of design accidents with passive systems), is dedicated to the development of a coupled program system for the simulation of passive systems as well as their interactions, consisting of the AC² modules ATHLET and COCOSYS, and the validation of this program system by both single component tests and integral tests of design basis accidents, conducted at the abovementioned INKA facility. The objective of the project is the creation of a tool for verification and assessment of newly built nuclear power plants.

The second project, PANAS (Passive Nachzerfallswärme-Abfuhrsysteme), is about the investigation of passive residual heat removal systems, with a focus on experimental analysis, model development and validation of both system codes and CFD codes. The GRS subtask is the development and validation of heat transfer models for evaporation and condensation at horizontal heat exchangers for ATHLET.

Both projects are complementary in that PANAS concentrates on code development and validation by single effect tests while EASY is focused on the simulation of integral experiments. Hence, insights gained about particular physical phenomena within PANAS can be beneficially used for the integral calculations within the frame of EASY.

The following subchapters describe the latest, within the project PANAS, implemented condensation models in ATHLET as well as related simulation results of the aforementioned NOKO experiments.

3.1. New heat transfer models

In order to predict the condensation heat transfer of an emergency condenser accurately, the knowledge about the local flow pattern inside the condenser tubes is essential. So far, ATHLET lacks this information – its condensation heat transfer models only roughly distinguish between two basic flow regimes. A literature review within the project PANAS came to the conclusion that the flow pattern map-based heat transfer models of Thome et al. and KONWAR are promising candidates for improving the capability of ATHLET to predict heat transfer coefficients for condensation in horizontal tubes.

3.1.1. KONWAR

KONWAR (Kondensation in waagerechten Rohren, condensation in horizontal tubes) is a code module for the determination of heat transfer coefficients during the condensation in horizontal tubes. It is based on a modified flow regime map of Tandon et al. [7] and includes several empirical and semi-empirical correlations for the determination of the HTCs.

KONWAR was developed in the 1990s as an extension module of ATHLET version 1.1 and it was validated against the NOKO experiments. The validation results showed a very good agreement between simulation and experimental data. A thorough model description as well as the validation results can be found in [5].

3.1.2. Thome et al.

The heat transfer model of Thome et al. [8] is based on the condensation flow pattern map of El Hajal et al. [9]. The facts that the model is based on mechanistic reasoning and that it can be extended for the prediction of heat transfer coefficients for condensation in the presence of non-condensable gases make it an appropriate choice for the usage in ATHLET.

3.2. NOKO simulations applying the new heat transfer models

To prove the operability of the coupled heat transfer packages together with ATHLET and to get a first impression of the predicted heat transfer, the simulations of the NOKO experiments described in chapter 2 were repeated with coupled versions of both ATHLET 3.1A/KONWAR and ATHLET 3.1A/Thome. The results of some of these simulations can be seen in Fig. 2.

Compared to ATHLET 3.1A alone, the coupled versions of ATHLET with KONWAR and Thome yield slightly higher heat transfer coefficients and thus predict a better performance of the EC. While all models predict the EC power accurately for experiments with small differences between primary and secondary side saturation temperatures, the EC powers for intermediate temperature differences are predicted best by ATHLET/Thome, and the experiments with the largest ΔT are approximated closest by ATHLET/KONWAR. This rough description applies not only for the shown simulation cases, but also for the other calculated NOKO experiments.

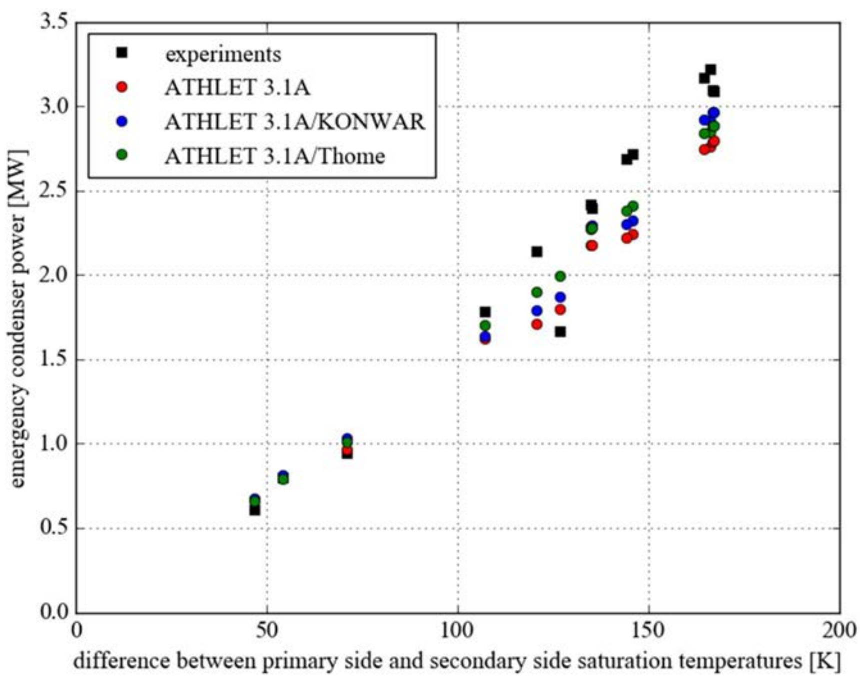

FIG. 2. Comparison of experimental data and simulation results, using different versions of ATHLET.

4. CONCLUSIONS AND OUTLOOK

It can be concluded that the code developments accompanying the version upgrade from ATHLET 3.0A to 3.1A led to significant improvements of the calculation of both condensation and boiling heat transfer at the condenser pipes. The newer developments which comprise the introduction of more detailed flow pattern map-based heat transfer models lead to slightly improved simulation results of the NOKO experiments; however, these results are still not yet satisfying and – for the test cases with large condenser powers – differ clearly from the experimental data.

Although the NOKO test facility was a single component test stand, it was not a single effect test stand because several physical processes acted in parallel: condensation inside the tubes, boiling and circulation outside the tube bundle and heat conduction through the tube walls (the conductivity of which remains unclear). For this reason, within the project PANAS, GRS engages in ATHLET simulations of the COSMEA facility at Helmholtz-Zentrum Dresden-Rossendorf (HZDR) [10]. The COSMEA facility is a single effect stand designed to study condensation heat transfer and flow structure. The test section mainly consists of a single, slightly inclined tube which is cooled by forced convection. The instrumentation includes among other things thermocouples for a detailed measurement of the temperature distribution and an X-ray tomography system for the investigation of local flow patterns. It is expected that the simulations of the COSMEA single effect experiments will help to localize the major deficits of ATHLET's heat transfer models with regard to emergency condenser simulations.

ACKNOWLEDGEMENTS

The project EASY (reference number RS1535A) is funded by the German Federal Ministry of Economic Affairs and Energy (BMWi).

The project PANAS (reference number 02NUK041), where GRS is a subcontractor of Technische Universität Dresden, Chair of Hydrogen and Nuclear Energy, is funded by the German Federal Ministry of Education and Research (BMBF).

The work presented under the headline "Simulations and developments of the recent past" was funded by the German Federal Ministry of Economic Affairs and Energy (BMWi) within the projects RS1507 and RS1519.

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EFFECT OF NON-CONDENSABLE GAS ON THE PERFORMANCE OF PASSIVE CONTAINMENT COOLING SYSTEM IN VVER-1200 DESIGN

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Abstract

As a result of catastrophic events on the nuclear power plant "Fukushima" in Japan, there are a lot of concerns about the safety issues of evolutionary NPP design which rely on passive safety systems to provide the ultimate heat sink and deal with design basis accident (DBA) and beyond design basis accident (BDBA). However, the passive safety systems which use natural forces in operation is lack of practical operating experience and their performance reliability depends on the environmental, physical, nuclear, or chemical phenomena, to a greater extent than active systems. The passive containment cooling system's performance might be deteriorated by non-condensable gases that come from the containment and from the gases produced by cladding/steam interaction during a severe accident. These non-condensable gases degrade the heat transfer capabilities of the condensers in the passive containment cooling systems since they provide a heat transfer resistance to the condensation process. The paper presents the cooling capacity analysis of PRHR/C in VVER-1200 design using RELAP5 Mod3.2 to confirm the performance of conceptual design as well as to assess the efficiency of steam condensation outside heat exchangers within the presence of non-condensable gas. Comparison results are also presented and analyzed.

1. INTRODUCTION

Condensation heat transfer is a primary concern in passive systems used in advanced plants to increase the inherent safety such as the Passive Containment Cooling System (PCCS) of AP1000, VVER-1200 design, the Isolation Condensation System (ICS) of ESBWR design, and the Passive Residual Heat Removal System in AP1000, APR1400, VVER-1200. The principle working of PCCS in VVER-1200 design is based on using an air heat heat exchanger (HEX) which is connected with a pool located on the top of the contaiment. Natural circulation and heat removal capability are generated when air HEX receives heat from the containment: this occurs through liquid heating and straticication that produces a difference between densities in the rising and decending leg of the pool-type HEX [1]. Therefore, steam vented in the containment following DBA or BDBA will condense on the containment condenser tube surfaces to provide pressure suppression. In these systems, condensation heat transfer in the outer surface of vertical tubes is the main heat transfer mechanism, and non-condensable gases can be present during the accident scenarios. When condensation occurs at the interface of a liquid film on the wall of a vertical tube, a non-condensable gas will accumulate and form a non-condensable gas layer. This increases the non-condensable gas concentration at the interface between the liquid film and gas, which in turn reduces the condensation heat transfer rate. A lower condensation heat transfer rate causes the performance of the heat exchanger to deteriorate, which affects the heat removal capacity in accident conditions and impacts plant safety.

Many experimental/numerical studies have been performed to examine condensation heat transfer efficiency in the presence of a non-condensable gas inside a vertical tube [2-9]. Accordingly, these results has been only used to support the design of a passive system with steam condensation in the inner surface of vertical tubes. It should be also noted that these experimental test have very small scale in purpose of phenomena investigation or correllation development. To obtain appropriately experimental validation of Passive Containment Cooling System in VVER-1200 V491 Design with steam condensation in the outer surface of vertical tubes, OKBM Afrikantov has developed a large-scale test facility and performed experimental investigations to support code validation at full-scale analysis of safety system as well as to prove the effectiveness and serviceability of the cooling loop for removing heat from the protective envelope [10]. The present paper provide some calculation results of above experimental test facility using RELAP5 Mod 3.2 Code to investigate the performance of cooling loop with an elevated concentration of non-condensable gases and with pure steam. The calculation resutls are also compared with experimental data to show the predictive capability of condensation heat transfer models implemented in RELAP5 MOD3.2 code.

2. DESCRIPTION OF EXPERIMENTAL TEST FOR VALIDATION OF THE COOLING LOOP

The construction of the full-scale cooling loop design which removes heat from the protective envelope were completed in 2008 at OKBM Afrikantov (Russia) and this test facility has been selected as benchmark data of this study. The arrangement of the cooling loop corresponds to a real cooling loop of PCCS in VVER-1200 V491 design. The design information of full-scale cooling loop and those of the protective envelope tank (which model the containment) can be obtained in details in [10], [11], respectively. The operation of test facility is illustrated in Fig. 1 and briefly introduced as follows: Steam generated from electrical steam generator flows into separators to lower its moistrure content, flows along steam lines into the bottom part of the protective envelope, and then flows to the outer surface of heat exchanger tubes. The condensate water is collected into a collector which is positioned beneath the tank. In the cooling loop, water flows from the evaporator tank in to the heat exchanger (condenser) and receives heat to increase its temperature, partially evaporates, and then flows into steam receiver in which steam is separated and discharged along a pipeline into the atmosphere.

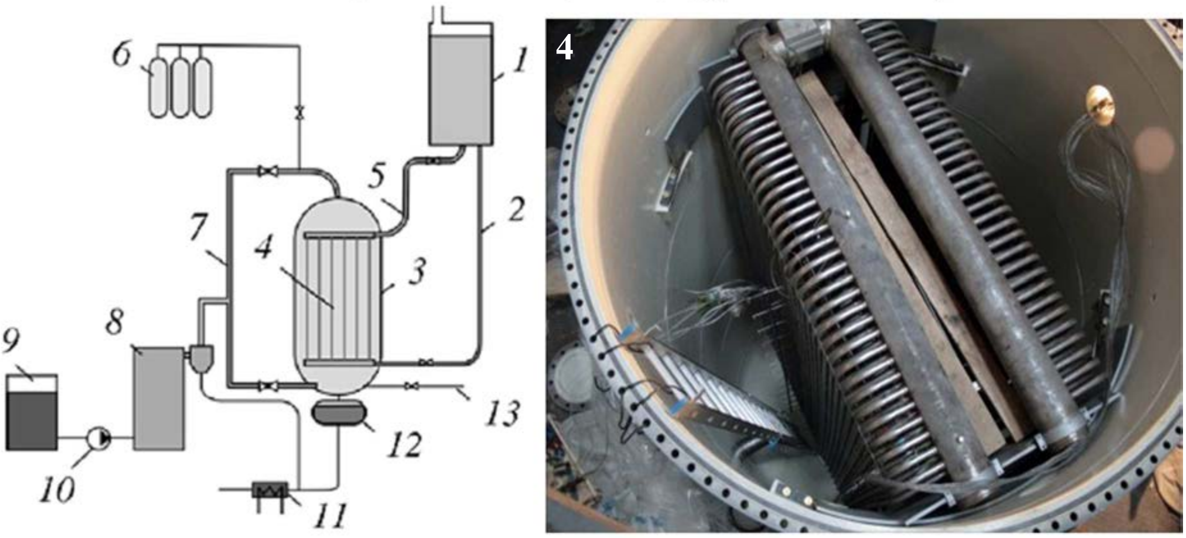

FIG. 1. Schematic diagram of a passive system for removing heat from a protective envelope:
1) evaporator tank with a steam receiver; 2) feed line; 3) tank modeling the protective envelope; 4) exchanger-condenser; 5) discharge pipeline; 6) tanks with air; 7) pipelines feeding warming steam into the modeling tank; 8) electricity generator with a steam separator; 9) tank with a salt solution; 10) pump; 11) secondary cooler; 12) condensate collector; 13) air blow-off pipe [10, 11].

3. CALCULATION SETUP FOR OKBM AFRIKANTOV TEST FACILITY

Fig. 2 shows the RELAP5/MOD3.2 code nodalization scheme for the OKBM experiments. The RELAP5/MOD3.2 nodalization used for this simulation contained two loops as described in Fig .2: The cooling loop and the protective envelope loop. Main components can be seen from this figure: 100 (Atmospheric, cooling loop), 102 & 104 (evaporator tank, cooling loop), 119-124 (Heat Exchanger Tubes, cooling loop), 202 (condensate collector, protective envelope loop), 204 (containment modelling tank, protective envelope loop), 215 (electrical steam generator, protective envelope loop). Experimental test cases with different gas (air) content in the modelling tank and total power of the electricity generators ranging from 0.5 to 1.8 MW (controlled by electrical heater inside steam generator) are selected for calculations using RELAP5/MOD 3.2 in the present study. The gas content is setup by partial pressure in the modeling tank prior to discharging steam from steam generator to the modeling tank with following cases: 0, 150, 200, 250 and 300 kPa. For each test case, steam is discharged into the modelling tank with total power of the electricity generator 1.8 MW. Then the power of the electricity generator is lowered to 1.5, 1.0, and 0.5 MW and the parameters is allowed to stablized over a time of 1 hour at each power level [10].

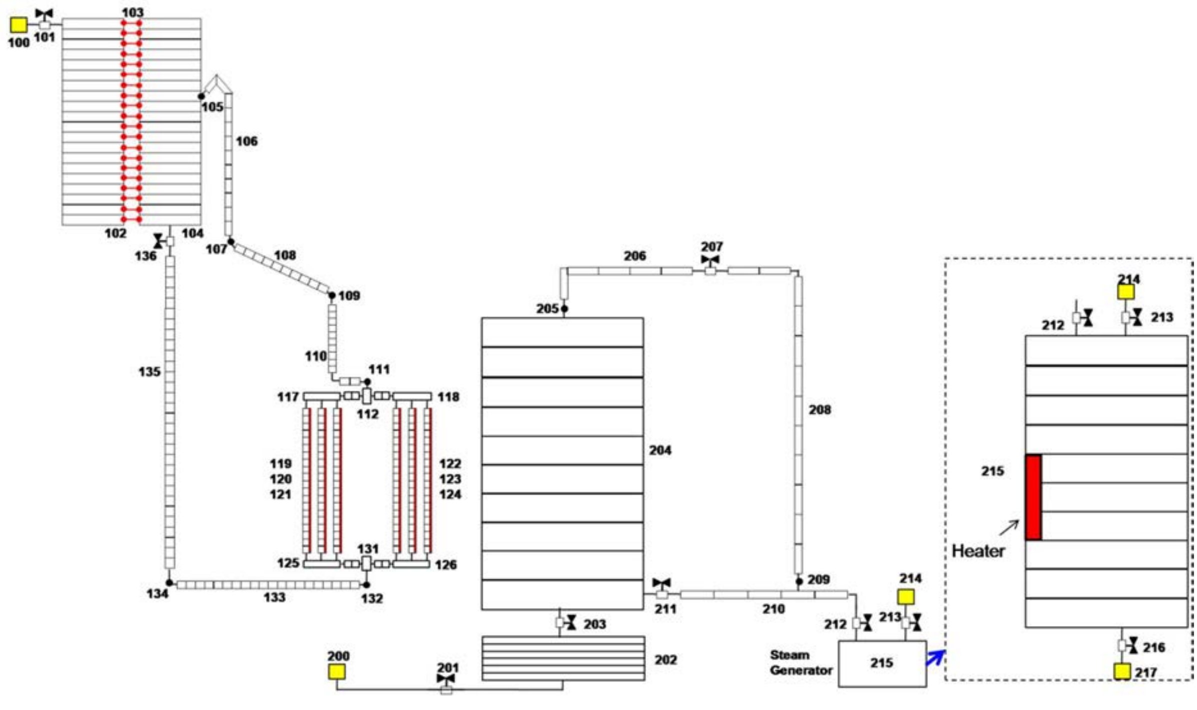

FIG. 2. Nodalization Scheme for OKBM Afrikantov Test Facility.

4. RESUTLS AND DISCUSSION

Comparison results of the coolant temperature at the entrance (CV 135) and the exit (CV 110) of the cooling loop between RELAP5/MOD3.2 calculations and experimental data are shown in Fig. 3 (a). In this case, the initial air pressure in the tank modelling the protective envelope prior to heating was equal to atmospheric pressure (100 kPa). In can be seen that the calculation results have quite similiar trends with the experimental data. The deviations may be accounted for differences in the initial condition of water temperature in the evaporator tank. However, the trend of coolant mass flow rate in the cooling loop is quite different. The experimental evidence of periodic oscillations of the cooling flow rate due to the onset of steam formation in the ascendending section of the discharge pipeline when coolant temperature reach 343 ^{o}K was not captured in the calculation results. It is reported that the oscillations of the flow rate along the cooling loop was observed only in a particular of regime with the transition of the total power of the electrical steam generator to 0.5 MW. As shown in Fig. 3 (b), this behavior was repeated at all power level (1.8 MW, 1.5 MW, 1.0 MW, 0.5 MW). The oscillations of the flow rate are related with a decrease of the average flow rate and an increase of the transit time of a volume of coolant through the heat exchanger tube and the discharge pipeline into the evaporator tank, which results in a delay of the back effect of the coolant temperature changes [10]. Most important parameter here is the pressure inside the protective envelope which indicates how efficient is the steam condensation. Fig. 3 (c) shown quite similar trends but very big deviations between RELAP5/MOD3.2 calculations and experimental data of the pressure inside the protective envelope. It can be explained by the weakness of condensation heat transfer models implemented in the code as already raised by some authors [2-4, 9]. They have suggested that the effect of the interfacial shear stress was not sufficiently considered in the correlations using the Reynolds number.

Figs. 3 (d), (e), and (f) clearly shown the effect of non-condensable gas (air) on the efficiency of steam condensation when the intial air pressure inside the protective tank is increased. The presence of even a small quantity of non-condensable gas in the condensing vapor has a profound influence on the resistance to heat transfer in the region of the liquid-vapor interface. The non-condensable gas carried with the vapor towards the interface where it accumulates. The partial pressure of gas at the interface increases above that in the bulk of the mixture, producing a driving force for gas diffusion away from the surface.

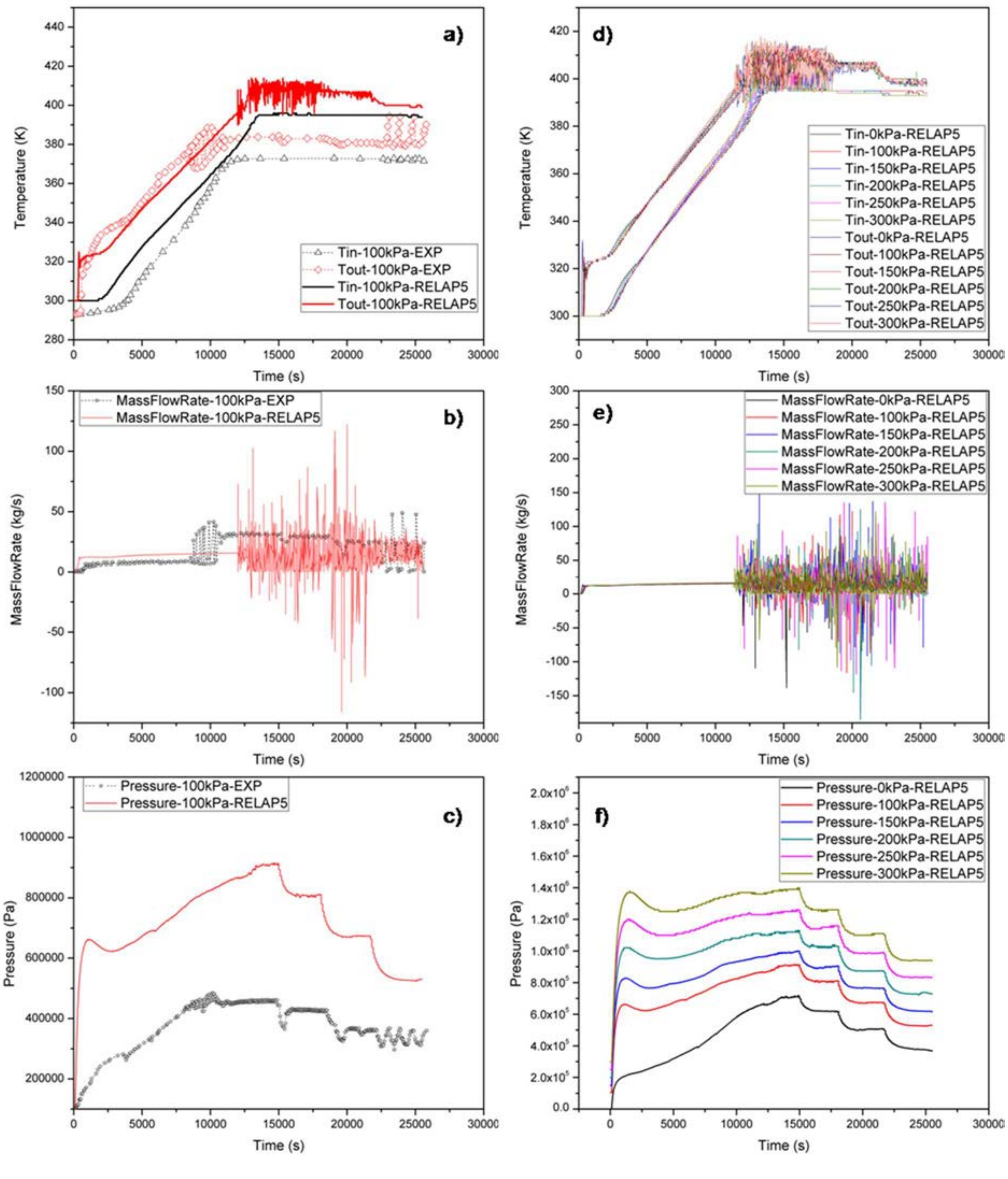

FIG. 3. Typical calculation results of Temperature, Mass flow rate in the cooling loop and Pressure in the protective envelope with RELAP5 MOD3.2.

5. CONCLUSIONS

The capability of the RELAP5/MOD3.2 code to investigate the performance of cooling loop of PCCS in VVER-1200 V491 design with an elevated concentration of non-condensable gases was assessed in this study. Overall, the RELAP5/MOD3.2 captured quite well the effect of the intial air pressure inside the containment to the performance of the PCCS. However, the current RELAP5/MOD3.2 code strongly underestimated the condensation heat transfer coefficient which leads to a strong overestimation of the pressure level.

ACKNOWLEDGEMENTS

This work was supported by the Ministry of Science and Technology (MOST) R&D Project: "Construction of Thermal-hydraulic Test and Investigation of Hydrodynamic Phenomena of Two-phase Flows", grant code DTCB.17/16/VKHKTHN.

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THE ROLE OF PASSIVE SYSTEMS IN ENHANCING SAFETY AND PREVENTING ACCIDENTS IN ADVANCED REACTORS

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Abstract

Most of the new reactor designs are introducing inherent and passive safety features that do not depend upon external source of power or human actions for their successful performance. In view of this, a desirable goal for the safety characteristics of an innovative reactor is that its primary defence against any serious accidents is achieved through its design features preventing the occurrence of such accidents without depending either on the operator's action or, the active systems. Passive systems are credited a higher reliability with compared to active systems, because of greater availability due to lower probability of hardware failure or human error.

1. INTRODUCTION

Passive system is the system whose functioning does not depend on an external input or human action such as actuation, mechanical movement or supply of power. As a result, passive safety systems are being considered for numerous reactor concepts (including in Generation III and III+ concepts) and are expected to find applications in the Generation-IV reactor concepts [1, 2]. Passive systems are credited a higher reliability with respect to active ones, because of a smaller unavailability due to a minimize hardware failure and human error. The use of passive safety systems such as accumulators, condensation and evaporative heat exchangers, and gravity driven safety injection systems reduces the costs associated with the installation, maintenance and operation of active safety systems that require multiple pumps with independent and redundant electric power supplies.

In this paper the categorization, types and nature of passive systems are discussed. Typical deployment of passive systems among several advanced reactor are shown. The impact of introduction of passive systems on advanced Reactors are analyzed for advanced pressurized and boiling water reactors for two measures; the core damage frequency (CDF) and the large early release damage frequency (LERF) and compared with typical reactors. The deterministic and probabilistic analyses for passive systems are discussed as well as the challenges facing passive systems.

2. PASSIVE SYSTEMS CLASSIFICATIONS

Passive systems are composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation. Passive systems can be categorized into four classes A, B, C and D according to the degree of inclusion of passive action in the system. Actions required to operate passive or active system may include: input signal, external power source, moving mechanical parts and moving working fluids [1, 3]:

Class A: this category is characterized by no input signal, no external power source, no moving mechanical part and no moving fluids. Examples are: physical barriers against the release of fission products, such as nuclear fuel cladding and pressure boundary systems hardened building structures for the protection of a plant against seismic.

Class B: this category is characterized by no input signal, no external power source, no moving mechanical part but with moving fluids. Examples are: reactor shutdown/emergency cooling systems based on injection of borated water produced by the disturbance of a hydrostatic equilibrium between the pressure boundary and an external water pool; containment cooling systems based on natural circulation of air flowing around the containment walls, with intake and exhaust through a stack or in tubes covering the inner walls of silos of underground reactors;

Class C: this category is characterized by no signal inputs, no external power sources, with moving mechanical parts, whether or not moving working fluids are also present. Examples are: accumulators and or storage tanks and discharge lines equipped with check valves; overpressure protection and/or emergency cooling devices of pressure boundary systems based on fluid release through relief valves.

Class D: This category is characterized by signal inputs of ‘intelligence’ to initiate the passive process, energy to initiate the process must be from stored sources such as batteries or elevated fluids, active components are limited to controls, instrumentation and valves to initiate the passive system manual initiation is excluded. Examples are emergency core cooling and injection systems based on gravity that are initiated by battery-powered electric or electro-pneumatic valves, Core makeup tank, Elevated gravity drain tank, passive cooled steam generator natural circulation, passive residual heat removal heat exchangers and isolation condensers.

3. PASSIVE SYSTEMS IN TYPICAL NUCLEAR POWER PLANTS

Passive systems and components are included in all advanced Nuclear power plants of GEN-III. Typical passive cooling systems can be demonstrated in several reactors such as ABWR-II, APWR+, AP-1000, WWER-1000, and ESBWR.

3.1. Advanced Pressurized Water Reactors (APWR+)

The APWR+ is a four loop type PWR with 1750 MW(e) output, which is being developed as the successor of APWR and conventional PWRs, aiming at more enhancements in economy, safety, reliability, reduction of the operators’ workload, and harmony with the environment as shown in Fig. 1. APWR+ employs the following concepts for its safety system including passive features [2]:

- Passive core cooling system using steam generator;
- Advanced Accumulators;
- Advanced Boric Acid Injection Tank.

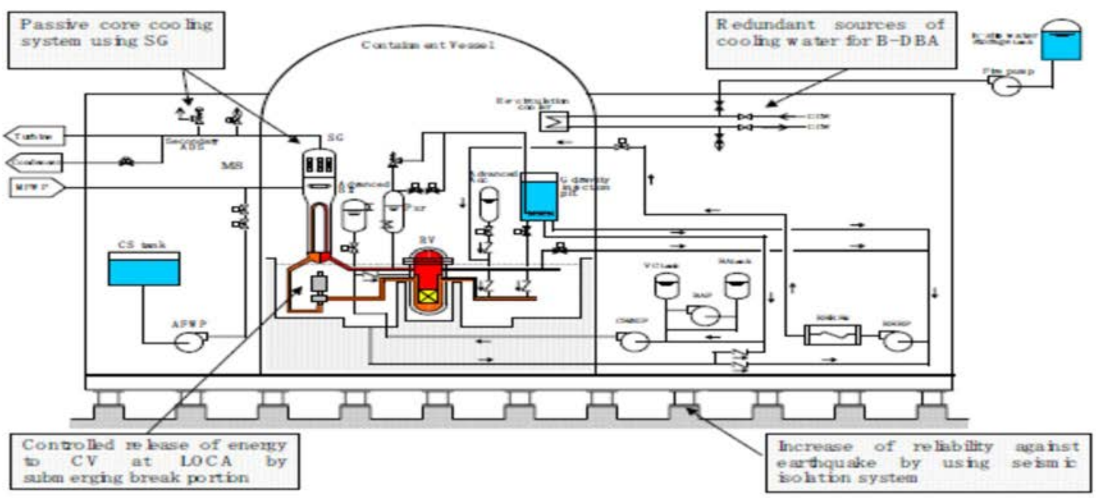

FIG. 1. Passive systems in APWR+.

3.2. AP -1000 Reactor (Advanced Passive)

The AP600 and AP1000 are pressurized light water reactors designed by the Westinghouse Electric Corporation to produce 600 MW and 1100 MW of electric power, respectively [1, 4]. Both designs employ passive safety systems that rely on gravity, compressed gas, natural circulation, and evaporation to provide for long term cooling in the event of an accident. The reactor employs the following features as shown in Fig. 2:

- An in-containment refueling water storage tank (IRWST);
- A passive residual heat removal (PRHR) system;
- Two core make-up tanks (CMTs);
- A four-stage automatic depressurization system (ADS);

— Two accumulator tanks (ACC);
— A lower containment sump (CS);
— Passive containment cooling system (PCS).

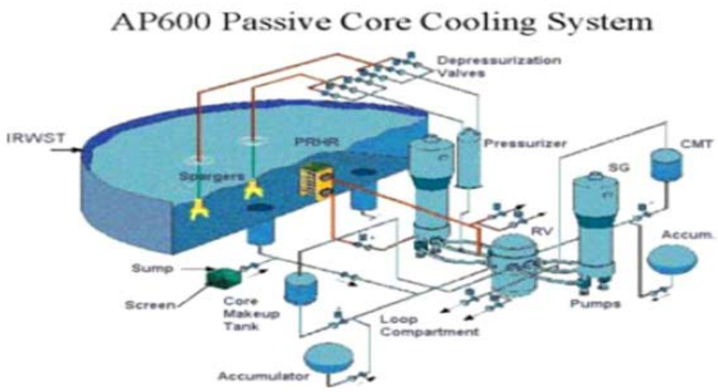

FIG. 2. Passive Safety Systems used in AP 1000 Designs.

3.3. VVER -1000 Reactor

WWER employ the following passive features as shown in Fig. 3 [1, 2]:

— Passive quick boron supply system;
— Passive subsystem for reactor flooding HA-1 (hydro accumulators of first stage);
— Passive subsystem for reactor flooding HA-2 (hydro accumulators of second stage);
— Passive residual heat removal system via steam generator (PHRS);
— Passive core catcher.

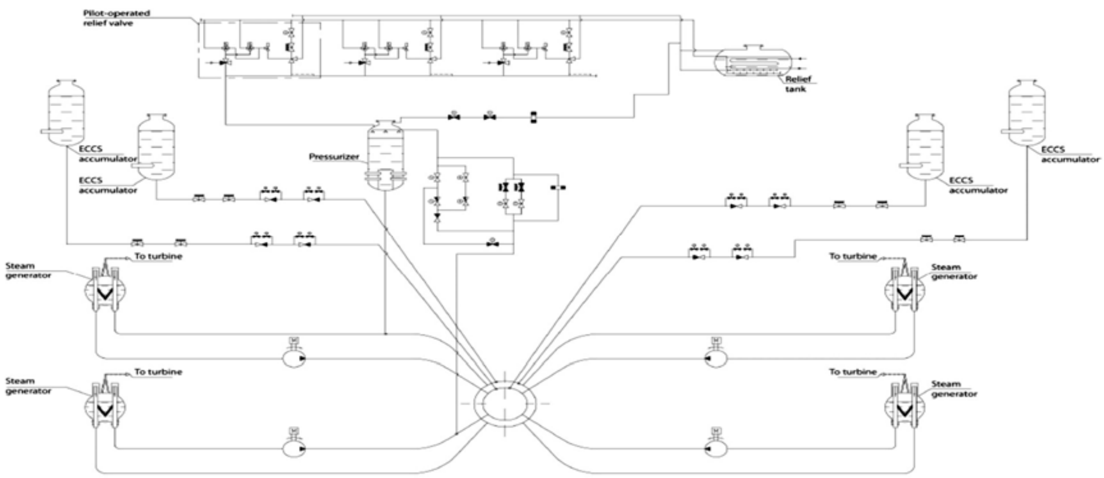

FIG. 3. Passive Safety systems in WWER-1000 Reactor.

3.4. Economic Simplified Boiling Water Reactor (ESBWR) 4500 MWth

ESBWR employs the following safety features in the design [2]:

— Gravity driven cooling system (GDCS);
— Automatic depressurization system (ADS), which consists of the depressurization valve (DPV) and safety relief valve (SRV);
— Isolation condenser system (ICS);
— Standby liquid control system (SLCS);
— Passive containment cooling system (PCCS); and
— Suppression pool (SP).

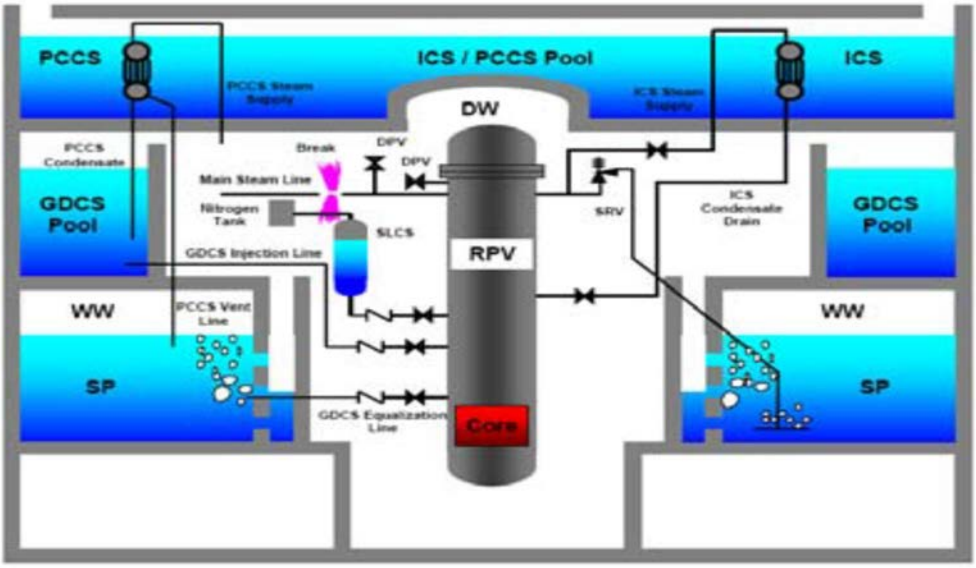

FIG. 4. Passive Safety systems in ESBWR.

4. ROLE OF PASSIVE SYSTEMS TO STRENGTHEN SAFETY AND MITIGATE ACCIDENTS

Passive systems do not require human intervention or input signal to start, so *they* eliminate or minimize both errors due to human intervention and failure of both on and off site power supply, Accident frequency (failure probability per reactor per year) is reduced and safety is enhanced. In PSA reactor accidents can be measured by Core Damage Frequency (CDF) and large early release Frequency (LERF). So, Safety goals for advanced reactors are to reduce CDF and LERF as compared to the exiting reactors and reduce release of radioactive material to the environment. Table 1 illustrates CDF and LERE for small and medium reactor. Table 2 illustrates CDF and LERE for some advanced reactors as compared to existing NPP. From the tables, we see that inclusion of Passive systems reduce accidents (CDF and LERE) by two fold.

TABLE 1. CORE DAMAGE FREQUENCY AND LARGE EARLY RELEASE ACCIDENT FOR SMR [1, 2]

Reactor type	SMART	IRIS
CDF/reactor .year	8.56×10^{-7}	2×10^{-8}
LERE/reactor .year	$< 10^{-8}$	6×10^{-10}

TABLE 2. CORE DAMAGE FREQUENCY AND LARGE EARLY RELEASE ACCIDENT FOR LARGE REACTORS [1, 2]

Reactor type	Existing reactors	ABWR (1700 MWe)	APWR+ (1750 MWe)	AP-1000 (1000 MWe)	VVER-1000 (1000 MWe)	ESBWR (1500 MWe)
Core Damage Frequency /reactor .year	10^{-5}	1.6×10^{-7}	$<10^{-7}$	5×10^{-7}	$< 10^{-5}$	$\sim10^{-8}$
Large Early Release Frequency/reactor .year	10^{-6}	$\sim10^{-8}$	$<10^{-8}$	5×10^{-8}	$< 10^{-7}$	$<10^{-8}$

5. DETERMINISTIC AND PROBABILISTIC ANALYSIS

Deterministic analysis for passive systems model system design and phenomena which occurs during operation, focus on accident types, consequences and releases without considering the probabilities of different event sequences. Thermal Hydraulic computer codes are usually used for deterministic analysis of passive systems such as RELAP and ATHLET computer codes [5].

Probabilistic analysis (PSA) for passive systems evaluate the failure rate (or frequency per unit time) for the components and systems, and the probability for certain accident sequence to occur (such as core damage Frequency). PSA constructs both event tree and fault tree for certain accident scenarios. Three levels of PSA are generally recognized. Level 1 comprises the assessment of plant failures leading to determination of the frequency of core damage. Level 2 includes the assessment of containment response, leading, together with Level 1 results, to the determination of frequencies of failure of the containment and release to the environment of a given percentage of the reactor core's inventory of radio nuclides. Level 3 includes the assessment of off-site consequences, leading, together with the results of Level 2 analysis, to estimates of public risks [6, 7, and 8].

6. CHALALLENGES FACE PASSIVE SYSTEMS

The following challenges and problems still face passive systems [4, 5, and 6]:

- Passive systems have little operating experience and their driving force is small, which can be changed even with small disturbance or change in operating parameters.
- Physical behavior of passive systems should be studied carefully especially in case of reactor transients.
- Aging of passive systems must be considered for long plant life.
- Flow instability which include density wave, flow pattern transition instability should be analyzed carefully.
- Phenomena such as Thermal stratification in large pools and effect of non-condensable gases on condensation should be studied.

7. CONCLUSION

- The reliability of the passive systems is high because it can continue to work in severe conditions such as loss of electricity and station blackout.
- Passive systems strength improves safety of the reactor and it should be used in combination with active systems to prevent accidents.

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INVESTIGATION ON CORIUM COOLING

Chairperson

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RECENT PROGRESS IN PHENOMENOLOGY AND TECHNOLOGIES RELEVANT TO IN-VESSEL MELT RETENTION AND EX-VESSEL CORIUM COOLING

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Abstract

The IAEA recently held the Technical Meeting on "Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling". It provided international experts with a platform for detailed presentations and technical discussions on recent progress in R&D activities on in-vessel melt retention and ex-vessel corium cooling during severe accidents in water cooled reactors.

This paper summarizes the major outcomes from the Technical Meeting focusing on recent progress and current status of related R&D, and remaining challenges and open issues, mainly based on the presentations and the discussion at the meeting.

1. INTRODUCTION

The Fukushima Daiichi accident highlighted some areas where the knowledge and understanding regarding severe accidents (SAs) in water cooled reactors could be strengthened to enhance nuclear safety.

The IAEA held, in cooperation with the OECD/NEA, the International Experts' Meeting (IEM) on "Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant", February 2015, to facilitate the exchange of information on these R&D activities and to further strengthen international collaboration among Member States and international organizations.

It was one of the main conclusions at the IEM that the Fukushima Daiichi accident had not identified completely new phenomena, but it highlighted several challenges that should be addressed by reconsidering R&D strategies and priorities. It has been highlighted that the R&D area regarding in-vessel melt retention and ex-vessel corium cooling/stabilization is one of the highest priority areas, and that better understanding of accident progression should be gained to reduce uncertainties as to the effectiveness of these strategies.

As one of the follow-up meetings for the IEM, the IAEA held the Technical Meeting (TM) on "Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling", Shanghai, China, October 2016. More than 60 experts from 18 Member States exchanged information on their activities, discussed the recent progress and current status of related R&D, and highlighted remaining challenges and open issues.

2. OBJECTIVE

The objective of this paper is to summarize and disseminate the major outcomes from the above-mentioned TM, especially the technical consensus established among the participating experts.

The TM included 6 Technical Sessions (TSs): 4 TSs on in-vessel melt retention (IVMR) and 2 TSs on ex-vessel corium cooling (EVCC). In total, 33 presentations were given, each of which was followed by active technical discussion. General issues related to the two main topics, i.e., IVMR and EVCC, were discussed at Discussion Sessions (DSs), respectively. A draft meeting summary of the highlights from the TSs and DSs was reviewed, discussed and agreed by the participants at Summary Sessions. All presentations and the meeting summary are available on the IAEA website [1], and an IAEA Technical Document (TECDOC) summarizing the TM outcomes will be prepared in 2017.

3. PHENOMENOLOGY AND TECHNOLOGIES REGARDING IVMR

It is commonly recognized that the IVMR strategy achieved by external reactor vessel (RV) cooling and/or in-vessel flooding is one of the most effective measures to prevent the melt-through of RVs during severe accidents in water cooled reactors. Several operating nuclear power reactors and new ones use or aim to use IVMR strategy, and some of them have dedicated systems.

When a SA occurs, the core melt usually relocates into the lower plenum and forms a molten pool. The radioactive fission products in the molten pool continue to generate decay heat. One of the two success criteria of the IVMR strategy is 'thermal criterion' to make sure that the heat flux from in-vessel molten pool is less than the critical heat flux (CHF) at the outer surface of the RV lower head (LH) that is determined by external cooling conditions with water pooled in the reactor cavity. The other success criterion is 'structural criterion' to ensure the long-term integrity of the RV taking into account ablation (i.e., thinning of the RV wall) by the molten pool and survivability of penetrations and welds at the RV LH.

3.1. Recent Progress and Current Status of R&D on IVMR

A lot of R&D activities have been done and are still on-going to develop IVMR strategy and technologies at national, regional and international level, and the Fukushima Daiichi accident revitalized it. Most of the efforts have been made in understanding of key phenomena, both inside and outside of the RV, with experiments and numerical analyses, code improvement/validation, and application of IVMR strategy to specific reactors and its optimization.

It is understood that molten pools can separate into either 2 layers (i.e., lower oxidic and upper light metal layers, or upper oxidic and lower heavy metal layers) or 3 layers (i.e., heavy, metal oxidic and light metal layers). Behaviour of the upper metal layer and impact of its thickness on heat flux focusing (so-called 'focusing effect') has been a subject of research in many organisations because it determines the maximum heat flux from the RV.

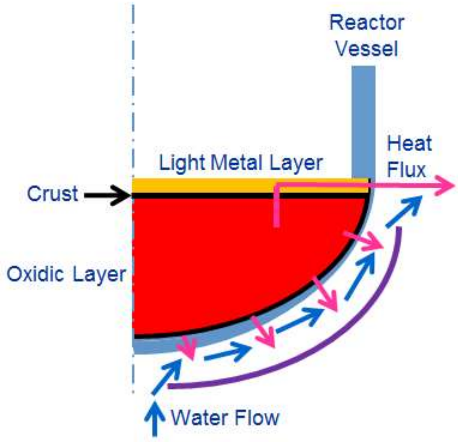

FIG. 1. Schematic view of IVMR.

During the past years a significant progress has been achieved in understanding and modelling of behaviour of molten pools in the RV, and new tests are expected to provide data for the validation of computer codes allowing them to be used at reactor-scale conditions, which was difficult to achieve.

Main factors affecting the maximum heat flux able to be removed by external water flow (i.e., CHF) include: stability of the natural circulation; outer surface conditions of RV LH; geometry of the flow path; and water subcooling at the inlet of the flow path. Recent R&D results suggest that the most effective measures to increase CHF might be optimisation of the flow path and outer surface conditions.

The application of IVMR strategy requires the following design considerations such as: depressurization of the reactor coolant system (RCS); water source for external reactor vessel cooling and/or in-vessel flooding; initial flooding, followed by a long term water supply; venting and condensation of generated steam; and management of potential negative impacts if the IVMR with external cooling fails (e.g., threat of steam explosion in the containment). Various kinds of active and passive systems have been designed. It is agreed among experts that the probability of successful retention of melted core in RVs is generally higher in lower power reactors, but it also highly depends on the specific designs.

3.2. Remaining Challenges and Open Issues on IVMR

Regarding phenomenology inside the RV, behaviour of stratified molten pools is still a key issue and requires additional information in terms of experimental data and material properties. Especially, transient behaviour of molten pools is important to determine local heat flux values which may result in larger threat to the integrity of the RV than the fully-developed (steady) state. Larger scale corium pool tests are desirable to provide more realistic data.

There are still uncertainties in phenomenology inside the RV, and they mainly come from insufficient knowledge in key phenomena related to accident progression and limitations of experimental facilities and instrumentation.

Larger scale corium tests are desirable to provide more realistic data, and it is still open how to compare results obtained in facilities having different scales and what is the influence of geometry of test facilities.

Phenomenology outside the RV is much clearer than that inside the RV: a lot of CHF data are available now. However, some of them are contradictory (e.g., the effect of surface oxidation), and experimental results should be classified and presented in a consistent way to be used easily by analysts, designers and regulators. Two new large experimental facilities are designed to measure CHF at the outer surface of the RV LH under more realistic configurations and flow conditions.

Code improvement and validation is important to simulate transient behaviours of molten pools inside the RV, water flow outside the RV and their interaction, which will give an assessment of the IVMR strategy under the given conditions. At present, different models and codes produce quite different results partly due to the absence of a proper validation matrix for IVMR-related phenomena. The current models for focusing effect and transient behaviour of molten pools need to be improved. Full height experimental facilities to measure CHF are necessary for validation data, and they should be designed as closely as possible to the real conditions.

Concerning application of IVMR strategy and technologies to reactor designs, there is a consensus among TM participants that there are not enough analyses of structural integrity made for different shapes of RVs and different pressure and temperature conditions. Hence, the structural integrity should be evaluated based on detailed phenomenology and material properties at realistic severe accident conditions. Probabilistic approaches are considered necessary as complement for deterministic approach in the analysis of IVMR strategy effectiveness.

4. PHENOMENOLOGY AND TECHNOLOGIES REGARDING EVCC

It is recognized that EVCC strategy, combined with other measures, remains the ultimate means to limit molten corium-concrete interaction (MCCI) in case IVMR is not applied or fails. Several NPPs under operation or under construction apply or aim to apply EVCC strategy.

Ex-vessel progression of SAs depends on the RCS pressure at the moment of RV failure, the possibility of containment over-pressurization, presence of water in the reactor cavity, corium cooling (e.g., by flooding), thickness of the concrete, and accident management measures to address non-condensable gases generation. There can be several paths which end by early containment failure, delayed containment failure or a manageable situation. An ex-vessel manageable situation is a situation in which the melt/debris has been cooled and quenched and kept in this state for a long time. In this paper, we focus on the phenomena related to ex-vessel corium cooling to stop MCCI and reach a manageable situation after a significant amount of the in-vessel corium inventory would pour by gravity in the reactor cavity.

When presence of water in the reactor cavity is allowed, the following melt fuel-coolant interaction can occur: (i) if an energetic steam explosion occurs outside the RV, the melt involved in the explosion may be dispersed out of the reactor cavity while the one not directly involved in the explosion can still form a particle debris bed; (ii) otherwise, the melt jet can break up in the water pool and a portion of the initial melt may form a particle debris bed and the remaining part may form a cake.

In case of dry reactor cavity conditions, the poured debris accumulates at the bottom, transfers heat to the atmosphere by radiation and convection and ablates the concrete substrate, possibly leading to containment melt-through. Reactor cavity flooding and/or corium spreading to reduce the heat flux are necessary to reach a manageable situation. Melt coolability may be improved by top flooding.

4.1. Recent Progress on and Current Status of R&D on EVCC

During the last three decades, there have been many tests (both integral and separate effect) and analyses completed and significant progress has been made towards understanding phenomenology related to MCCI and corium coolability [2].

Lessons learned from these tests and associated analyses are being applied in severe accident management planning for operating plants and in the design of new plants. They highlighted the following:

- For dry cavity conditions, melt temperature evolution and concrete ablation shape are influenced by concrete composition (radial ablation rate is faster than the axial ablation rate for siliceous concrete while a more isotropic ablation is observed for limestone-rich concrete). In a few tests, ablation appears to be more pronounced and faster oxidation kinetics in the areas where metal is in direct contact with concrete are observed;
- Under wet cavity conditions, several corium cooling mechanisms (e.g., bulk cooling, water ingression, melt ejection and crust breaching) were identified, and potential enhancement of coolability was observed.

For current plant applications, MCCI phenomena are analysed based on conservative assumptions with respect to the weakness of the containment design. In general, extrapolating the results of experiment results at plant scale requires some idealization of plant geometry and configuration; as a matter of fact, dedicated MCCI computer codes can currently analyze idealized cases where corium is spread uniformly over a dry reactor cavity. In addition, empirical correlations are needed as ablation anisotropy cannot be explained by existing phenomenological code models.

A few physics-based models for debris cooling mechanisms and debris beds have been developed and implemented in system codes to support accident management planning and plant safety analyses. However, there are limitations in addressing local corium accumulations in the case of an initially flooded reactor cavity, and very few codes model the impact of top flooding, in particular due to the lack of data for the top flooding of metal-oxide melt.

4.2. Remaining Challenges and Open Issues on EVCC

In spite of the significant progress mentioned above, the following challenges and open issues are still to be addressed:

- the lack of data to support analysis of long duration transients. As a matter of fact, tests are typically limited to a few hours while Fukushima Daiichi accident indicated that plant accident conditions can last for several days before reaching a controlled state;
- the need to perform more realistic plant simulation. Most codes are unable to address non-uniform core debris distributions and/or particle beds as initial conditions in real plant configurations. Also, realistic containment features such as deep sumps, sump drains, and cable penetrations need to be considered in analysis and testing activities;
- Data gaps in the experiment database related to EVCC concerning: (i) high metal content in the melt, (ii) rebar in the concrete, (iii) non-uniform melt accumulations, crust formation/failure mechanisms and their effect on MCCI, and (iv) the effect of raw water on coolability;
- Depending on reactor design and accident scenario, debris coolability might not be ensured for all situations. Therefore, additional engineered features might be needed to ensure coolability in some plants;
- Recriticality in debris beds formed for MOX fuel should be considered.

It is recognized from the presentations and discussions that the implementation of EVCC backfitting measures may be more complicated for operating reactors, in particular due to already existing design and layout and radiation protection issues induced by possible modifications.

5. INTERNATIONAL COLLABORATION

The TM underlined the importance of the international scientific cooperation in having better and common understanding of IVMR and EVCC phenomenology and technologies, and hence in increasing the safety level of operating and new nuclear power plants.

Several national and regional R&D programmes both on IVMR and EVCC are ongoing or planned to start soon. However, information exchange in existing bilateral and multilateral cooperation is limited among contracting parties for confidentiality reasons and funds involved, and it is difficult to disseminate the information beyond them.

The following activities could be carried out, as a complement, in the frame of international cooperation:

- Code benchmarking against well-defined experiments, including blind test calculations, will be useful for code/model development and validation;
- In the case of IVMR, RV integrity including characterization of material properties at severe accident conditions and improvement of the mechanical modelling could be an interesting topic for cooperation;
- A R&D activity on top cooling may be interesting for both IVMR and EVCC; and
- Education and training of young nuclear professionals should be a key element of future planned activities to ensure knowledge transfer in the area of severe accidents from senior to younger generations.

6. SUMMARY

The IAEA TM on “Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling”, Shanghai, China, October 2016, gathered more than 60 experts from 18 Member States, and from regulatory bodies, technical support organizations, operators, vendors/ designers, research institutes and universities. Based on the presentations, active discussions took place on recent progress and remaining challenges and issues related to IVMR and EVCC, as well as on related activities that could be performed in the frame of international cooperation.

The IAEA will continue to play an essential role in providing a platform to foster the exchange of information on recent progress and challenges in the addressed topics and its dissemination to Member States by using its different means (e.g., TMs, Coordinated Research Projects (CRPs), International Collaborative Standard Problems (ICSPs)) to support international collaboration.

ACKNOWLEDGEMENTS

The authors express their gratitude to the consultants who attended the consultancy meeting to plan the TM for their suggestions and advice. The authors also thank the participants in the TM for their contribution in presentations and discussion, and especially for their cooperation to explore consensus.

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NUMERICAL STUDY OF IN-VESSEL CORIUM RETENTION IN BWR REACTOR

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Abstract

Recently, the strategy of melt retention inside the Reactor Pressure Vessel (RPV) in case of severe accidents in the nuclear reactors is getting more and more attention worldwide. The paper presents a numerical study of severe accident in the nuclear reactor and the analysis of corium coolability inside the reactor pressure vessel. The RELAP/SCDAPSIM 3.4 code and the integrated module COUPLE were used to analyze this problem. A full plant model of a ~2000 MW thermal power BWR reactor was used and a large LOCA with total failure of cooling water injection was assumed. The paper presents full accident sequence from normal operation conditions to core heat-up, melting and relocation into lower head. The temperature of RPV wall was used as a criterion for the integrity of the vessel. The results showed that if the reactor cavity is filled (to the level above the lower head) before meltdown of the core and slumping of the material to the lower head, the RPV could be cooled so that corium should stay inside the vessel.

1. INTRODUCTION

Nuclear energy is currently counting more than half of a century of its existence. That being said, the technology still keeps advancing and evolving, eliminating more and more safety issues in the Nuclear Power Plants (NPP). The safety issues of course include severe accidents as the most dangerous scenarios for the NPPs. Recently the strategy of melt retention inside the Reactor Pressure Vessel (RPV) in case of severe accidents in the nuclear reactors is getting more and more attention worldwide. This paper presents a numerical study of severe accident in the nuclear reactor and the analysis of corium coolability inside the reactor pressure vessel. The RELAP/SCDAPSIM 3.4 code and the integrated module COUPLE were used to analyse this problem. For the analysis of the IVR strategy in BWR, the General Electric BWR-5 reactor installed in Mark-II containment was chosen. The schematic view of a typical BWR-5 with Mark II containment and are shown in Fig. 1 and Fig. 2.

FIG. 1. Scheme of MARK II containment [1].

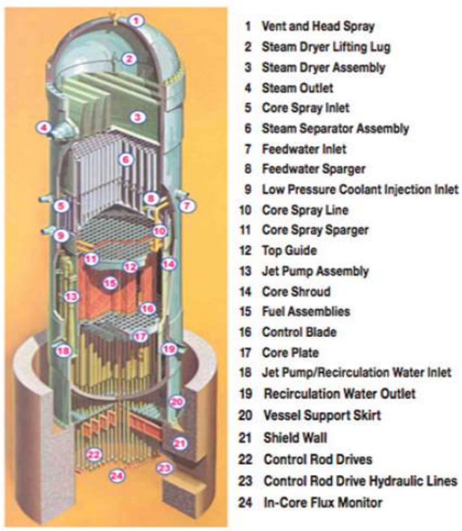

FIG. 2. Scheme of typical BWR-5 reactor [1].

2. ACCIDENT SCENARIO DESCRIPTION

In this study the worst case scenario, i.e. a guillotine break of external circulation pump suction pipe - large Loss of Coolant Accident (LOCA) with total failure of cooling water injection due to station blackout and the failure of the Emergency of Core Cooling Systems (ECCS) is considered. The main assumptions for this scenario are:

— Reactor SCRAM at time moment t = 0 s;
— Main coolant pumps are tripped at time moment t = 0 s;
— Main safety valves closes at time moment t = 0 s;
— No feed water and emergency core cooling system water supply.

This is a hypothetical scenario in order to cause the core melt rapidly and observe the slumping of corium to the lower head, which then leads to heating up of the bottom of the RPV. The primary containment is simulated and external cooling of the reactor vessel is considered. The technical measures for supplying the water into the reactor cavity are not analysed in this study. It was assumed that the cavity is filled before the meltdown and slumping of the core material into the lower head.

After the initiating large LOCA with total failure of cooling water injection accident, there were two sub-scenarios considering different conditions for lower head cooling were analysed:
The water is continuously supplied to the reactor pit so that the water level does not decrease;
The water fills the reactor cavity and then stops, which leads to gradual boil-off of the water and uncover of the lower head.

3. MODEL OF THE REACTOR

The RELAP/SCDAPSIM computer code was used to simulate the severe accident [3], [4]. RELAP/SCDAPSIM is designed to describe the overall reactor coolant system (RCS) thermal hydraulic response and core behaviour under normal operating conditions or under design basis or severe accident conditions. The SCDAP models calculate the behaviour of the core and vessel structures under normal and accident conditions.

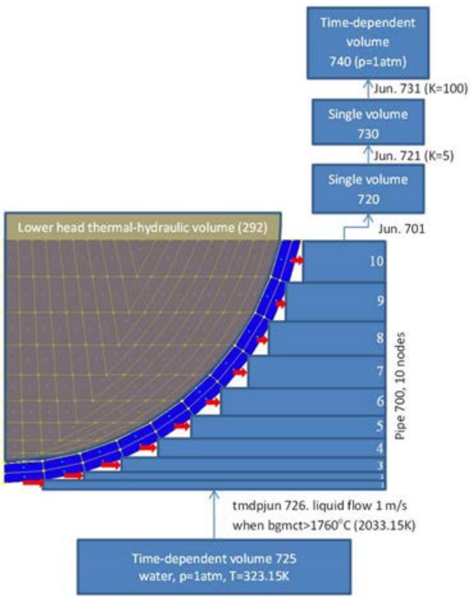

FIG. 3. Scheme of the reactor ex-vessel model.

In our analysis, as a basis, the conventional General Electric BWR-5 model, initially developed by CNSNS, was used 0, 0. Because our analysis takes into account not only the full plant behaviour, but also the ex-vessel environment to simulate cooling of the RPV exterior, the special model of lower head of the RPV was added. This RPW lower head was modelled with COUPLE 0 code, which is a part of the RELAP/SCDAPSIM Mod3.4 code0. The COUPLE model solves two-dimensional, finite element based heat transfer problem. This COUPLE model takes into account the decay heat and initial internal energy of slumped debris and then calculates the transport by conduction of this heat in the radial and axial directions to the wall structures and water surrounding the debris.

In the model the convective boundary is connected to the external surface of the RPV lower head, and the heat from COUPLE finite-element mesh is transferred to hydrodynamic volumes of RELAP5 code. The model for simulating the ex-vessel cooling is presented in Fig. The ex-vessel model consists of 2 time dependent volumes, 2 single volumes, one vertical pipe component and time dependent junction. Time dependent volumes used in order to give the boundary conditions to the developed model. 2 single volumes are used as the buffering volumes in the model. Vertical pipe component represents the available volume around ex-vessel, according the available Mark II containment data (see Fig. 2). Pipe component have 10 internal nodes. Each node is connected to the RPV vessel modelled using COUPLE. In order to have more precise calculation results flow area and height is varying according to the nodes developed for the COUPLE model. The height of the first nodes is small and increasing at the top of the lower head. In order to simulate the ex-vessel cooling conditions the time dependent junction was used.

4. RESULTS AND DISCUSSION

The objective of the analysis was to investigate the effectiveness of ex-vessel cooling, so that the RPV would stay intact. The main parameter for such evaluation was the wall temperature of the RPV wall. It was assumed in our analysis, that if temperature of the wall reaches the melting temperature of carbon steel (1723 K) the geometry is assumed as melted.

The LOCA occurs at the time = 0 s, and at the same moment the feed water stops. Reactor scram occurs immediately. After LOCA is initiated, the top of active fuel is uncovered in 32 seconds after the break, and fully uncovers in 50 seconds. The remaining water in the lower head fully evaporates in 3400 s after the accident, as it is heated by slumped debris. The more rapid decrease of the water level (at 2600 s after the accident) is due to starting of UO_2 slumping into lower head.

After the fuel is uncovered, the heat-up of the fuel rods takes place. The first rupture due to ballooning occurs in 421 s after the accident. In case of no ECCS in operation as it was assumed in our work, the heat up of the reactor core progresses and ends up with melting of the core materials and relocation of the debris into the lower head.

The melting and relocation of the core begins with the lighter metals, and the earliest slumps of debris contain Fe, Zr, ZrO_2 and B_4C absorber. The first slump is observed at 847 s after the accident. The slumping of debris to the lower head by material type is shown in Fig.

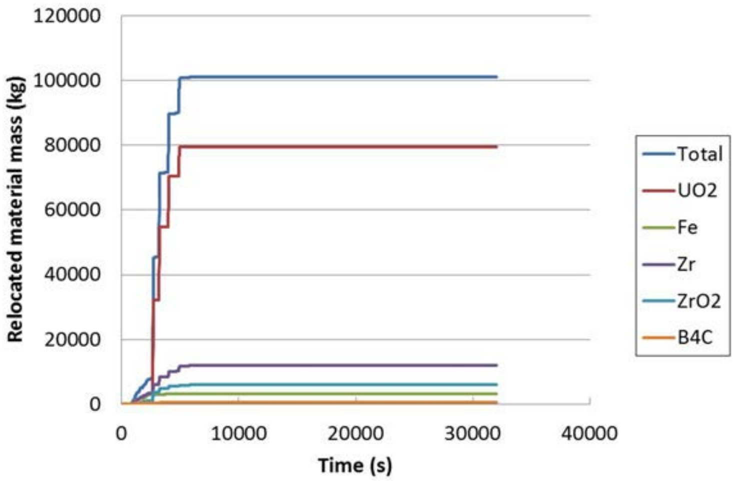

FIG. 4. Mass of different materials in the lower head.

In our case the reactor cavity is filled with air during steady state, and then is filled with water before the debris starts slumping into the lower head. It was assumed in the model that the injection of water in the ex-vessel cooling system starts when the peak fuel cladding temperature exceeds the melting temperature of carbon steel, before the beginning of corium slumping into the lower head. It was assumed, that the water supply stops after filling the reactor cavity just a bit above the lower head.

The temperature of the external RPV wall is shown in Fig. The reference nodes correspond to the model shown in Fig., where the 1st node is at the bottom, and are placed along the external wall up to 11th. As we can see from the results, the external surface temperature is maintained around 430 K and the critical heat flux is not reached. As long as the wall is submerged in the water, it is cooled successfully. When the water level decreases and the wall dries out, the overheating and melting of the RPV wall is inevitable. Therefore, the main factor for the given reactor is to maintain the water level in the reactor pit.

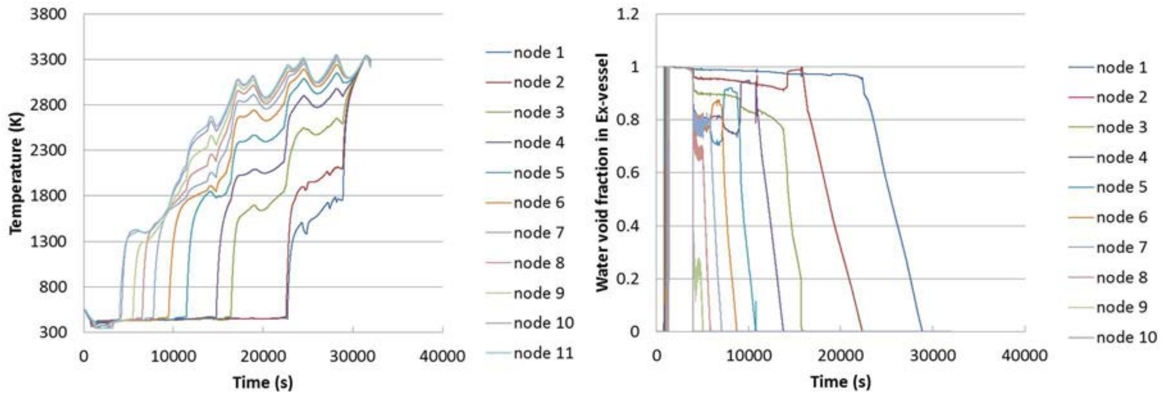

FIG. 5. Temperature and void fraction at the external wall surface of the lower head.

The wall thickness of the lower head is 190 mm, and due to high power in debris the steel wall starts melting from the inside. RELAP/SCDAPSIM does not take into account melting of the wall, but the remaining wall thickness was calculated by the temperature profile.

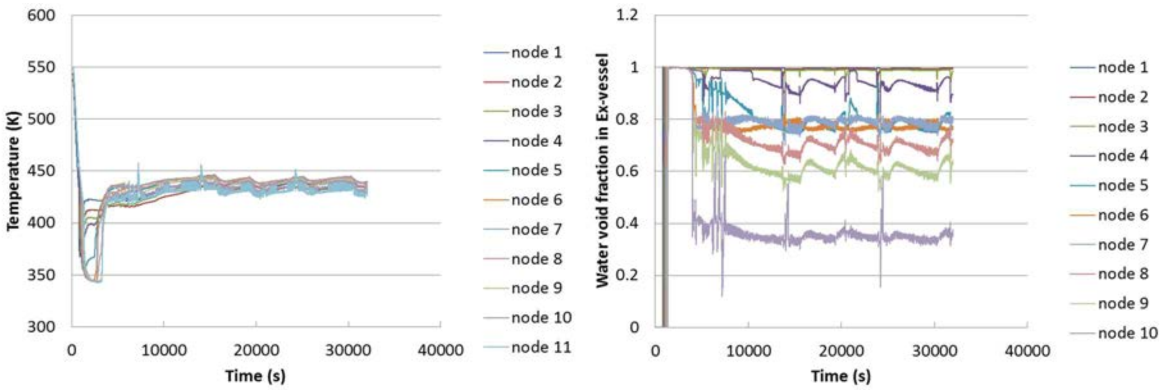

FIG. 6. Temperature and void fraction at the external wall surface of the lower head.

The results of this first calculation show that if the lower head is submerged into the water, the heat is successfully removed and no CHF is expected to occur. For this reason, additional calculations were performed to determine the minimum water supply to the reactor pit, which is needed to avoid the dry-out. The analysis showed that if the water supply of 15 kg/s to the reactor pit is maintained, the in-vessel retention is achieved successfully. The results of this calculation are presented in Fig. As we can see from this figure, the external wall temperature remains stable at around 430 K, which means it's cooled and no CHF has been achieved.

5. CONCLUSIONS

For the analysis the most conservative initiating event – guillotine break of suction pipe in recirculation loop with total failure of cooling water injection was selected. This scenario leads to the very fast overheating of the core and the start of fuel melting at high decay heat level (~600 seconds after the beginning of the accident). The dry-out of the lower head occurs approximately 1 hour after beginning of the accident. It was assumed in the modelling that the reactor cavity (pit) around the reactor vessel is flooded by water in advance. For this initiating event two cases were analysed:

a) without additional supply of water in the ex-vessel compartment (insufficient cooling, which leads to the reactor vessel failure);
b) with constant cold water flow (15 kg/s) around the lower head which allows the sufficient lower head cooling through the wall of bottom part of reactor vessel; the heat transfer coefficient from external surface of reactor vessel varies in interval 5000 – 7000 W/(m^2 K).

The modelling results show that critical heat flux is not reached in the analysed cases. Only the dry-out of water due to evaporation and decreasing water level can cause the failure of the RPV. Therefore, for given cases, if the reactor cavity is flooded, the ex-vessel cooling is sufficient and failure of the lower head does not occur. Although it must be mentioned that the analysis did not take into account the failure of the guide tubes, penetrating the lower head of RPV. In case of uncover of the external surface of the lower head (below the level of the debris bed) the RPV failure occurs almost instantly.

ACKNOWLEDGEMENTS

The work is performed in the frame of project "In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs" – IVMR under H2020 grant agreement No 662157.

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CORIUM IN VESSEL RETENTION STAKES AS REGARDS THE SAFETY DEMONSTRATION

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Abstract

Within the severe accident strategies and safety demonstration, In Vessel Retention (IVR) has emerged as a topic of major interest within the international community.

The variety of the associated phenomenon has been and is currently subject to considerable international R&D: corium formation, relocation and composition; permanent and transient phenomenon which constrain the heat flux through the vessel; critical heat flux reachable through outside vessel cooling; vessel thermochemical and mechanical behaviour; steam explosion physics; vessel pit and containment robustness under steam explosion; etc.

Given the stakes as regards the residual heat removal, the basemat protection against ablation and the potential consequences of steam explosion, IVR safety demonstration mobilizes significant discussions at global level.

These are all the more vivid than IVR considerations shape some structuring options of the containment lower part design and construction and that some of the questions involved are related to the practical elimination of situations which could lead to early and large releases.

In order to contribute to the discussion, this paper addresses:

- The phenomenon driving the critical heat fluxes, the order of magnitude of heat flux which can be reached and the associated vessel strength;
- The phenomenon concerning the corium behaviour inside the vessel and the associated heat flux (orders of magnitude and uncertainties);
- The design options to be considered.

The paper will present the main parameters to master in order to assess an IVR safety demonstration. It addresses the roles of the deterministic and probabilistic approaches and presents the practical elimination questions related to IVR.

1. INTRODUCTION

To cope with consequences of a severe accident with core melting in Light Water Reactors (PWR, VVER, BWR), In Vessel Corium Retention (IVR) strategies with external cooling of the vessel have been incorporated for the back-fitting in some GEN 2 reactors and in some new GEN3 reactors.

The corium decay heat is removed through the vessel wall, by boiling of water injected in the reactor pit. Thus doing, the vessel wall can be partially ablated. Because there is a limitation of the effectiveness of such cooling, corresponding to the Critical Heat Flux (CHF), safety margins towards the vessel failure, which could occur in few minutes at dry-out limit, are reduced for large power reactors (e.g PWR power above 600 MW).

In the light of remaining uncertainties during the corium relocation in the vessel lower head, impacting the heat flux distribution, it had been assessed that the probability of vessel failure could not be negligible even for around 1000 MW PWR [7]; for some new high power reactors, ex-vessel retention strategies with implementation of core-catchers have been preferred.

In case of vessel failure with IVR, the interaction of the corium jet released with the water of the reactor pit could lead to an ex-vessel steam explosion. Steam explosions are still an open R&D issue; Ex-vessel steam explosions which could threaten the containment should be practically eliminated because it could lead to large and early radiological releases.

On the other hand, a demonstration of the robustness of IVR solution and the evaluation of the associated reactor power limit providing margins against the vessel failure risk remain serious challenges. They deserve reassessing by taking into account new R&D insights, in particular those related to the uncertainties of thermochemical phenomenon within the corium.

The main phenomenon and main parameters to master in order to build and assess an IVR safety demonstration are presented below.

2. MAIN PARAMETERS

The IVR effectiveness evaluation is mainly based on the results of recent phenomenological analyses and the evolution of the state of art of knowledge over the past years related to the following three parameters:

a) In-vessel behaviour of the corium: the transient configurations of corium phases' stratifications leading to loads on the vessel and a focusing effect higher than previously anticipated.

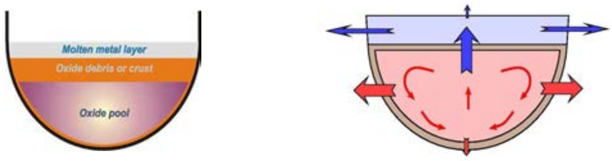

FIG. 1. In-vessel behavior of the corium.

Because of the miscibility gap between in-vessel components of corium (U-Zr-Fe-O), stratification occurs between the oxide pool (U02, ZrO2, Zr) and the metal phase (Fe, U, Zr) in long term configuration, with lower density metal located above the oxidic pool surrounded by a crust limiting the heat transfer to the vessel wall.

The turbulent convective heat transfer in the oxidic pool (which contains the major part of the decay heat) impacts heat distribution sideward and upward to the top metal layer; it is an important parameter. One part of the heat in the top metal layer is removed by heat radiation upwards to the surrounding structures (around 20 to 50%), according to their emissivity which is also an important parameter. The main part corresponds to the lateral heat transfer to the vessel wall, which induces a peak of local heat flux named "focusing effect".

The sideward power of the top steel layer, which generates the focusing effect and ablates locally the vessel wall, depends on the thickness of the top metal layer: the thinner the metal layer, the stronger the focussing effect. The thickness of the top metal layer can indeed be considered as the important critical factor for the success of the in-vessel corium retention.

The decay heat is also an important parameter impacting the level of heat fluxes entering the inner wall of the vessel; it is why core melt scenarios with rapid core degradation beyond the scram, as a Large or Medium break LOCA associated with the unavailability of safety injection would be the most critical scenarios.

b) The CHF which can be credited to remove the decay heat, depending on the optimization of the water circulation in the reactor pit and on the outside vessel wall cooling. The local flux at outer vessel wall is compared to local CHF.

c) The residual thickness and associated mechanical behaviour of the vessel partially melted by the heat flux from oxide and in particular steel layers.

3. REACHABLE CHF BY OUTSIDE VESSEL COOLING

The effectiveness of external cooling of reactor pressure vessel has been examined for more than 20 years. Cooling of outer vessel wall, by submerging the vessel pit at severe accident signal, should be done before corium relocation in the lower head.

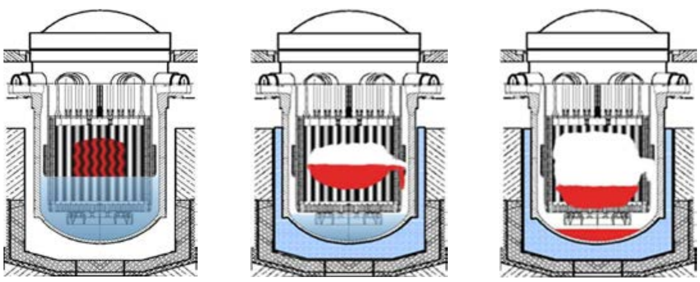

FIG. 2. Relocation of corium.

The dry-out limit is based on the local flow characteristics. This limit is higher when the vessel wall is vertical (resulting in easier steam evacuation) or when the water is sub-saturated at the inlet of the external vessel cooling system. Submerging the vessel is not sufficient to remove heat flux higher than 1 MW/m²; it is why many solutions have been investigated to prevent the film boiling regime and to ensure an efficient heat removal by nucleate boiling.

3.1. CHF enhancement by optimizing water circulation in the pit

An optimization of the shape of the vessel insulation is necessary to enhance a natural circulation, ensuring with an upward flow along the vessel and a downward flow of cooler water. This requires implementation of adequate clearances and passive water/steam flow openings to prevent steam blockage and water inlet at bottom of the insulation and steam vents near the top.

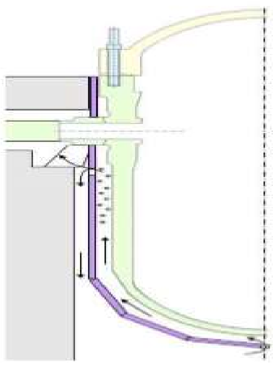

FIG. 3. Circulation of water in the pit.

The enhanced geometry aims at streamlining the two-phase motions, thus reducing the pressure drop and inducing more liquid flow in the annular flow channel at a given heat flux. There is an international consensus that CHF in the range of [1.5 MW/m²-2MW/m²] can be obtained in this way.

Solutions with implementation of a forced convection during a certain time which could be combined at long term with a natural convection of water could be also efficient but with a drawback to have to rely on an active system at a very earlier phase of the severe accident.

3.2. CHF and wall/water Heat Transfer Coefficient enhancements

Some significant enhancements (factor higher than 1.5 on CHF) have been obtained by treatment of the vessel material surface; in addition, real oxidation conditions met during reactor life are favourable with comparison of some experimental tests. The oxidation behaviour of pure iron and iron-based alloy can change physical and chemical properties of material and dominantly affects boiling heat transfer capability.

Boiling architecture can be optimized by implementing a nano-particle-based coating for the vessel, which slows the phenomenon of dry-out. Modifying porosity characteristics can also increase the heat transfer [8]. To increase nucleate boiling sites can also provide higher CHF.

In addition to material surface treatment, an approach is to find solutions to be injected into the reactor pit water in order to prevent film boiling by increasing the Leiden frost temperature.

The R&D research concludes that CHF significantly higher than 2 MW/m² could be obtained. Nevertheless, the limiting factor would in this case be the residual thickness of the vessel, It is reduced to 1 or 2 cm (for the material at a temperature < 650°C, ensuring the load bearing capacity) instead of the initial thickness in the range [15 cm-20cm]), when the heat flux at the outer wall reaches the range [1.5 – 2MW/m²].

There are some uncertainties in calculations of core degradation and relocation into the vessel lower head, because expected 3D effects are generally not considered. To cope with these uncertainties and to better ensure a long-term resistance of the vessel, some margins regarding its thickness must be taken account. These margins could be also necessary regarding to the risk of pressure spikes which are possible in case of reflooding in the vessel, even though the mitigation strategy has required to fully depressurize the primary circuit.

It therefore seems of little interest to justify a significant enhancement of CHF, because it would be associated with a higher wall ablation: the range [1.5-2MW/m²] seems a reasonable limit, and related R&D to justify such heat transfer can be well mastered.

4. CORIUM LAYERING AND FOCUSING EFFECT AT THEMOCHEMICHAL EQUILIBRIUM OF THE MOLTEN POOL

The maximum heat flux from the corium is the most important parameter, and the one with the highest uncertainties. It remains to be determined with a sufficient accuracy to justify some margin towards the vessel failure risk. Recent assessments, identifying the need to address transient thermochemical phenomena with mass transfers between relocated corium layers due to the changes of density of metal and oxide phases, have outlined the risk of increased heat loads with comparison to previous methodology applied at the beginning of years 2000, which used to solely address the final state when all the corium is in the lower head.

4.1. Corium layering in safety analysis

Many analyses to determine the focusing effect have considered that the bounding configuration would be the final configuration of all corium and its decay heat power relocated into the vessel lower head with a top steel layer including all the molten steel from melting of internal structures and of the vessel) in direct contact with the vessel located above the oxidic pool embedded by a crust which is a thermal resistance.

The validity of two- layers configuration has been questioned when the OECD MASCA R&D program [9] showed that the addition of steel in a sub-stoichiometric corium (UO2-ZrO2-Zr) may extract Uranium and Zirconium from the reduction of UO2 and ZrO2 at the top metal interface and form a metal alloy containing Uranium-Steel and Zirconium which becomes heavier than the oxidic pool. Such phenomena with downward migration in the bottom of the vessel of a heavy metallic layer depends on some conditions, in particular the ratio of Uranium and Zirconium in the corium as well as on the degree of the Zirconium oxidation during the core degradation, which is dependent on the core melt scenario.

The compositions of the three layers, a light metal at top, an oxidic layer at the middle, and a heavy metal layer at the bottom, can be determined, by using thermodynamic calculations assuming global thermochemical equilibrium of the corium, without addressing the transient processes of mass transfers between the different layers.

The formation of heavy metallic layer has the drawback to reduce the thickness of the light metal layer and to increase the focusing effect [1], [10].

4.2. Potential design improvements: additional in-vessel reflooding

Some PWR applications as in [3], have shown, according to their assumptions, that the vessel integrity could be challenged, because the reduced thin metal layer would lead to maximum heat flux significantly higher than 2MW/m² to overcome the related worsened focusing effect and thermochemical uncertainties, implementing a system of in-vessel reflooding to be activated during the severe accident has been proposed to complement the outside vessel cooling.

Because the heavy metal layer (and therefore removed from the top metal layer) is known to be strongly dependent on the mass of un-oxidized zirconium mixed into the oxide debris, the purpose of the in-vessel reflooding can be to oxidize more zirconium, reducing the amount of metallic uranium that may form in debris bed, and preventing according to experimental MASCA results, the formation of the three corium layers.

At the same time, the in-vessel reflooding would extend the time to dry out and reduce the decay heat of corium and associated heat loads through the vessel wall.

In addition to other design measures to increase steel masses during the mixing with corium to be able to increase the metal layer thickness and to reduce the heat fluxes, the implementation of an in-vessel reflooding requests to justify its reliability, because its seems difficult to guarantee a systematic availability of injection whereas the core melt started due to unavailability of sufficient in-vessel cooling.

Dedicated accumulators (pressurized tank), dedicated core make-up tanks, both sides of which are connected with the primary circuit, and elevated gravity tanks are contemplated solutions.

A question mark regarding such a solution is indeed related to the safety demonstration: how to justify the capability of fast actuation of a safety injection in the vessel in a core melt situation, created by the lack of injection in the vessel

5. TRANSIENT EVOLUTION OF CORIUM LAYERS AND WORSENED FOCUSING EFFECT RISK

5.1. Transient layering of corium

The melting of steel internals, core support plate, partially the vessel, is a transient process, as the oxide relocation into the lower head. The conditions of corium stratification of corium and the evolution of the different layering depend on the kinetics, which is plant and scenarios dependent. By taking into account thermochemical effect and by including in the severe accident codes the transient mass transfers between layers due to the changes of density of oxide and metal phases, it could be possible to meet transitory worsened focusing effect compared to the assumed final bounding configurations addressed in many safety analyses.

To identify risk of thinner metal layer topping significant oxidic corium is one target of new on-going development of kinetics models dealing with corium phase stratification.

The previous modelling of steady state configuration, when all corium is located in the lower head, has been determined at a global thermochemical equilibrium, whereas the experimental tests with prototypic corium have shown that a downward mass transfer of steel through the horizontal crust between oxide and metal layer could be possible.

In addition to the possibility of heavy steel layer formation with downward flow, when the oxidation level of the oxide phase is low, MASCA tests also identified possibility of reverse flow of steel from the bottom heavy layer to the top: the reduction of UO2 and ZrO2 at the top metal interface produces Oxygen which can slowly oxidize the bottom metal layer until a limit of Oxygen concentration in oxide pool,, leading to transfer U and Zr from the heavy steel layer into the oxide phase and an upward flow of super- heated steel up to the horizontal upper crust, and through it to the top the pool. According to the available steel at top, such arrival of heated steel from bottom could also lead also to a worsened focusing effect. This upward metal transient will be limited by the Uranium mass transfer from the heavy layer to the oxidic layer. The knowledge of Uranium mass transfer is mandatory and an experimental validation is requested.

After the first relocation phase, different configuration of initial corium layering is possible before starting the vessel melting:

- Only bottom metal layer (U, Zr, Fe) with only oxide (U, ZrO2, Zr) above, when the relocated amount of steel is small and a high amount of heavy metal is not fully oxidized in the oxidic phase: in this case, if the top layer is only fed during a certain time, during the growth of top steel layer; high focusing effect could then occur with corium heat flux higher than 2 or 3 MW/m² [11] for a 1000MWe PWR.
- For higher steel masses at first phase of corium relocation, other initial configurations with steel above the oxide is also possible, and worsened focusing effect could occur during the thermochemical downward transfer through the crust of top steel, by reduction of its thickness.

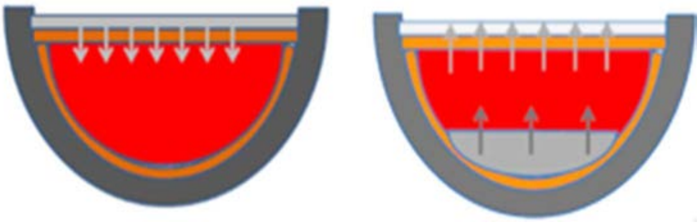

FIG. 4. Thermochemical steel transfers through horizontal crust.

Three phases of possible transient focusing effect could worsen the loads:

- During the growth of a top metal layer (high flux obtained with some tons of steel, around 10 tons).
- During the reduction of a top metal layer with downward flow through the horizontal crust

— During an upward flow of super-heated steel from heavy bottom metal layer reaching the top of the pool.

5.2. Transient models development

These transient transfers cannot be supported by the use of a global thermochemical equilibrium assumption which is only reached at long term: new models are developed to better represent the stratification kinetics. Various assumptions can be considered in new models [2], [5], [6] regarding the molten steel pathway:

a) upward pathway of molten vessel steel which directly feed the top layer in direct contact with the vessel

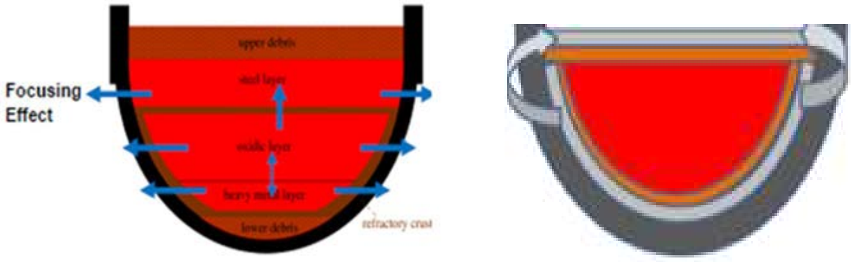

FIG. 5. (a) layers (b) upward pathway of molten vessel steel.

It is applied in PROCOR and MAAP 5 EDF version codes development [2] and [6] in which different permeability of the horizontal crust are considered.

This pathway is assumed to be caused by the pool weighing down on the crust and pushing the steel upward. The heavy metal, oxide and light metal layers are enclosed by the oxide crust and supposed to be close to the equilibrium state. Under certain conditions, the current in-equilibrium position of the layers depending on the current chemical composition of the pool can change, producing an inversion of stratification calculated. The out-of-equilibrium metal layer corresponding to the melted steel of the vessel is located above the crust in direct contact to the vessel and provides the focusing effect.

Fig. 5 shows an evolution of lateral heat flux of the focusing effect metal layer during the decrease of top metal layer thickness with downward mass transfer through the crust (PROCOR code application)

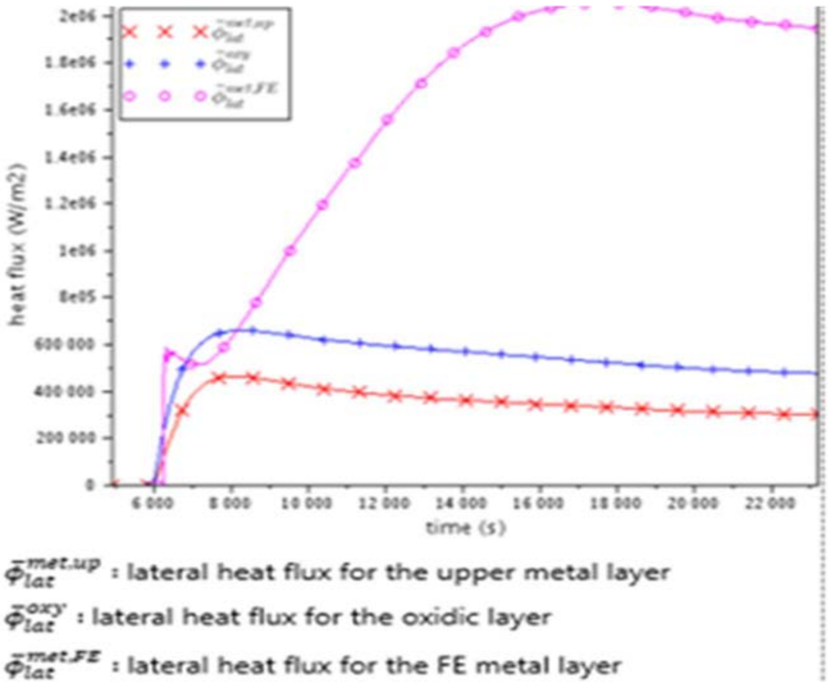

FIG. 6. PROCOR code application, example of lateral heat flux evolution.

b) Molten vessel steel incorporated in corium layers (oxide/metal)

It is applied in ASTEC code [5] by assuming that all the lateral crust along the inner vessel wall is not a physical barrier. In such model, adding a certain amount of steel in the whole molten pool, where a heavy bottom layer formed at corium relocation is covered by oxide corium, can lead to the inversion of stratification with an upward flow of heated steel through oxide layer reaching the top of the pool and leading to a transient focusing effect. Fig. 6 is an example of maximum heat flux calculated by ASTEC code in [5] resulting from such inversion of stratification.

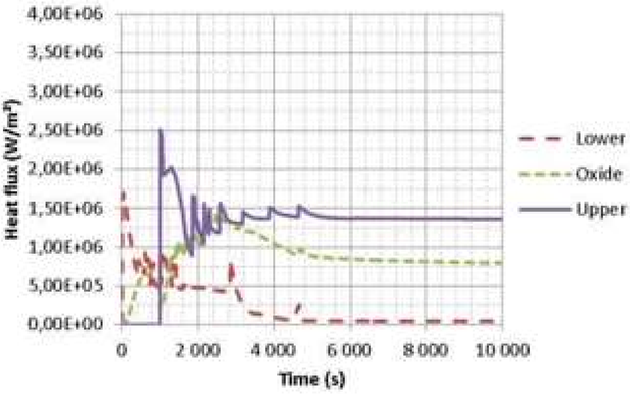

FIG. 7. ASTEC code application in [5] example of lateral heat flux evolution.

These transient models are expected to determine worsened focusing effect with comparison to steady state configurations. Many developments are carried out in the frame of on-going European Union IVMR project.

Transient evolutions induce non-uniformities such as local variations of compositions and temperature gradients which may result in conditions far from the overall thermochemical equilibrium conditions. This leads to uncertainties in the predictions, in particular those induced by scale effects of experiments: R&D has already been identified in order to validate mass transfers between layers, depending on crust resistance permeability, oxidation progression in corium pool and Uranium diffusion.

5.3. Specific phenomena to be addressed related to focusing effect

a) The 2 D heat transfer in the vessel wall

During the thermal ablation of the vessel wall, the 2 D heat diffusion in the wall could maintain the heat flux below CHF assumed in the range [1.5-2 MW/m²]. In [4], analysis performed in steady state conditions outlines that a factor up to 2 could be obtained between inner and outer heat flux. Even though a gain is expected, such result is questionable in transient conditions during the local vessel ablation. The differences between maximum inner and outer wall fluxes can depend on the duration of the peak of heat flux at top layer.

b) The natural convection in the top steel metal layer which impacts the maximum sideward heat flux of the focusing effect

When top layer is fed by molten steel at molten temperature, a period of time is necessary to develop the natural convection and the maximum radial heat flux. For some scenarios where high heat fluxes in the top layer are only reached during a spike, such grace period could limit the heat flux through the wall.

According to the steel layer thickness, the effectiveness of natural convection cells which transfer heat to the vessel wall have to be determined: for very thin steel layer, less than around 10 cm, a limitation of radial heat flux is expected; on the other hand, with larger thickness, natural convection in transient conditions could lead to higher temperatures of molten steel at the upper part of the layer in contact with the wall.

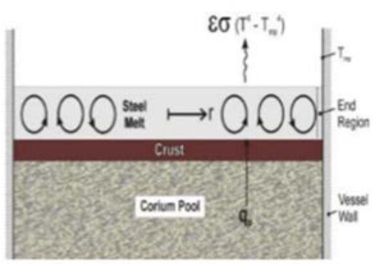

FIG. 8. Natural convection cells in the top metal layer.

6. REMARKS ON METHODOLOGY OF IVR SAFETY ANALYSIS

Recent analysis and insights from R&D results [1] have outlined that some important parameters as enhancement of CHF and load bearing capacity are well mastered and that significant uncertainties regarding transient evolutions of corium phases could challenge the standard approaches to compute the maximum heat flux and focusing effect. It would be necessary to update the models coupling thermochemical and thermal-hydraulics phenomena of corium in the vessel lower head, in order to determine, by taking into account worsening and favourable factors, the safety margins regarding vessel failure risk for selected core melt scenarios.

When the robustness of IVR with safety margins cannot be justified in all cases, it means that probability of vessel failure is not negligible: corium would then be released into the water filled reactor pit, through the vessel break, and could lead to an ex-vessel steam explosion. In a defence in depth approach, the steam explosion safety case is therefore to be resolved.

Ex-vessel steam explosion mechanical consequences are depending on the reactor pit design, with risk of missile formation which could hit the containment liner, or loose the leak-tightness of containment penetrations due to vessel movement. There are still uncertainties to evaluate with sufficient accuracy, at the reactor scale, the impulses of an ex-vessel steam explosion and there is no international consensus on consequences of such a scenario.

7. CONCLUSIONS

Among the parameters which drive the IVR effectiveness (for which outside vessel cooling should be limited by CHF in the range [1.5-2MW/m²] to keep a sufficient ablated vessel thickness), the most important and the one which shows the highest uncertainties is the maximum heat flux generated by the corium through a top layer of steel, leading to the focusing effect. It is related to the thickness of top steel layer of corium in contact with the vessel, which depends on the competition between the kinetics of evolution of stratified layers of corium and on the kinetics of steel addition from melted vessel and relocated internal structures.

New insights of corium R&D and new development of models let think that worsened focusing effect, compared to steady state analysis, could challenge the vessel integrity when the transient layering of corium phases is addressed during the corium relocation into the vessel lower head, by taking into account thermochemical effects. Significant progress of R&D is expected in the frame of on-going European Union IVMR project.

At the light of new analysis, it will be possible to re-assess the ability to establish an IVR safety demonstration which provides sufficient margins as regards vessel failure, steam explosion and containment integrity, which is the main safety objective under severe accident conditions.

The need to have to implement an in-vessel injection in case of a severe accident to overcome with uncertainties is a question mark concerning the safety demonstration.

When IVR reliability cannot be guaranteed, in a defence in depth approach, the ex-vessel steam explosion case is to be resolved. It is still an open R&D issue with large uncertainties. Safety margins are required in this field, as steam explosion, which could jeopardize the containment and lead to early and large radiological releases has to be practically eliminated. The ex-vessel steam explosion risk appreciation appears as a major driver as regards IVR: continued R&D and international consensus are much needed.

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IMPLEMENTATION OF THE IN-VESSEL RETENTION STRATEGY FOR RIVNE NPP UNITS 1, 2

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Abstract

According to the Comprehensive Safety Improvement Program for Ukrainian NPPs one of the measures to be implemented is an "In-vessel retention strategy for WWER-440 (Rivne NPP Units 1 and 2)", which has been included in the Program as one of the so-called 'post-Fukushima' safety upgrades based on results of the 'stress tests' conducted for all operating Ukrainian reactors in 2011-2012. The paper discusses the results of the first analytical studies conducted to support implementation of the strategy at Rivne NPP (RNPP). As a first step a screening of existing similar technical measures at European WWER-440 plants (Loviisa and Paks, etc) and analysis of applicability were done. Using the same calculation strategy as Paks NPP, calculations were done for specific conditions and geometry of RNPP Units 1, 2. Reactor core melting and corium relocation to the bottom was simulated with MELCOR. Flooding of gap between the reactor vessel and cavity, efficiency of external cooling and circulation of water in cooling loop were calculated with AHLET. Based on calculation results the recommendations for preliminary design of ex-vessel cooling system were proposed.

1. INTRODUCTION

Ukrainian utility, NNEGC "EnergoAtom", which operates all 15 WWER Units, is obligated to implement the Comprehensive Safety Improvement Program for Ukrainian NPPs, – which is a set of technical measures strengthening safety of operational Units. One of the key safety measures, which have increased significance after Fukushima accident, is "in-vessel retention" of corium in case of severe accident. Integrity of reactor vessel gives a chance to significant reduction of radiological accident consequences. Other major aspects are a reduction of hydrogen generation and lower pressure in the containment. Thereby "in-vessel retention" (IVR) measure is co-jointed with two other measures: installation of passive autocatalytic recombiners (PARs) and containment filtered venting.

IVR measure has been realized on most of WWER-440 reactors in Europe. Large amount of borated water in Bubble Condenser Tower together with sufficient elevation difference allows to deliver passive system with high reliability and performance.

Analysis of IVR strategy implementation on WWER in EU countries shows that general approach was almost the same for all units: Loviisa, Paks, etc. And the most challenging problem was the sufficiency of gap between reactor vessel and concrete cavity to provide flow rate and heat sink for reactor cooling. Thus, the main objective of investigation was defined as follows, to check the applicability of common approach to the real geometry of Rivne Units 1, 2 and to recommend a preliminary design of ex-vessel cooling system.

2. OTHERVIEW OF EXISTING IVR STRATEGIES AND SELECTION OF PREFFERABLE DESIGN

An otherview of ex-vessel cooling system, implemented on WWER-440 in EU countries has shown two different technical designs:

- Fist one is implemented in Loviisa NPP [1], [2], and;
- Second one was realized on several NPPs in Eastern Europe. Paks NPP, could be taken as a reference Unit [3].

The principal difference between these variants is a design of the biological shield below the reactor bottom. In first case, the shield is moved down by pneumatic driver to ensue enough free space for sufficient cooling. Alternative idea is to save the initial design of the shield and to install float passive valve, which would opened by itself after flooding of reactor cavity. In this case, the coolant will circulate in the gap between reactor vessel and existing thermal shield. Second case is much easy, cheaper and do not lead to additional significant exposure of staff. Respectively, it was decided, to try adopting this design for Rivne Units 1 and 2.

Principal scheme of the ex-vessel cooling system is presented below on Fig. 1. The cooling circuit consists of:

— Inlet valve located in the corridor connecting BCT and reactor unit compartments and provides water to the ventilation ducts in concrete;
— Part of the ventilation ducts isolated by u-type siphon, this predicts water junction in other compartments;
— Reactor cavity compartment;
— Float valve installed on the biological shield below the reactor bottom;
— A gap between the reactor bottom and the shield;
— The vertical heated part in the annular gap between the reactor vessel and the reactor cavity with biological shield;
— Outlet between the reactor vessel and the bearing ring.

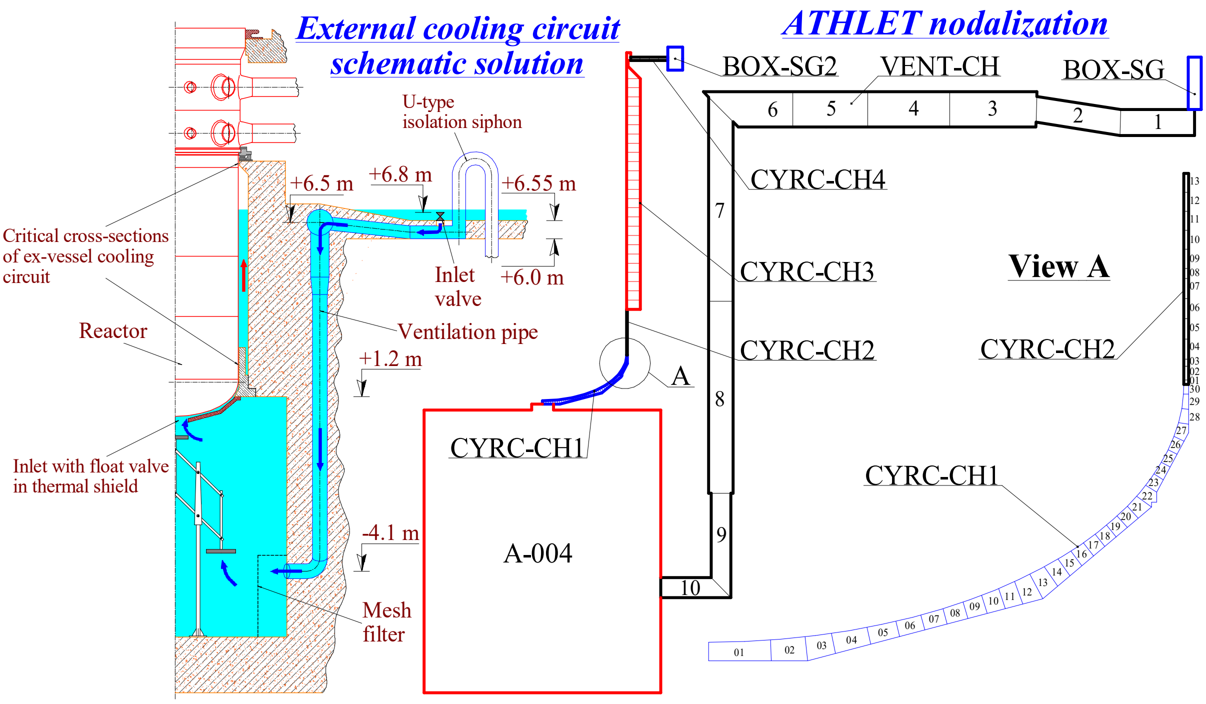

FIG. 1. Schematic solution of the external cooling circuit of the VVER-440 reactor vessel and nodalization for ATHLET model.

3. DEFINITION OF THE PROBLEM AND NUMERICAL ANALYSIS

First of all, operability of the above described circuit depends of availability of sufficient water inventory. The analysis of geometry of reactor compartment and the vessel of Rivne NPP Units 1, 2 has been done. The minimal free volume, which should be filled to obtain stable natural circulation is 863 m^3, is much less than a volume of water on trays in BCT (1463 m^3).

Next step is the analysis of the key aspects of ex-vessel cooling, as follows:

— Departure from nucleate boiling ratio at the reactor bottom, and
— Sufficiency of coolant flow rate and heat sink in the circuit for reliable reactor cooling.

The following method was used to provide a reliable solution of the safety problem:

- Selection of severe accident scenarios, based on results for Paks and severe accident assessment previously done for Rivne NPP. The “Large break LOCA 2*D_n 500 with total station blackout” is the most challenging scenario leading to the Leading to the earliest melting and formation of corium.
- Calculation of in-vessel phase of accident with MELCOR computer code and comprehensive analytical model of NSSS of Rivne-1 to define transient progression and timing of main events, such as start of melting and further formation of corium. Time of corium formation gives us a total decay heat power, which should have been removed by ex-vessel cooling. The results of the calculations demonstrate that the maximum power of the decay heat is about 9 MW, which correlates well with the values obtained for the Paks NPP [4], 9.63 MW for the 107% of the initial reactor power.
- Definition of heat flux from corium to the coolant through the reactor vessel during in-vessel phase of severe accident was done using AIDA module of ATHLET-CD 3.1A. The AIDA model considers a segregated melt pool consisting of a lower oxide layer with decay power and an upper metal melt layer enclosed in a hemispherical reactor bottom. Moreover, considering the impossibility of modeling the semi-elliptical bottom of the WWER-440 reactor, the configuration of the melt pool with the minimum area equivalent to the hemispherical reactor bottom is simulated. The power of decay heat was taken from MELCOR calculations. At the outside surface of the reactor bottom model, constant third-type boundary conditions are set, simulating the heat transfer conditions to the coolant of the ex-vessel natural cooling circuit (the heat transfer coefficient for bubble boiling is 10,000 W/m^2/K, the coolant temperature of the main flow is 120°C). This approach is similar to the assessment of ex-vessel cooling of the VVER 440 reactor of the Paks NPP [5], [5].
- Considering the significant uncertainties of calculations of key physical phenomena under severe accident the special uncertainty and sensitivity analysis was performed using SUSA 4.0 [6]. 34 initial parameters of AIDA model with most potential influence on calculation results were selected, including thermal and mechanical properties of materials of corium, type of mathematical models, ratio of zirconium oxidation and others). Sample ranking and preliminary analysis of the distribution of hypotheses, considering recommendations of SUSA developer [7] and [8], were done. 100 input decks for AIDA calculations were made by SUSA for combinations of parameters. SUSAs random number generator is based on the Monte Carlo method and the boundaries and the nature of the distribution of hypotheses. 100 calculations are sufficient to obtain a qualitative and reliable sensitivity analysis based on Wilks distribution [9] for a two-sided statistically tolerant interval with an upper and lower limit of 95% / 95%. To analyze the ranking of variable model parameters, the Spearman rank correlation coefficient [10] is used in SUSA. In this case, the application of the GRS method, included in the SUSA software, has a number of significant advantages in comparison with the methods of stochastic approximation in the analysis of uncertainty and sensitivity [11].
- The ex-vessel cooling system is simulated with AHLET code to check the operability natural circulation and sufficiency of flow rate for cooling.

4. RESULTS OF ANALYSIS

The maximum heat loading on RPV from melt pool is the most significant result of the analysis due to it defines heat transfer conditions on outer surface of RPV and, primarily, it could lead to the thermal failure of RPV. Thus, the analysis of sensitivity and uncertainty of maximum heat flux on the outer surface of RPV was done with SUSA.

Results of the analysis as a probability density function are presented on Fig. 2. It is easy to see; that maximum density of heat flux is less than 750÷800 kW/m^{2}is in good agreement with results of a probabilistic study of the melt behavior for the Loviisa NPP [12]. The most conservative values of the critical heat flux are obtained in experimental study on SBLB test facility [13]. Therefore, it could be concluded, that departure from nucleate boiling ratio resulting from comparison of the mentioned above values of the heat flux and critical heat flux, is sufficient.

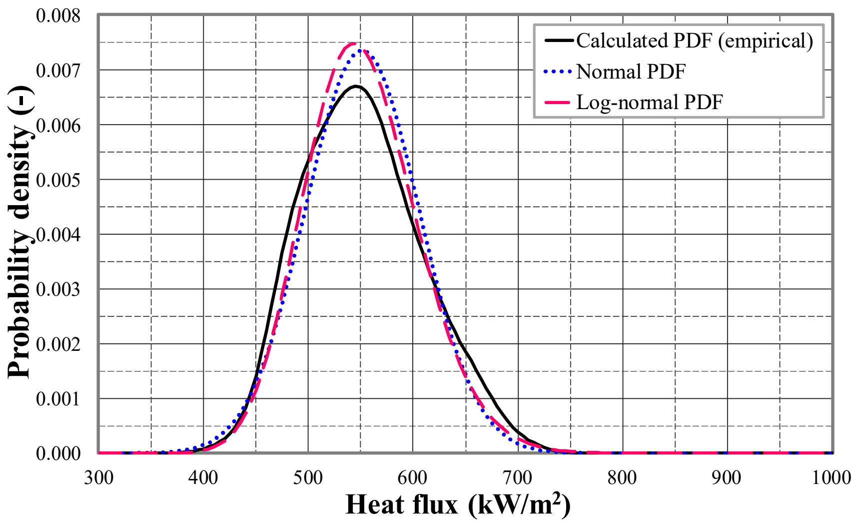

FIG. 2. Probability density function of maximum value of heat flux density at outside surface of RPV.

A sufficiency of heat transferred from outside surface by natural circulation of ex-vessel cooling was checked with ATHLET 3.1A code. Fig. 1 presents the nodalization scheme of ATHLET model of ex-vessel cooling circuit. Model consist of following TFOs: BOX-SG – time dependent volume simulating boundary conditions for coolant inlet; VENT-CH – ventilation duct, located in concrete wall of reactor cavity; A-004 – lower part of reactor cavity compartment with float valve; CYRC-CH1,2,3 – the heated part in the gap between the reactor vessel and the biological shield; CYRC-CH4 – a gaps between blocks of biological shield surrounding inlet and outlet of primary system pipes; BOX-SG2 – time dependent volume simulating boundary conditions for coolant outlet..

The distribution of heat flux density at the reactor pressure vessel (RPV obtained by results of uncertainty and sensitivity analysis for the maximum thermal load in the peak heat flux region is set as a boundary condition for AHLET calculations. The influence of a gap between RPV and thermal shield is checked by variant calculations with design gap and a gap reduced due to the possible thermal and mechanical deformation of RPV.

In both cases, the results of calculations confirm the reliability of heat removal and integrity of RPV. As expected, the behavior of a two-phase flow in the gap with variable geometry and coolant intensive boiling has an oscillating nature. This is intensified by the considerable unevenness of the heat load in the heated area, which leads to the irregularity of the vapor fraction in the flow, and, taking into account the interfacial friction, the outlet of generated vapor through the narrowest point of the circulation circuit (see Fig. 1). The maximum temperatures of RPV surface were obtained in points with the narrowest gap (see Fig. 1), due to a short-term transition to the dispersed flow of the two-phase coolant. Nevertheless, this does not lead to a noticeable increase in the surface temperature of RPV in those points.

As it shown on Fig. 3, the process is a quasistationary, the average value of heat transfer coefficient in area with average heat load (nodes 30÷35) is set at approximate value of 9000÷12000 W/m^2/K. Maximum values of heat transfer coefficient were obtained for maximum heat flux areas (nodes 40÷50). The average value of coolant flow rate for two-phase circulations is equals to ≈150 kg/s (see, Fig. 3).

5. CONCLUSIONS AND OPEN QUESTIONS

Analysis of IVR strategy implementation on WWER in EU countries the detailed consideration of the structural and schematic features of the reactor facilities for the RNPP-1,2 were done, and applicability of EU approach was confirmed. Numerical analysis of ex-vessel cooling system shows the sufficient DNBR and coolant flow is enough for reliable cooling of the reactor.

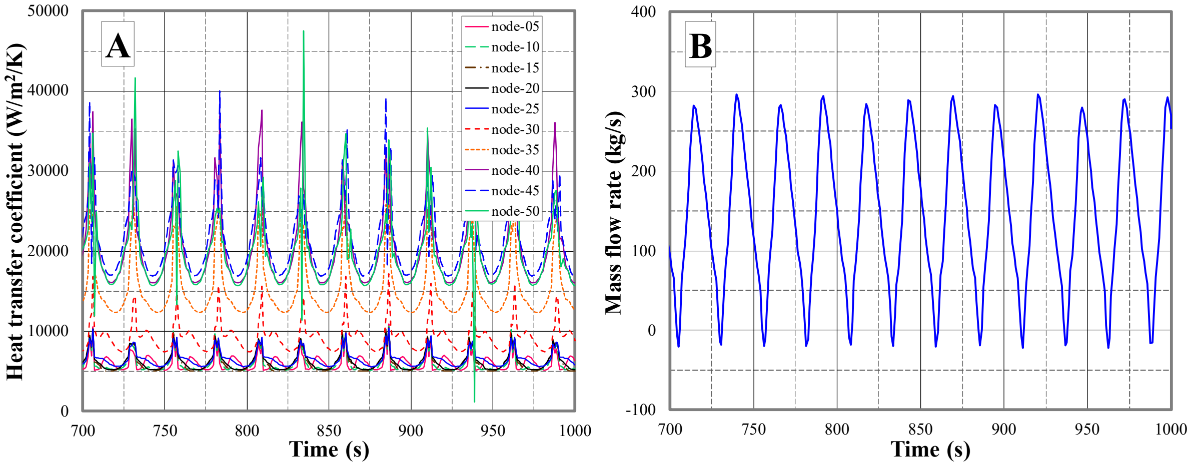

FIG. 3. ATHLET calculation results: A – heat transfer coefficient at the outer surface of the reactor vessel in a section with an average heat load; B – coolant mass flow rate in elevating circuit.

Open questions are:

Is a subcriticality of melted fuel will be maintained in case of relocation of fragments and possible in-vessel cooling (reflooding)?

What could happen in case of inadvertent operation of ex-vessel cooling system under normal operation of reactor facility? Would it lead to the brittle fracture of the reactor vessel?

ACKNOWLEDGEMENTS

We thank our colleagues from GRS: Thorsten Hollands for assistance with ATHLET-CD methodology and Martina Kloos for SUSA methodology and training.

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INITIAL STARTUP PROCEDURE INVESTIGATION OF A BWR-TYPE SMALL MODULAR REACTOR

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Abstract

Purdue Novel Modular Reactor (NMR) is a BWR-type small modular reactor design with an electric output of 50 MWe that relies on natural circulation driven flows for both normal operation and accident management. NMR and other similar natural circulation driven systems are often subject to flow instabilities, which could affect system control and safety under low-power and low-pressure conditions due to small driving force compared to forced circulation. Therefore, flow instabilities during the initial startup transients of NMR were experimentally investigated in a natural circulation test facility with an operating pressure limit of 1.0 MPa. The overall height of this test facility is seven meter, close to that of the prototype design. This makes it appropriate to study flow instability phenomena (mainly flashing instability occurred in the chimney due to reduced hydrostatic head) during reactor startup transients. In our study, four initial startup procedures with different power ramp rates were experimentally investigated with the aim of eliminating the flow instabilities observed from tests using a normal startup procedure scaled from that for SBWR-600. Specifically, a very slow startup transient test and two pressurized startup transient tests were performed. The experimental results indicated that both startup procedures were applicable to the initial startup of NMR. However, the pressurized startup procedure might be preferred due to its shorter operating hours required.

1. INTRODUCTION

Natural circulation flow driven systems have been widely adopted in the design of GEN III+ nuclear reactors and other passive safety systems due to the passive nature, simplicity, and economics for eliminating circulation pumps. Purdue Novel Modular Reactor (NMR), a BWR-type small modular reactor (SMR) design with an electric output of 50 MWe (NMR-50) [1], has been designed based on natural circulation driven flow and mature BWR technologies. The NMR-50 conceptual design characterizes both enhanced safety and improved economics by adopting a double-layer passive safety system, fewer penetrations on Reactor Pressure Vessel (RPV), and a modular design with a smaller containment and reduced-size reactor buildings. In addition, due to its low core power density, NMR-50 is designed with a target of ten-year fuel cycle length [2, 3].

However, natural circulation driven systems are often subject to flow instabilities due to small driving force compared to forced circulation systems [4]. The natural circulation flow instabilities have been summarized by the author to include density wave oscillations, flashing instability due to reduced hydrostatic head, and condensation induced flow instability, also namely, Geysering instability [5, 6]. Specially, the flashing instability could occur at low-power and low-pressure conditions, such as reactor startup transients or accident conditions for natural circulation driven systems [7-10]. Flow instability could cause mechanical vibration, affect system

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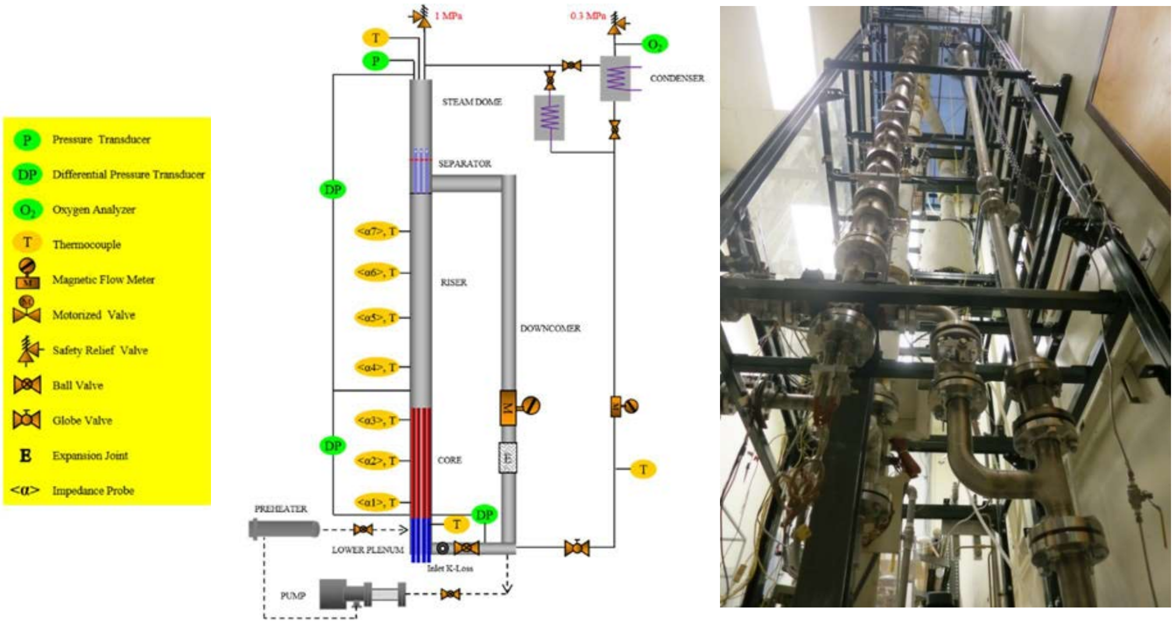
control, and even deteriorate reactor safety performance. Thus, it is necessary to eliminate the flow instabilities for natural circulation driven systems at low-power and low-pressure operating conditions.

To investigate the flow instabilities during NMR-50 startup transients, the initial startup procedures should be simulated experimentally in a natural circulation test facility. The initial startup procedure of the NMR-50 is also scaled from that of General Electric's SBWR-600 [11]. The SBWR-600 startup transients are initiated from a vacuum pressure of 55 kPa, which is followed by a deaeration process to remove non-condensable gases [12]. The reactor coolant is heated by fission power through withdrawing the control rods to establish the reactor criticality. Similar startup procedures were applied in the test facility to experimentally investigate the flow instabilities during the startup transients of NMR-50. In the experiments, the coolant was heated by four electric heater rods with a startup power curve scaled from that of NMR-50. It is worth mentioning that only the thermal-hydraulic flow instabilities, i.e., without considering the void reactivity feedback, are presented in this paper, since the effects of the void reactivity feedback on the flow instability was not very significant, especially for the flashing instability [8].

This paper presents the experimental investigation of flow instabilities during the startup transients of NMR-50. Section 2 describes a scaled natural circulation test facility and its instrumentation. Four startup transients are compared with different startup procedures in Section 3. Key conclusions from this research are summarized in Section 4.

2. TEST FACILITY AND INSTRUMENTATION

Figure 1 shows the schematic and a photo of the test facility before it was insulated. This test facility was scaled down from the NMR-50 design based on a three-level scaling method developed by Ishii et al. [13] for the design of Purdue University Multi-dimensional Integral Test Assembly (PUMA). The detailed scaling analyses were performed for developing a scientific design and then an engineering design (considering scaling distortions due to space and material limitations) of this test facility [7].



(a) *(b)*

FIG. 1. Test facility and instrumentation: (a) Schematic of the test facility [7] and (b) Test facility without insulation.

This test facility includes the following major components to simulate the prototype reactor NMR-50: (a) a lower plenum housing the unheated section of four electric heater rods and inlet flow distribution pipes, (b) a four-flow-channel core section, (c) a chimney or riser section, (d) a steam dome section with a simplified separator, and (e) a downcomer section. The total height of this test facility is about seven meters and close to that of the prototype. The test facility aims to study the flashing instability caused by the reduced hydrostatic head in the chimney section without elevation scaling distortions. Important design parameters are summarized in Table 1 [7].

This test facility was equipped with instruments measuring various thermal hydraulic parameters to investigate flow instabilities. Table 2 lists the instruments installed and their measuring uncertainties. The evolution of the void fraction measured by impedance void probes made in-house, in the core and chimney sections was analyzed. In addition, the inlet single-phase mass flow rate measured by the magnetic flow meter is another key parameter to identify any presence of flow instabilities. LabVIEW was used during the experiments for measurements, control, and data acquisition.

TABLE 1. DESIGN PARAMETERS OF THE TEST FACILITY [7]

Item	Specification	Value
Test facility	Design pressure (MPa)	1.0
	Total height (mm)	7,000
	Wall materials	304 Stainless Steel
	Wall thickness (mm)	3.05
	Top of heated section (mm)	1,950
	Top of chimney (mm)	5,260
Heated section	Active heated length (mm)	1,130
	Hydraulic diameter (mm)	23.0
	Inner pipe diameter (mm)	82.8
Chimney or riser	Inner pipe diameter (mm)	82.8
Downcomer	Inner pipe diameter (mm)	54.8

TABLE 2. TEST FACILITY INSTRUMENTATION

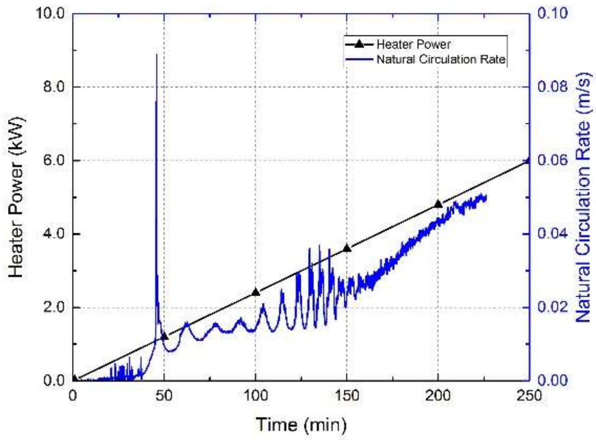
Item	Uncertainty	Manufacturer
T-type thermocouple	Greater of 1°C or 0.75%	Omega Engineering
Differential pressure transducer	±0.0375% of the span	Honeywell
Absolute pressure transducer	±0.0375% of the span	Honeywell
Magnetic flow meter	±0.5% of the reading	Honeywell
SCR power controller	±0.5% of the power output	WATLOW
Impedance void probe	0.5% in absolute value	Homemade

3. RESULTS

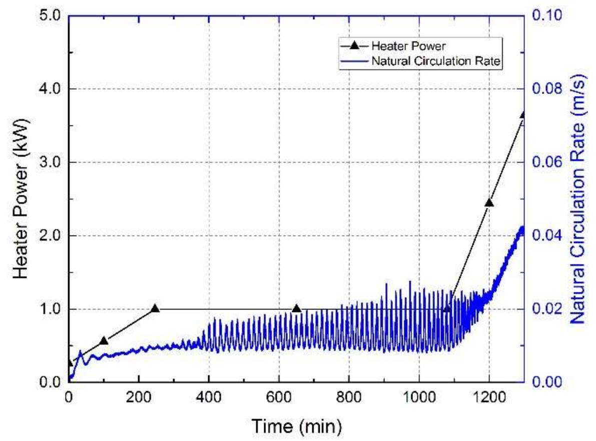
Startup transients were performed in this natural circulation test facility using different initial startup procedures. First, the normal startup procedure with a scaled power-ramp rate from NMR-50 was applied to the startup transients from a vacuum pressure of 50 kPa. System pressure, temperature, void fraction, and natural circulation rate at the core inlet were measured during the startup transients. Flow instabilities, including density wave oscillations and flashing instability, were observed in this normal startup transients [7]. The electric power of the heaters was the only parameter that can be controlled during the startup transients. To eliminate the flow instabilities, two additional startup procedures were then investigated and analyzed. The second startup test involved a very slow startup procedure with variable power ramp rates. Next, a third startup procedure, a pressurized startup procedure with a venting process, was also tested with either a slow or fast power ramp rate. Due to the length limit of the paper, only the natural circulation rate profiles are presented here to show the flow instability phenomena and the results of four different startup procedures. Figures 2 shows the heater power curves and natural circulation rate profiles during these four startup transients. All startup transients were performed under low pressure conditions due to the pressure limit of the test facility.

For the normal startup transients as shown in Fig. 2(a), the entire transients lasted about 220 minutes with a linear heating power ramp rate of 0.4 W/s. The natural circulation rate profile can be roughly divided into four phases: (a) single-phase natural circulation (0-50 minutes), (b) periodic flashing in chimney (50-100 minutes), (c) core net vapor generation (100-160 minutes), and (d) two-phase natural circulation (after 160 minutes). Regarding flow instability mechanisms, flashing instability was dominant between 50 and 160 minutes. When the system pressure reached 500 kPa, flashing instability was stabilized, but small density wave oscillations could be observed during the two-phase natural circulation phase. The first velocity peak in Fig. 2(a) is caused by a sudden coolant flashing near the top of the chimney when the loop natural circulation rate was low in the single-phase natural circulation phase. Therefore, flashing instability was the main flow instability mechanism that needed to be addressed during the low-pressure startup transients.

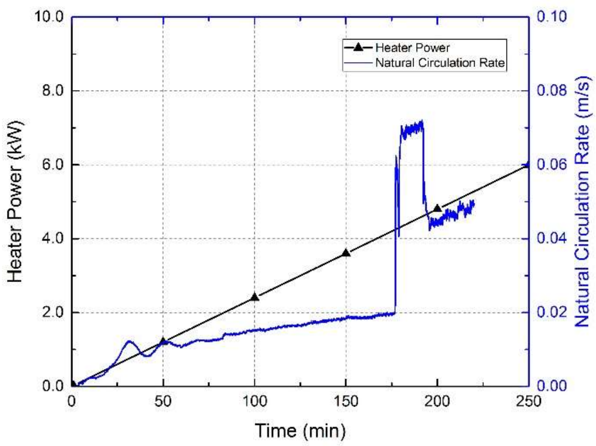
During the normal startup transients starting from a vacuum pressure, the heater power level is the only boundary condition that can be changed. If the coolant is heated in the core section at a very small rate, the vertical fluid temperature distribution in the core region could be more uniform and the flashing instability due to reduced hydrostatic head in the chimney section could be stabilized. Therefore, a very slow power curve shown in Fig. 2(b) was applied to very slow startup transients in the test facility.



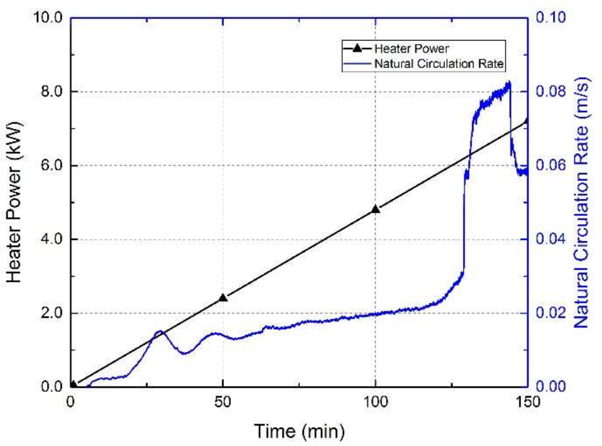
(a) Normal startup transients.



(b) Very slow startup transients.



(c) Pressurized slow startup transients.



(d) Pressurized fast startup transients.

FIG. 2. Heater power curves and natural circulation rate profiles during startup transients:
(a) Normal startup transients [7], (b) Very slow startup transients, (c) Pressurized slow startup transients, and
(d) Pressurized fast startup transients.

This variable power curve can be divided into three regions. In the first four hours, the power was increased at a power ramp rate of 0.05 W/s to initiate the single-phase natural circulation and to eliminate the flashing phenomenon. In the next phase, the power was kept at 1.0 kW for about 14 hours to avoid big vertical temperature gradient. In the third phase, the power was increased to a rate of 0.2 W/s. As can be seen, the single-phase natural circulation rate was stabilized largely by using this very slow heat up rate. The big peak flow shown in the normal startup transients was not observed. The net vapor generation in the heated section started at about 350 minutes and ended at 1200 minutes as shown in Fig. 2(b). During this period, there were continuous oscillations with a period of about 15 minutes due to very small power density. After 1200 minutes, the flow oscillations reduced and the system was dominated by the two-phase natural circulation. Although the flow oscillations still existed, the magnitude of the flow oscillations was much smaller than that in the normal startup transients. Since only four flow channels exist in our test facility, the flow oscillations can be further stabilized in an actual reactor due to multiple flow channels.

By analyzing the flow instability phenomena, most flow instabilities observed disappeared when the system pressure reached 500 kPa [7, 8]. Therefore, two pressurized startup procedures were investigated with the goal of eliminating the flow instabilities or reducing them significantly by setting the initial pressure to 300 kPa using nitrogen gas. Figures 2(c) and 2(d) show the power curves at two different heating power ramp rates. Once the venting process was initiated at 500 kPa, the natural circulation rate increased from 2 to 7 cm/s at 180 minutes in Fig. 2(c), and to 8 cm/s at 120 minutes in Fig. 2(d), respectively. After the venting process, the natural circulation rates reduced to a typical two-phase flow velocity. From the natural circulation rate profiles, the pressurized startup procedure has a positive effects on eliminating the flashing instability. In addition, the comparison between two pressurized startup transients shows that a fast power ramp rate is preferred to eliminate the flashing instability.

4. CONCLUSIONS

Natural circulation driven systems are subject to flow instabilities at low-pressure and low-power conditions. This paper investigates the flow instabilities, especially the flashing instability, for a BWR-type SMR, i.e., NMR-50, during startup transients. A natural circulation test facility was designed and constructed to perform startup transient tests. Different initial startup procedures were investigated in four startup transient tests. Flashing instability was the main instability mechanism encountered during the normal startup transients. Very slow startup procedures can be used to stabilize the flow oscillations in real natural circulation driven systems based on the current conservative assessments in this test facility. However, the very slow startup procedures requires much longer operation time for reactor operators. Another option is to adopt the pressurized startup procedures to eliminate the flashing instability by filling the RPV with non-condensable gas at the beginning. But this method requires an additional venting process to remove the non-condensable gas when the system pressure reaches about 500 kPa according to various experimental investigations.

ACKNOWLEDGEMENTS

This material is based upon work supported under a Department of Energy Nuclear Energy University Program.

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INVESTIGATION OF PERFORMANCE OF PASSIVE HEAT REMOVAL SYSTEM FOR ADVANCED NUCLEAR POWER REACTORS UNDER SEVERE CONDITIONS

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Abstract

The work presented in this paper discusses issues related to post shut-down cooling of advanced nuclear power reactors by totally relaying on passive processes. The paper investigates the Passive Residual Heat Removal System (PRHRS) capability for removing all the core decay heat without any AC power source or even any operator intervention to provide a significant grace time, during which essentially passive natural processes can provide adequate cooling, before any operator action is needed. It also investigates the compliance of the design of PRHRS with IAEA requirements (SSR2/1 and GS-R-4). The reactor which was investigated is the VVER-1000 AES-92 model which is the proposed design to be constructed in Jordan. A deterministic safety code will be used to model the essential components of the primary, secondary loop and passive safety systems with corresponding control systems for AES-92 model to analytically address the performance of the PRHRS taking into the severe environmental conditions of the NPP site in Jordan.

1. INTRODUCTION

The reliable removal of decay heat, after shutdown following some fault or external event, is one of the major challenges in designing adequately safe nuclear plant. Fukushima accident has highlighted the desirability of making the plant robust against events that have led to the loss of active, powered systems to perform this function. In particular, there is a great benefit in engineering the plant such that it provides a significant grace time, during which essentially passive natural processes can provide adequate cooling, before engineered, powered intervention is needed, if, indeed, it ever does become necessary.

In the VVER-1000 AES-92 design, Passive safety features are widely implemented to deal with design basis and design extension conditions. One of these passive safety features is the PRHRS which removes the residual heat in the event of non-availability of active heat removal systems on account of complete loss of normal and emergency AC Power supply (station blackout condition).

A review against IAEA requirements (SSR-2/1 and GSR, Part4) for passive safety systems has been addressed. This paper also discusses the deterministic safety analysis of PRHRS using RELAP 5 MOD3.2 and its behaviour under SBO.

2. DESICRIPTION OF PRHRS, AES-92 DESIGN

The system consists of four independent closed circuits with natural circulation (4x33%); each one is connected to the secondary side of each steam generator in the primary loop. Each circuit has three air-steam heat exchangers modules which are located outside the containment building, pipelines of steam supply and removal of the condensate; air ducts that supply and discharge air, air flaps and control or adjusting devices. (See Fig. 1 which represents one of these Circuits).

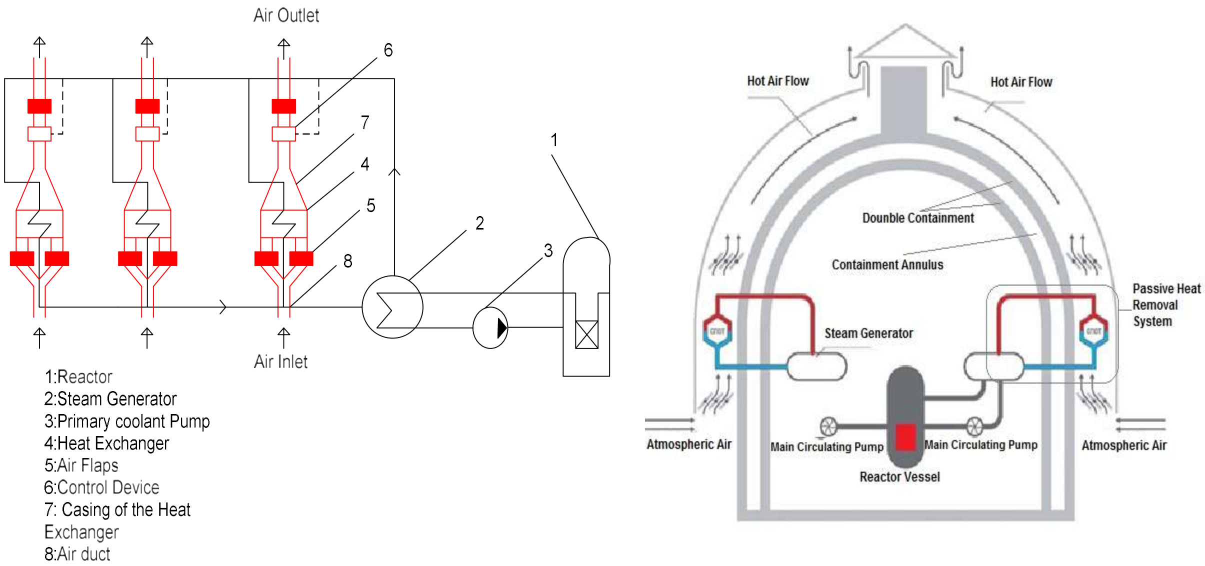

FIG. 1. Left Fig. 1-A: PRHRS scheme, Right Fig. 1-B: PRHRS layout.

As shown in Fig. 1-A, the steam taken from the secondary side of steam generators is condensed in the finned tube heat exchangers of these circuits, the condensed liquid returns back through down-coming pipelines to the secondary side of the steam generators by gravity and establish a continuous natural circulation; the cooling media for the PRHRS heat exchangers is atmospheric air that flows through the air ducts. During Station black out accident, the air flaps, located at air path upstream and downstream of the PRHRS heat exchangers open due to release of holding electromagnets which allow the heat to be transferred to the atmospheric air. Inside each heat exchanger module, the heat is transferred form the steam to the air which enters the draught parts of the air ducts that end with common collector-deflector at the top of the containment building as shown in Fig. 1-B [1].

PRHRS has a special controller with two operating modes: SG pressure maintaining mode and cool down mode; SG pressure maintaining mode is conducted by the passive drive of the controller, which is driven by the pressure from SG to maintain the hot standby parameters of the reactor plant; cool down mode is conducted by the active drive which is powered from DC batteries to cool down the reactor plant.

3. REVIEW OF PHRS AGAINST IAEA REQUIREMENTS

According to IAEA Glossary, it is not absolutely clear what is considered as a passive system. However, in the IAEA glossary the definition of passive component is “A component whose functioning does not depend on external input, such as actuation, mechanical movement or supply of power.” [2]. Referring to EUR glossary, the passive safety system is defined as “A system which is essentially self-contained or self-supported, which relies on natural forces, such as gravity or natural circulation, or stored energy, such as batteries, rotating inertia, and compressed fluids, or energy inherent to the system itself for its motive power, and check valves and non-cycling powered valves (which may change state to perform their intended functions but do not require a subsequent change of state nor continuous availability of power to maintain their intended functions).” [3].

There are not much specific requirements or guidance on passive systems in the IAEA Safety Standards, however the safety requirements for active safety systems are vast and comprehensive, some of these requirements should be apply on passive safety systems such as single failure criterion. After Fukushima the requirements for ultimate heat sink and associated heat transfer chain to the ultimate heat sink have been significantly strengthened. The table below shows the compliance of PRHRS with IAEA requirements.

TABLE 1. COMPLIANCE OF PRHRS DESIGN WITH IAEA REQUIREMENTS [4, 5, 6]

Requirement	Paragraph	Review
10: Assessment of engineering aspects	**4.29:** Where innovative improvements beyond practices current practices have been incorporated in the design…	• Design of the PRHRS was tested on a dedicated facility at OKB GP. References to experimental documentation (design, scaling, and experiments) are provided. The tests were performed in summer and winter conditions. Detailed analyses of the PRHRS were performed using GAMBIT code. The code was extensively validated using experimental data. [6] • Performance of the PRHRS (with steam generators) during beyond design accidents was addressed using analytical methods that were validated using integral experiments conducted at FEI's GE2M-SG facility including effects of non-condensable gases. • Performance of the PRHRS under different wind conditions was addressed by performing wind tunnel experiments on a model of the reactor building.
15: Deterministic and probabilistic approaches	**4.53:** Deterministic and probabilistic approaches have been shown to complement one another …	• DSA and PSA methods were used to assess safety of the AES-92 design. • In the design provide the highest impact on the PSA results. It was shown that elimination from the design of only PRHRS would result in an increase in CD frequency by a significant number. • In this paper the deterministic analysis is elaborated.
16: Postulated initiating events	**5.11:** Where prompt and reliable action is necessary in response to postulated initiating events….	• The automatic response of active safety systems is complemented by an “automatic” start of the passive safety systems of the hydraulic accumulators of stage I, II, III and the PRHRS. • Other passive safety systems specifically designed to cope with BDBA need manual operator action for initiation (passive filtering system of the inter-containment space (KLM), melt trap (1 JKM)).
17: Internal and external hazards	**5.20:** The design shall be such as to ensure that items important to safety are capable of withstanding the effect of external events…	• The AES-92 design provides an effective protection against all types of external initiators that have limited potential or low probability of damaging the reactor building and reactor unit items. This can be explained by implementation of PRHRS, which does not require any active system operation and can be automatically actuated in black-out conditions. • The most important is the PRHRS that is designed to operate under extreme environmental conditions. For example due to its passive actuation and layout (four independent natural circulation loops connected to the steam generators secondary sides) this system can function in event of external fires with very low failure probability.
25: Single failure criterion	**5.39:** Spurious action shall be considered to be one mode of failure…	• The PRHRS has twelve heat exchangers, where nine are sufficient. Hence, there are two redundancies. Concern: in case of a SBLOCA, however, one loop out of four is inoperable, which means that three HXs are lost. If we assume single failure in one channel, only three heat exchangers (HXs) are operable, which may not be sufficient. Similarly, in case of SGTR, three HXs are in the affected loop - their operation under this condition has not been analyzed. In case of a single failure in one channel, there are only three HXs operable.
32: Design for optimal	**5.58:** The design shall be such to promote the	Measures have been taken in the design to promote the success of operator actions and to prevent errors occurring. The main measures are as follows: • Passive systems have been incorporated to carry out the safety functions that need to be performed after the occurrence of an initiating

Requirement	Paragraph	Review
operator performance	success of operator actions….	event. These systems do not require operator actions to actuate them or for their operation; • the active safety systems and actions are actuated automatically; and • Interlocks are incorporated into the design to prevent the operators carrying out incorrect actions.
32: Design for optimal operator performance	**5.59:** The need for intervention shall be minimum…	The design aim is that no operator actions should be required in the first 30 minutes following an initiating event. This is achieved by the incorporation of passive systems and automatic initiation of active systems.
53: Heat transfer to ultimate heat sink	**No paragraph**	For DBAs and also for DECs without loss of primary circuit integrity, heat removal to the ultimate heat sink is provided for an indefinite time. If the active systems are available, then the service water acts as the ultimate heat sink, to which heat is transferred through the intermediate circuit. If the active systems are unavailable, then the outside atmosphere acts as the ultimate heat sink, to which heat is transferred via heat exchangers of the PRHRS.
61: Protection system	**6.32:** The protection system shall be designed with fail-safe…	In addition to the inherent scram feature in case of power loss, passive safety systems have functional capability to cool the plant even in the case of complete loss of power.
68: Emergency power supply	**6.44:** The combined means to provide emergency power…	One of the set of the batteries of each train powers required power for thee monitoring of the operation of PRHRS during 24 hours (with possibility of 72 hours) without recharging.

4. MODELLING OF PHRS FOR AES-92 DESIGN

In this paper, analysis of station black out (SBO) accident scenario with complete loss of all AC power sources including emergency diesel generators is performed. The main purpose of the analysis is to investigate the performance of the PRHRS at a high temperature of Jordan NPP site and to check if the safety acceptance criteria are met. In order to model and study the natural circulation characteristics of PRHRS, RELAP5 MOD3.2 is used to develop the models of the Primary Loop, Secondary loop and the PRHRS.

4.1. RELAP Mod 3.2

RELAP5 3.2 is best-estimate system analysis code designed especially for the modeling of a wide range of operational, emergency and transitional processes that may occur in systems equipped with nuclear or electric heat sources and using as the main heat transfer medium with water in one- or two-phase state. [7]

Basic characteristics RELAP5 computer code are as follow:

- A one-dimensional model of two-phase flow, including:
  - 2 mass conservation equations;
  - 2 energy equation;
  - two equations of conservation of momentum.
- One-dimensional neutron kinetics model.
- Hydrodynamic modeling system using the following basic components:
  - pipe;
  - simple volume ("single volume");
  - boundary condition ("time-dependent volume" and "time-dependent junction");
  - simple connection ("single junction");
  - "branch" (branching flow models);

- pump;
- valve (various types).

4.2. Boundary conditions

In normal operation of the VVER-1000 AES-92 design, the initial and boundary conditions are given in the Table 2:

TABLE 2. INITIAL AND BOUNDARY CONDITIONS [1]

Parameter	Value
Thermal Power, MW	3000
Coolant temperature at the reactor inlet, C°	291.0
Coolant Pressure at the reactor outlet, MPa	15.7
Coolant flow rate through the reactor, m^3/h	86000
Pressurizer level, m	8.17
Collapsed level in SG, m	2.356
Steam pressure at the SG outlet, MPa	6.27
Feed water Temperature, C°	220.0
Air Ambient Temperature, C°	41

In the analysis, the following assumptions were considered:

- All PRHRS channels are available with delay of 30s in order to be connected from the moment of losing all AC Power sources.
- A delay of 30 s until the PRHRS channels reach the full power capacity.
- The PRHRS power characteristics (taken from the experimental data) are assumed at ambient air temperature of 41 C°.

4.3. RELAP nodalization

Nodalization of the primary loop is shown in Fig. 2-A. The core model is presented by 4 channels and one bypass channel. One of the four channels represents the hot channel with radial peak factor of 1.81 for the hottest fuel rode. Elevation between SG outlet and heat exchangers module is about 15 m. The air-steam heat exchangers modules for each PRHRS channel are modelled as one active heat structure (see Fig. 2-B) based on PRHRS power characteristics which are taken from the experimental data at ambient air temperature of 41 C°.

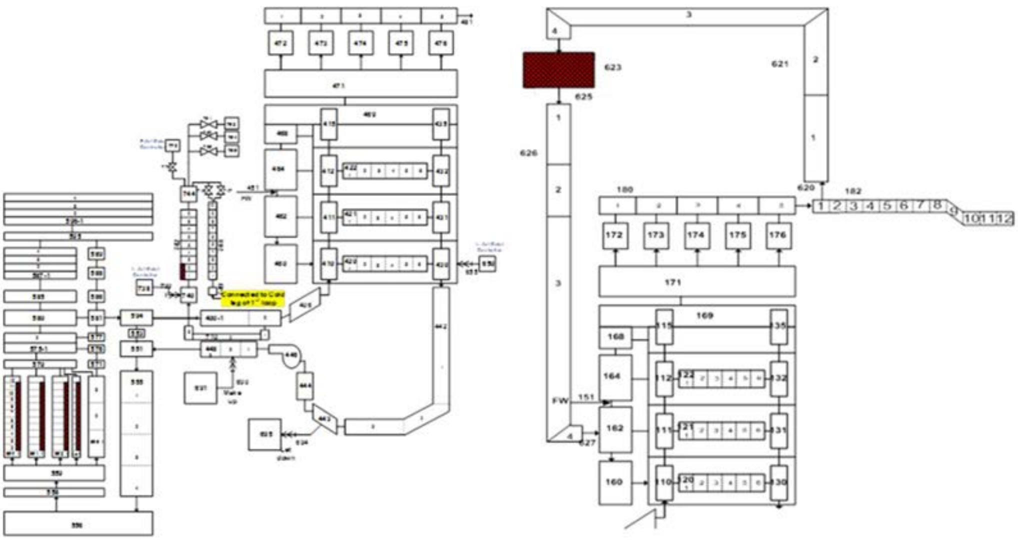

FIG. 2. Left Fig. 2-A: Primary loop, Right Fig. 2-B: PRHRS.

4.4. Results and discussion

The analysis was made to see if the PRHR is capable of removing the decay and residual heat after reactor scram, and to ensure the cooling is maintained, and to ensure that all parameters are within the safety margins. The main parameters analysed are the pressure, the fuel temperature, and cladding temperature.

The SBO scenario is analyzed with assuming no leakages in the primary circuit, when the SBO occurs the reactor is immediately scramed, and the power dropped from 100 % to 4% in one second as shown in Fig. 3.

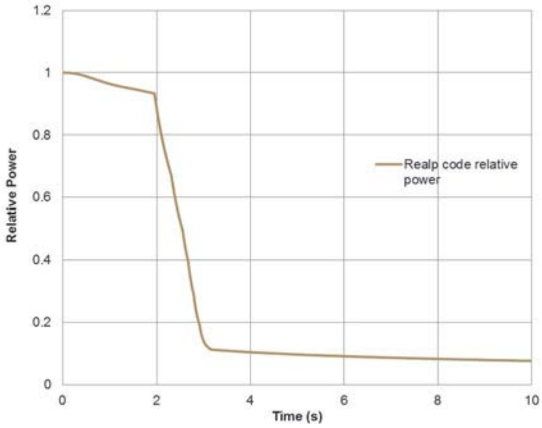

FIG. 3. Relative power.

As result of loss of all AC power supply, the turbine generator stop valve closes leading to increase the pressure in the secondary circuit as shown in Fig. 4 and then causing the steam dump to atmosphere (BRU-A) to open which immediately decrease the pressure in the secondary circuit and release the residual heat to atmosphere. By the end of this process the PRHRS reaches its nominal capacity which leads to further decrease in the pressure of secondary circuit until the BRU-valve closes. Then the pressure is stabilised due to the PRHRS is working properly in the SG pressure maintaining mode.

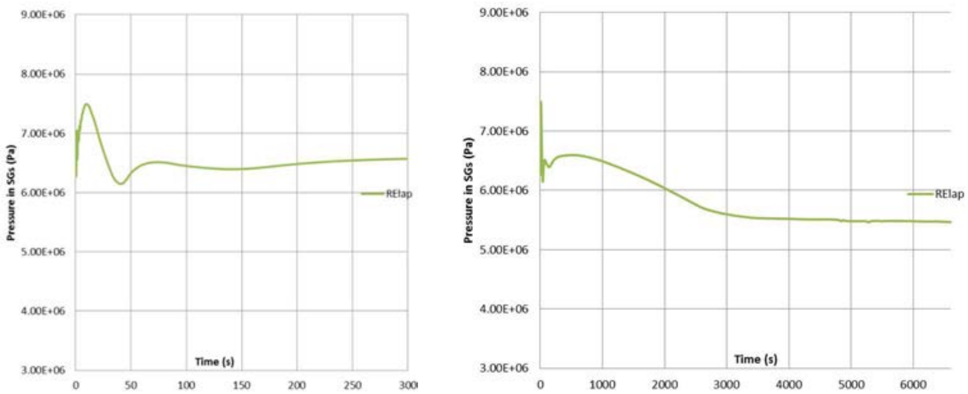

FIG. 4. Secondary pressure.

Fig. 5-A shows the pressure of the primary circuit, it is important to maintain the pressure under the safety limit in order to ensure that the integrity of primary circuit is maintained. As shown in Fig. 5-A there will be a sharp decrease in the pressure due to SCRAM, then it will increase due to the decay heat. But the pressure will eventually stabilize due to heat removal through the PRHRS. AS shown in Fig. 5-B the pressuriser level is proportional to the primary pressure.

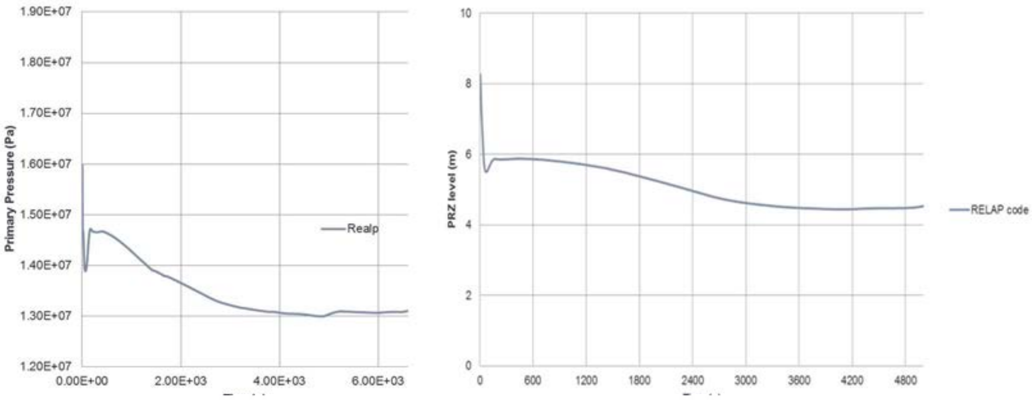

FIG. 5. 5-A: primary pressure, 5-B: pressurizer level.

The fuel cladding temperature is also important to analyse in order to avoid any fuel cladding failure and to make sure that radioactivity is confined and no release of fission products. Fig. 6 shows the behaviour of cladding temperature after scram, when the reactor is scracmed, there is no power produced by the fuel and all the power in the reactor is due to the decay heat, the cladding temperature will eventually decrease since no heat is generated from fission and there is sufficient cooling to cool the cladding, the cladding temperature is stable for a period of time when the pump is still working due to its inertia, however when the pump stops the cladding temperature slightly increase, and then stabilize as a result of PRHRS operation.

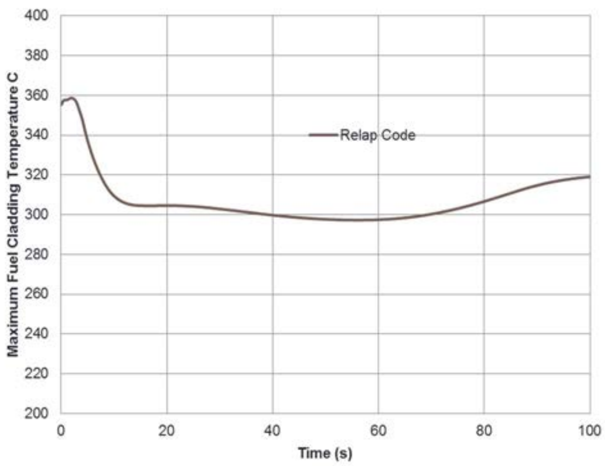

FIG. 6. Fuel cladding temperature.

5. CONCLUSION

In this paper the performance of PRHRS was investigated quantitatively by means of thermal-hydraulics software RELAP MOD 3.2 and qualitatively by reviewing its design against the IAEA safety requirements. Though there are no clear requirements for passive systems in the IAEA safety requirements but the design of PRHRS in AES-92 design complies with the general safety requirements and those specific to the active safety requirements.

Deterministic Analysis of PRHRS using RELAP Mod 3.2 gives the full picture of the behaviour of such system, PRHRS works properly for unlimited period of time as long as the primary circuit is leak tight, it is capable to cool the core and remove the residual heat in case of SBO with no available AC power sources, it is also capable to maintain all safety parameters within the design limits and safety margins.

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ASSESSMENTS OF A PASSIVE HEAT REMOVAL SYSTEM IN AN INTEGRAL REACTOR

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Abstract

This work is oriented to develop knowledge for supporting engineering tasks regarding the design of a "Passive Heat Removal System" (PHRS) of an integral-type reactor (CAREM-25), and analyze the plant evolution in case of Loss of Heat Sink (LOHS). The mission of this system is to reduce the pressure on the primary system by means of decay heat removal, by condensing steam from the primary system in condensers immersed in containment pools. The condensate returns to the reactor vessel establishing a two-phase natural circulation circuit. A model of PHRS is developed using RELAP5 code. Some studies are made with the system isolated, by setting boundary conditions, to characterize its phenomenology under nominal operation. Different configurations of tubes are analyzed. Then, this model is integrated to the plant model, with the aim to analyze the performance of this system. Two stages are observed: initially, when the PHRS is triggered the primary system pressure decreases sharply because the steam is condensed while the liquid remains subcooled. Then, the primary system continues with saturated depressurization until to reach the grace period (36 hours). Finally, the PHRS demand ensures cooling and decay heat removal without requiring power energy or human actions during 36 hours, avoiding the safety valves demand. Pressure is decreased allowing the action of normal shutdown systems.

1. INTRODUCTION

The introduction of passive safety systems in advanced reactor designs is one of the topics that pose challenges to the safety demonstration. The present work consists in analyze the response of CAREM reactor with the actuation of the Passive Residual Heat Removal System (PHRS) in case of Loss of Heat Sink (LOHS).

The PHRS condensers are located in a pool filled with cold water inside the containment building. Due to systems assembly and lay-out issues, it is of particular interest the study of different configurations of tubes. Some particular studies arise regarding the impact on of this construction issues in the system capacity.

2. REACTOR DESCRIPTION

A brief description of the reactor and the passive safety system that are related with the present work is presented [1] [2]. CAREM is an Argentine project to achieve the development, design and construction of an innovative, simple and small Nuclear Power Plant (NPP).

CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the design and also contributes to a high safety level. Some of the high level design characteristics are: integrated primary cooling system, self-pressurized primary system, safety systems relying on passive features, primary cooling by natural circulation.

2.1. Primary System and its main characteristics

CAREM NPP design is based on a light water integral reactor. The whole primary system, core, steam generators, primary coolant, steam dome and control rod mechanism, are contained inside a single pressure vessel. For low power modules (below 150 MWe), the flow rate in the reactor primary systems is achieved by natural circulation (Fig. 1). Several innovative features can be observed within the primary system, besides their passive safety systems, such as the self-pressurization (result of the liquid-vapour equilibrium), reactivity control without boron in the coolant, and as mentioned above, primary system coolant driven by natural circulation induced by the location of the steam generators above the core.

2.2. Passive Residual Heat Removal System

CAREM safety systems are based on passive features and must guarantee no need of active actions to mitigate events during the grace period (36 hours). They are duplicated to fulfill the redundancy criteria.

The Passive Residual Heat Removal System (PHRS) has been designed to reduce the pressure on the primary system and to remove the decay heat in case of Loss of Heat Sink (LOHS) with unavailability of the active safety systems. It is a simple system (Fig. 2) that operates condensing steam from the primary system in condensers. The condensers consist of an arrangement of parallel horizontal U tubes between two common headers which are located in a pool filled with cold water inside the containment building.

The top header is connected to the steam dome, while the lower header to the reactor vessel in a position below the reactor water level. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore in stand-by mode the tube bundles are filled with condensate. Each PHRS contains two modules with two condensers tubes.

In case of primary system overpressure, the reactor protection system demands automatic opening of the outlet valves. The water drains from the tubes and steam from primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensate is returned to the reactor vessel establishing a Natural Circulation (NC) circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water in condenser pools is conducted to the containment suppression pool, where condenses. This way the decay heat is stored within the containment during the grace period [3].

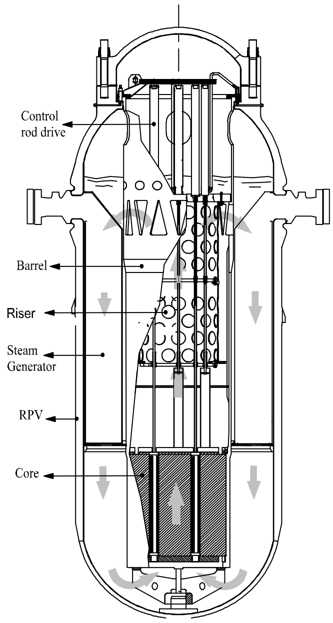

FIG. 1. CAREM primary system.

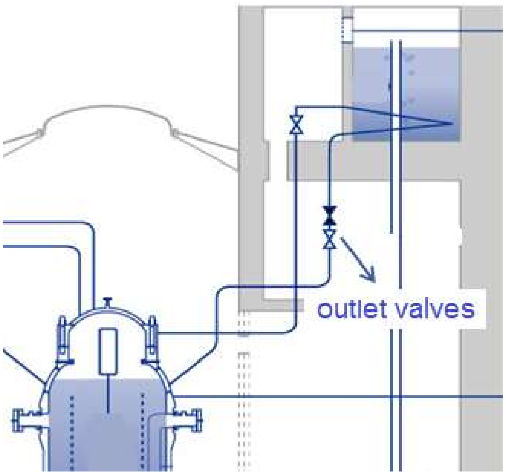

FIG. 2. PHRS layout.

3. MODEL DEVELOPMENT

In order to perform the calculations with RELAP5 a one-dimensional nodalization of CAREM reactor has been developed. The model involves the Primary System and the PHRS.

First some studies are made with the system isolated by setting boundary conditions and then the PHRS model is integrated to the reactor model to analyze its performance during a LOHS.

A brief description of reactor and PHRS models are presented. The models are developed within an integral platform of data nodalization and management, implemented for RELAP code for simulation of reactor transients. The system includes geometry and process input data, calculation of related parameters, data processing for nodalization development, and automatic generation of the input file for the RELAP code, supporting quality assurance and minimizing input errors. The primary circuit nodalization has been set-up dividing it into the most relevant components: RPV dome, steam generators (SG), down comer, riser, core and lower plenum. The SG secondary side is also included in the model, the rest of the secondary system and process systems have been modelled as boundary conditions.

The PHRS nodalization includes the following system components: steam line (steam line piping and inlet header), condensers (heat structures have been taken into account), condensate line (return line piping, outlet header and valves) and system pool.

4. PHRS EFFICIENCY CURVE

As it was described before, the PHRS condensers are located in a pool inside the containment building. In order to analyze important aspects for the assembly and location of equipment, some studies are made with the system isolated by setting boundary conditions. In this case, the removal power in function of primary pressure is analysed for several configurations, considering different pipe lengths.

Design conditions are modelled by means of boundary conditions: primary system pressure of 12.25 MPa and a pool temperature of 100°C (saturation state). The PHRS removal power is 2MW per redundancy (1MW per module) at primary pressure of 12.25 MPa. To analyze the efficiency of the equipment against possible changes in the effective length of the heat exchangers, three simulations are performed: DL (design length), DL/2 (half of design length) and x*DL/2 (including x as a length compensation factor) (Fig. 3).

Heat removal capacity must be maintained in order to fulfil design requirements. Thus, a reduction in the length of the tubes is compensated increasing the quantity of tubes: the case corresponding to the design length represents two tubes while the others, four tubes.

The heat transfer coefficients in condensers are observed in Fig. 4. The heat transfer mode recorded along the tube is filmwise condensation for the primary side and subcooled nucleate boiling for pool side. The tube coefficient (an equivalent coefficient due to the conduction process) is the one that will govern in greater proportion the heat exchange. So power will depend mainly on heat transfer area. However, the condensation coefficient is reduced with length, as result of liquid formation that make more difficult the condensation process. This would decrease heat transfer capacity in the last section of the tubes.

Due to aspects related to the equipment assembly, it is required to analyse the performance of the system with shorter tubes; in particular, it is proposed to use tubes with a half-length, and at the same time to double the number of tubes. This configuration was studied, and it is observed that it is not enough to remove a power of 1MW at the operating pressure of 12.25 MPa. This happens because, despite the condensation coefficient is higher near the inlet region, the overall condensation coefficient decreases for shorter tubes (Fig. 4). This is because the flow mass decreases, as long it has to split in higher number of tubes. So the tubes length must be increased to compensate this phenomenon, to x*DL/2.

When the primary pressure decreases, the efficiency of the equipment is reduced mainly due to the decrease in the associated primary side temperature. Nevertheless, the decrease is (slightly) higher for shorter tubes, mainly due to the increased sensitivity on condensation coefficients (Fig. 3).

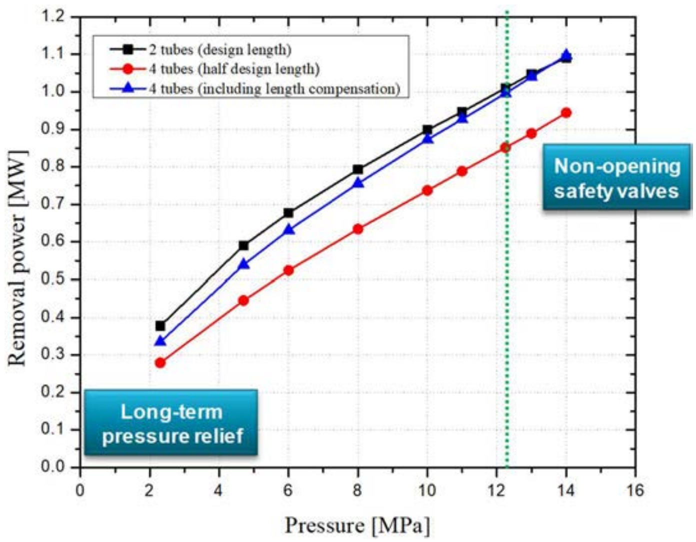

FIG. 3. PHRS efficiency curves for different tube lengths (results per module).

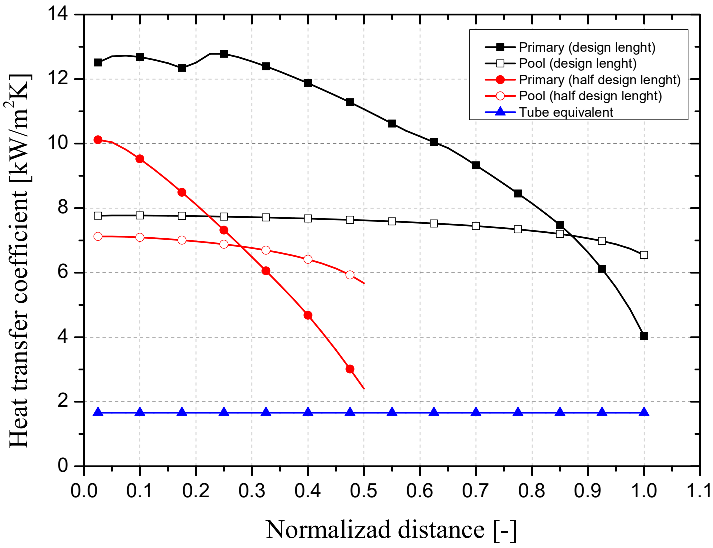

FIG. 4. Heat transfer coefficients for different tube lengths.

5. LOSS OF HEAT SINK SIMULATION

5.1. Modeling considerations

Previous to the transient code run, a steady-state has been performed with reactor operating at 100% power, maintaining the reactor pressure at the nominal value (12.25 MPa).

On the other hand, a series of considerations and hypotheses are postulated to simulate the event:

Failure of all control and regulation systems of the plant, in addition to process systems, is considered. Therefore, it must be demonstrated that the safety functions are effectively fulfilled by the Safety Systems.

Activation of only one redundancy of PHRS.

5.2. Event description

The transient to be analyzed is a LOHS: in this case the initiating event proposed is an abrupt loss of SG feedwater. For a better description and to establish a criterion for transient analysis four phases has been identified: Phases 1-2 are stages taking place before PHRS demand and Phases 3-4 after system actuation (Fig. 5).

Phase 1: Due to the LOHS, down-comer temperature increases. This causes, on the one hand, a reduction in the buoyant force (decreases in the water density), decreasing the flow. On the other hand, coolant expansion occurs increasing the liquid level, the steam in the dome is compressed increasing the pressure system.

First Shutdown System (FSS) is triggered when temperature at the exit of the SG reaches its correspondent set-point (high primary pressure signal is almost simultaneously). As a consequence of the power reduction, heat flux within the core is reduced decreasing subcooled boiling and core void generation stops. Thus, a temporal pressure decrease is verified.

Phase 2: Once finalized this brief depressurization stage, due to there is no power removal, the temperature goes on increasing in the down-comer and in the whole circuit driving again to the primary circuit coolant expansion with the subsequent primary system pressure increase. The rate of increased pressure is reduced compared with Phase 1, because the generated power drops to decay values.

During this pressurization phase, the primary system remains in a sub-cooled condition.

Phase 3: When the system pressure reaches the PHRS set-point, the system is activated by opening the outlet line valves. As it was explained before, the sub cooled water drains from the tubes and steam from primary system enters to the tube bundles condensing on them.

Immediately after the PHRS actuation a sharp depressurization phase takes place. This behaviour is mainly due to imbalance coming from the steam condensation in the dome without liquid boiling in the primary system. This condition is also accompanied, in a less extent, by the sub-cooled water (at pool temperature) coming into the RPV from condensers tubes immediately after the PHRS actuation.

Phase 4: Once the primary system reaches again its saturation condition the sharp depressurization phase ends and from this moment on, pressure begins to be ruled by saturation conditions, steam reposition into the steam dome (generated in the core) and steam condensation in the PHRS. Pressure continues decreasing steadily until the end of the grace period, allowing the action of normal shutdown cooling systems.

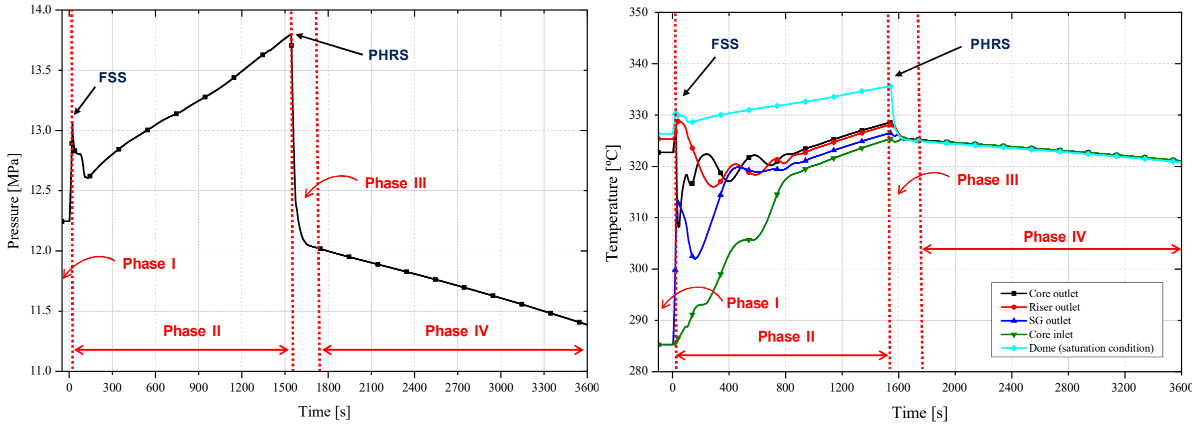

FIG. 5. Short-term pressure and primary system temperatures evolution for LOHS.

6. CONCLUSIONS

The reactor cooling function in safe conditions is guaranteed with PHRS demand, fulfilling with safety margins and design safety limits. PHRS ensures cooling and decay heat removal without requiring energy or human actions during 36 hours (grace period), avoiding the safety valves opening. Moreover, this occurs by using only one redundancy. Pressure is decreased allowing the action of normal shutdown cooling systems.

The efficiency of the equipment varies with effective length of the heat exchangers: a reduction of the tubes length, together with an increase in number of tubes, causes a local flow mass decrease so lower removal power. Therefore it is observed that, as the pressure decreases, the efficiency decreases comparatively to a greater extent for shorter tube configurations, due to the greater weight in the variations of the heat transfer coefficient. Nevertheless, as long as the transferred power is ruled mainly due to the conduction phenomena through the tube structure, a relatively low impact is observed.

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SAFETY ASSESSMENT OF ADVANCED REACTOR DESIGNS

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Abstract

The safety of the nuclear power plant is an important issue. The role of the safety systems is to ensure the three fundamental safety functions: control of reactivity, removal of heat from the core, confinement of radioactive materials and limitation of accidental releases. The safety has been evolved in the 50-years of nuclear energy history. Regardless the NPP's size, the efficiency of any safety system depends on the accident against which it is designed to protect. Therefore, to cover a wide range of postulated design-basis-accident, the suppliers have incorporated innovative safety systems that do not depend on the availability of electric power, called passive safety system. In fact, this innovation has marked the generation III reactors because it would be a genuine solution under many accident scenarios. However, passive safety systems are not infallible because it cannot withstand all accidents, eventually, the beyond-design-accidents. In the other hand, the use of advanced models, codes and standards ensure an accurate validation of these safety systems. In light of the introduction of the new concept of small modular reactors, we notice that safety is more and more enhanced due to the design characteristics allowing more efficient control of the radioactive materials and improving the capacity of the NPP to withstand the most severe accident without exceeding certain radiological release limits.

1. INTRODUCTION

The safety of the nuclear power plant is an important issue. The role of the safety systems is to ensure the three fundamental safety functions:

- Control of reactivity;
- Removal of heat from the core;
- Confinement of radioactive materials and limitation of accidental release.

This applies to the three radioactive sources in the NPP which are the core, the fuel storage pool and the radioactive waste treatment systems.

The approach is based on multi-barrier and defence-in-depth concept. It contains several levels of protection, including successive barriers preventing the release of radioactive material to the environment.

2. SAFETY ASSESSMENT OF ADVANCED REACTOR DESIGNS

According to the IAEA the safety function is "The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazard" [1].

2.1. Defence-in-depth and barriers

2.1.1. Defence-in-depth

According to INSAG-12, "the defence in depth concept is a fundamental safety principle that provides an overall strategy for safety measures and features of nuclear power plants. It helps to implement the three fundamental safety functions" [2].

The failure of one or more of these safety functions can lead to an unacceptable risk for the facility, the environment, and a risk of an excessive or unacceptable exposure of workers or populations to radiation.

Consequently, two levels of risk are to be considered:

- The potential risks, consisting of the risks incurred in the absence of any protective measures.
- The residual risks that remains in spite of the implementation of the protective measures.

The table below presents the five levels of defence-in-depth-concept.

TABLE 1. LEVELS OF DEFENCE-IN-DEPTH [3]

Level of defence in depth	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Conservative design and high quality in construction and operation
Level 3	Control of accident within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe of plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management.
Level 5	Control of severe of plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management.

2.1.2. Barriers

In order to prevent the release of radioactive materials to the environment, designers include many physical barriers. For the pressurized water reactors (PWR), there are three physical barriers as we can see below Fig.1:

— The fuel cladding.
— The primary reactor coolant system used to transfer the heat from the fuel to the steam generator.
— The containment building that plays an important role in the nuclear power plant. It is designed to palliate the radioactive release when both the primary and secondary barriers are lost. It ensures the protection of the nuclear power plant in the case of the most severe accident. Finally, it ensures the radiation shielding.

The role of the physical barriers is to contain the radioactive materials and protect workers, the population and the environment. The number of the barriers varies from design to design (heavy water reactors or boiled water reactors) depending on designing philosophy.

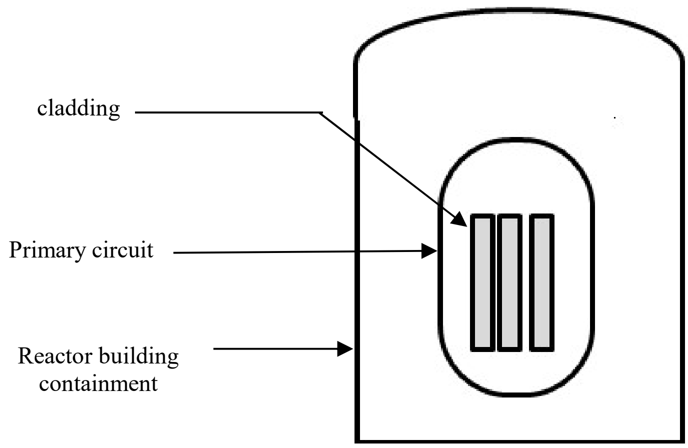

FIG. 1. The three containment barriers in the PWR.

2.1.3. Redundancy and diversity

First of all, safety isn't a target, it is a condition. The role of safety system is to ensure the safe operation of a nuclear power plant. If a design-basis accident occurs (level 3 of defence-in-depth), safety systems are required to ensure the core cooling and to prevent the release of radioactive materials.

Three important concepts should be taken into account in the design of the nuclear power plant:

— Redundancy that means to duplicate safety systems and equipment, as practical as possible, to cope with the single failure criterion and to promote one safety function.
— Diversification that aims to prevent common mode failure. Such failures can be due to a design error, a faulty operating manoeuver or an external hazard (fire, flood, etc.). So using diverse systems and mechanism can help to overcome such events.
— Physical separation of systems fulfilling the same function, in order to reduce the occurrence frequency of simultaneous failure due to an external hazard.

The physical separation of redundant systems and their diversification should be planned at design stage.

Definitely, the redundancy and the diversification enhance the safety of the nuclear power plant. But we cannot duplicate the safety systems indefinitely, because it would have an impact on the cost of the nuclear power plant. Optimally, safety should be ensured by several and different systems without compromising the cost of the facility.

2.2. Safety approach

The safety assessment isn't an obvious task. In fact, the efficiency of any safety system depends on the accident against which it is designed to protect. And we cannot imagine all the accident scenarios. But, we are aware of the most hazardous event: the radioactivity release in case of core melt accident.

There are two different approaches to deal with the safety assessment:

- *Deterministic approach* which consists on proving that a design is tolerant to a limited number of hazards defined as design-basis accidents, taking into account the most pessimistic scenarios. The analysis is based on credible accidents or not that can lead to the workers exposure to radiation or to environmental contamination. The deterministic approach is based on the defence-in-depth concept. First of all, it is about ensuring the normal operation of the power plant and predict failure mode. Then, if a fault condition occurs, immediate measures may be taken to avoid severe accident. Finally, it is about considering the worst-case scenario and how to limit the radiological risk to population and radioactive release.

- *Probabilistic approach* which is a risk assessment methodology based on a systematic survey of accident scenarios, used to assess the risks associated with nuclear facilities in terms of frequency of dreaded events and their consequences. The probabilistic analysis techniques were emphasized just after three miles accident o supplement conventional safety assessment procedures for nuclear power plants, and to facilitate the determination of acceptable safety levels for nuclear facilities. In the probabilistic approach, a dreaded situation is defined (the core damage, large radioactive release), and an event-tree is established using a list of initiating events (with Annual frequencies) as complete as possible.

Both approaches are intended to maintain the risks associated with nuclear facilities at acceptable levels. The difference between them is methodological. Each approach has its limitations so optimally, the design should be based on the deterministic approach, using the concept of defence-in-depth, and it must be supplemented by the probabilistic approach to analyse external events.

2.3. Safety systems

The three major accidents in the history of civil nuclear power (three miles island, Chernobyl, Fukushima) highlighted the importance of nuclear safety.

Active safety systems that rely on operator commands can operate reliably in normal circumstances and in the case of postulates accidents (design basis accidents). But, Fukushima accident emphasize the importance of considering the beyond design basis accident and calls into question the active safety systems limitation.

Passive safety systems, which do not depend on the availability of electric power, would be a genius solution under many accident scenarios, but not all. No safety system is one hundred percent infallible even if it is passive.

As a result, a reliable design should include both active and passive safety systems. This would add to cost but enhance the global safety of the design. The best compromise will probably be an optimal combination of the two types of safety systems.

3. SMALL MODULAR REACTORS

Safety is more and more enhanced in light of the introduction of Small Modular Reactors (SMR). This new design presents many characteristics allowing more efficient control of the radioactive materials and improving the capacity of the NPP to withstand the most severe accident without exceeding certain radiological release limits.

According to the IAEA, Small Modular Reactors refer to reactors with an electrical power less than 300 MWe. Lowering the reactor power leads to the reduction of the source term. And this can make the control of the radioactive inventory in the reactor easier.

In the other hand, the small size of these reactors enables the natural convection cooling and then keeping the core at safe temperature in the case of serious accident.

Some vendors propose the underground design that has some safety advantages, for example the optimization of the nuclear site. In fact, in some situation such as earthquake, underground reactor can be a genius solution. However, the design can increase the risk in other situations such as flooding. And the emergency intervention can be more difficult.

Other vendors present integral pressurized water reactor (iPWR). This innovative design is original because it is compact. In fact the reactor pressure vessel contains the main components, notably the steam generators, the control drive mechanisms and the pressurizer.

The fact to incorporate the main structures into one space requires novel layout and this can be a double-edged-sword. Indeed, the potential safety benefit is that the design eliminates the possibility of loss of coolant accident by eliminating the large diameter piping outside the reactor vessel. However, the steam generator is subject to intense radioactivity. As a consequence, some issues, such as corrosion, are affected. This innovative design is like a black box which can make inspection and maintenance more difficult.

Finally, the challenge for the Small Modular Reactor is to reduce size and cost without compromising safety and security.

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UTILIZATION OF ANALOGY EXPERIMENTAL METHOD FOR THE PRELIMINARY VERIFICATION OF THE SAFETY SYSTEM WITH HIGHLY BUOYANT ER EXTRME TEST CONDITION

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Abstract

Safeties of nuclear systems are to be verified during the design and licensing periods. The experiments are to be performed at severe conditions and require expensive facilities. The paper introduces an experimental methodology based on the analogy between heat and mass transfer. The basic idea is to replace heat transfer systems with corresponding mass transfer systems. Here the copper-sulfate electroplating system is used. This experimental method can achieve high buoyancy with small facilities due to the reduction of cupric ions at the cathode induces large density decrease of the fluid. A few published researches of the authors are summarized as the examples of the applications, simulating highly buoyant flow conditions of the In-Vessel Retention and External Reactor Vessel Cooling (IVR-ERVC), Reactor Cavity Cooling System (RCCS), and even a Critical Heat Flux (CHF) problems. Utilization of this analogy experimental method will ease the preliminary verification of the nuclear safety system.

1. INTRODUCTION

Design of the nuclear safety systems are one of the most important issues in the nuclear industry. The system performances must be verified during design and licensing periods. Especially for the thermal hydraulics point of view, basic phenomena of the safety system are to be confirmed through the preliminary works. However, it is hard to conduct the safety verification of the large facilities, due to its extreme condition in case of the severe accident or large size requirement for natural convection tests.

This paper suggests the experimental methodology for a highly buoyant or extreme test condition with compact test rig based on the analogy between heat and mass transfer. The copper-sulfate electroplating system is employed as the mass transfer system. The cupric ions are transferred from anode to cathode when the potential is applied and they work as the heat transferred in the heat transfer system and are easily and accurately measured by the electric current measurement. The reduction of the cupric ion concentration causes buoyant force near the cathode. Using this system, large buoyance can be established with relatively small test rig. This paper summarizes some papers of the authors to introduce the methodology regarding the In-Vessel Retention via External Reactor Vessel Cooling (IVR-ERVC) situation, Reactor Cavity Cooling System (RCCS) and Critical Heat Flux (CHF) phenomenon.

2. ANALOGY BETWEEN HEAT AND MASS TRANSFER

Heat and mass transfer systems are analogous as the mathematical models describing the two phenomena are of the same form [1]. Therefore, although the two phenomena are different in nature, these can be treated mathematically as the same. Table 1 summarizes the governing parameters for heat and mass transfer systems. Here we suggested is copper sulfate-sulfuric acid (CuSO4-H2SO4) electroplating system. This system has been well established in several decades by the pioneers. In this system, the ionic concentration gradient causes buoyancy force which achieves high Rayleigh number with relatively short length scale. Moreover, the measurements are achieved by the electric circuit which is superb in terms of the accuracy relatively to the other measurement tools. Thus, applying this methodology for the nuclear power plant systems, the high buoyant-induced systems can be simulated readily using compact test rig. This technique was developed by several researchers [2-5], and the methodology is well-established [6-13]. Compared with temperature measurements, this technique is attractive as it provides a simple, low-cost, accurate method through the electric current measurement [14, 15].

TABLE 1. ANALOGY OF DIMENSIONLESS NUMBERS OF HEAT AND MASS TRANSFER SYSTEM

Heat transfer	Mass transfer
$P_r = \frac{v}{\alpha}$	$S_c = \frac{v}{D_m}$
$N_u = \frac{h_h H}{k}$	$S_h = \frac{h_m H}{D_m}$
$R_a = \frac{g\beta\Delta T H^3}{\alpha v}$	$R_a = \frac{gH^3}{D_m v}\frac{\Delta\rho}{\rho}$

Fig. 1 depicted schematic diagram of the CuSO4-H2SO4 electroplating system. In this system, cupric ions are reduced at the cathode surface. The resulting decrease in the cupric ion concentration in the solution induces a buoyant force. Thus, the cathode is used as the hot wall. To minimize the electric migration of the cupric ions, sulfuric acid is added as the supporting electrolyte to increase the electric conductivity of the solution.

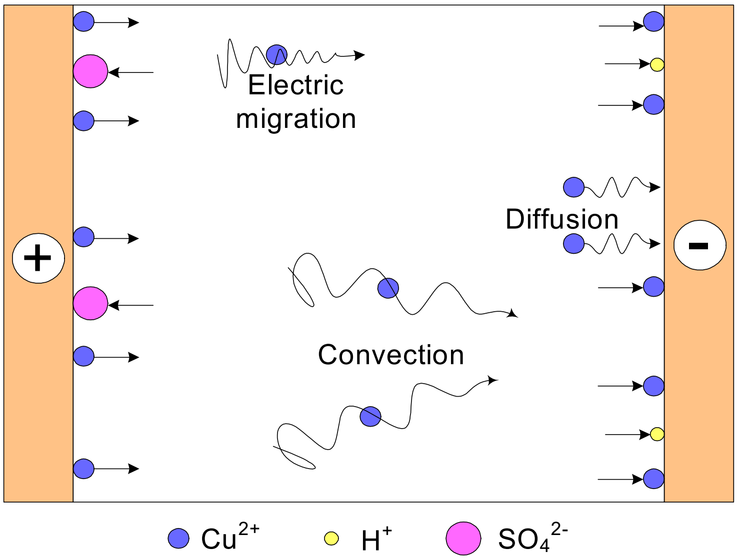
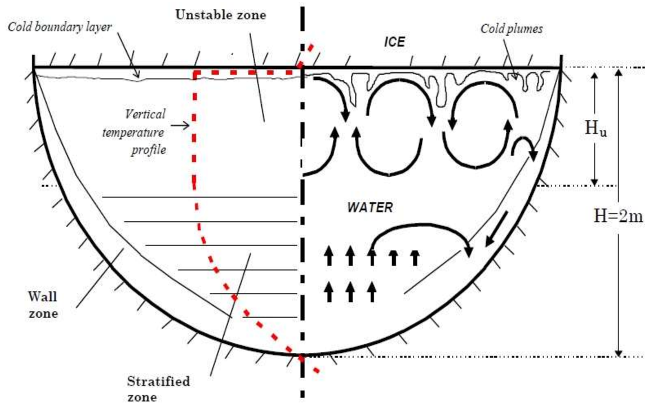


FIG. 1. Ionic transfer in copper sulfate-sulfuric acid electroplating system.

3. ACHIEVEMENTS

3.1. In-Vessel Retention via External Reactor Vessel Cooling (IVR-ERVC) situation

There are two severe accident strategies to mitigate release of the radioactive material from reactor vessels. One assumes the reactor vessel break and introduces core catcher underneath to cooling the core melts. The other assumes it maintains its integrity by the natural convection of the core melt and outside cooling (IVR-ERVC). The authors conducted the experiment for IVR-ERVC situation using Mass Transfer Experimental Rig for Oxide Pool (MassTER-OP) [13, 16]. Relocation of the nuclear fuels in the reactor vessel forms a hemispherical oxide pool under the metallic layer. This oxide pool was simulated by CuSO4-H2SO4 electroplating system. The modified Rayleigh number of over 1015 was established with all of 0.167 m test rig. Fig. 2 shows the phenomenon and the test rigs to simulate it. Tests were performed at the inverted arrangement to use the cathode for the measurements. And piece-wise electrodes enable to measure local average values and then phenomenological analysis can be performed. MassTER-OP simulate with 2D and 3D test rigs.



(a) General flow patterns of the oxide pool [17].

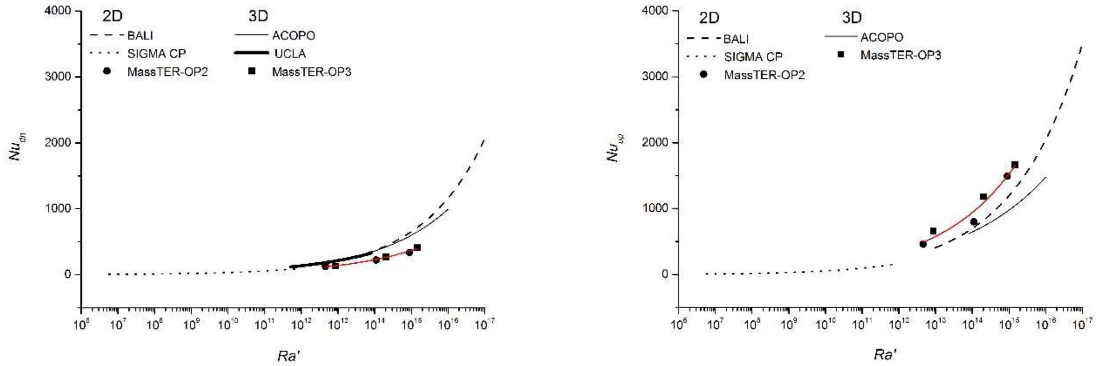
H=0.042 m H=0.1 m H=0.167 m

H=0.042 m H=0.1 m H=0.167 m

(b) 2D and 3D test rigs for simulate oxide pool.

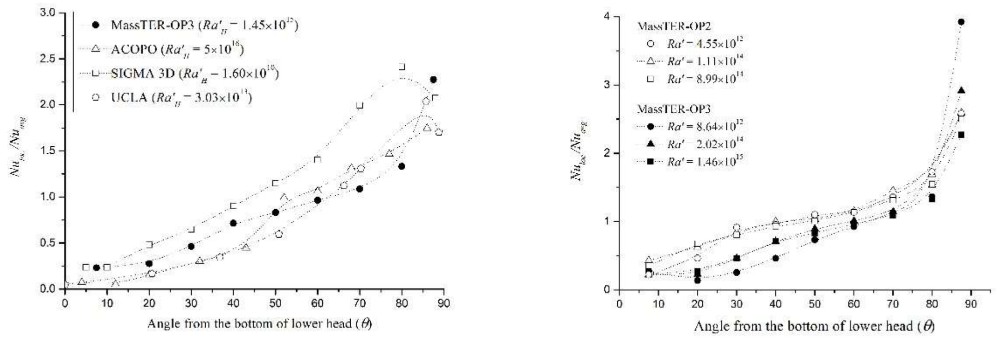
FIG. 2. General phenomena of the oxide pool and test rigs.

Nusselt numbers (Nu) at the curved surface of the reactor vessel and boundary between oxide pool and metallic layer (Top plate) were measured and compared with existing heat transfer results both 2D and 3D experiments. As shown in Fig. 3, the lower Nu's were measured at the curved surface and the higher Nu's were measured at the top plate both 2D and 3D cases. These results were caused by difference of the Prandtl number (Pr). The Pr of MassTER-OP was 2,014 and the others were less than 10. Therefore, thermal boundary layers of the MassTER-OP were much thinner and thus central rising flow contains more fresh bulk fluid than those of lower Pr cases. As a result, the higher Pr, the higher Nu of the top plate were measured. Fig. 4 shows the angular Nu ratios compared with existing studies. The general trends were similar but small discrepancies occurred at the uppermost region. The authors insisted that difference of Pr and geometrical feature influenced the results. Further details of these results can be found in the corresponding papers [13, 16].



(a) Mean Nu of the curved surface. *(b) Mean Nu of the top plate.*

FIG. 3. Comparison of mean Nu's for the curved surface and top plate.

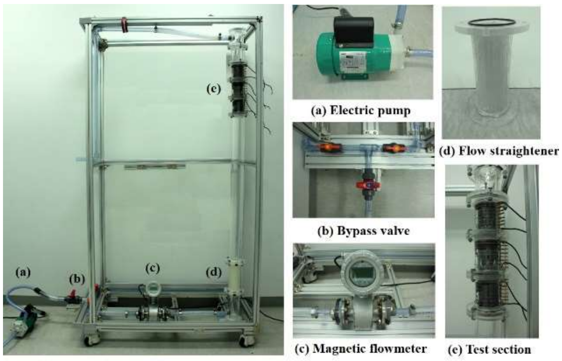


(a) Comparison of the existing studies. *(b) Comparison of the 2D and 3D results.*

FIG. 4. Local Nu ratios along the curved surface.

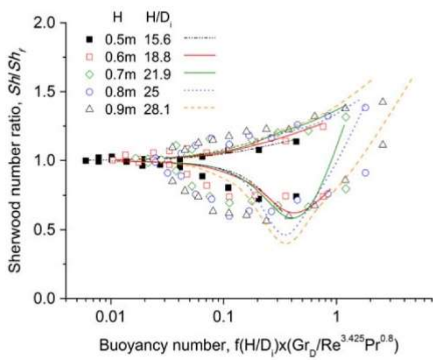
3.2. Reactor Cavity Cooling System (RCCS)

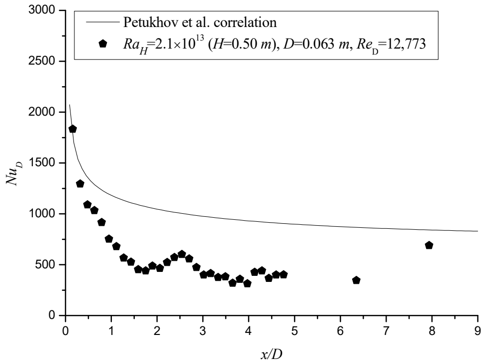
In the RCCS, where highly buoyant flows are induced in a duct geometry, the flow regime becomes similar to the mixed convection. As shown in the Fig. 5, the authors conducted mixed convection tests using a simple circular geometry in order to explore the fundamental phenomenon. Grashof number of over 1010 was achieved with the vertical pipe of less than 1 m. The empirical correlation was developed by varying the height of vertical pipe, as shown in Eq. (1). Also, the experiments were performed to investigate the impairment of local heat transfer in a vertical pipe. Fig. 6(a) shows the average heat transfers of buoyance aided and opposed flows according to the Buoyancy coefficient. Fig. 6(b) shows the non-monotonous behaviors of the mixed convection heat transfer along test section. Heat transfer decreases due to entrance effect during developing and enhances by the transition. In the turbulent mixed convection, buoyancy-aided flow showed an impairment of the heat transfer along the axial position due to the laminarization. And then, the heat transfer was enhanced as increasing buoyancy forces since the recovery of turbulence production. For this reason, mixed convection phenomenon of the cooling system must be established. Further details of these results can be found in the corresponding papers [18, 19].



(a) Test facility. *(b) Test section.*

FIG. 5. Experimental test rig for mixed convection.





(a) Mean Sh ratio with respect to the Bo [17]. *(b) Local Nu with respect to the elevation.*

FIG. 6. Mean and local heat transfer at a vertical pipe.

$$\frac{Sh}{Sh_f}=\left\{1\pm f(H/D_i)Bo\left(\frac{Sh}{Sh_f}\right)^2\right\}^{0.52}, \text{ Where } f(H/D_i)=-205.95(H/D_i)^2+7.89\times10^3(H/D_i)+3.45\times10^3. \quad (1)$$

3.3. Critical Heat Flux (CHF) phenomenon

Many studies have been performed for Critical Heat Flux (CHF) in last several decades. However, the experiments are not easy, due to the extremely high temperature and controlling problem by thermal inertia. The authors simulated boiling condition again with the electroplating system in order to overcome these problems. At a low electric potential, only the cupric ions are reduced at the cathode. But at a high electric potential, hydrogen ions in the solution reduces and forms a gas plume over the cathode surface and thus two-phase flow behaviour can be simulated by two-component flow. In spite of the difference between two-phase and two-component system, the prime phenomena of the CHF situation were observed and measured by Jeong et al. [20]. In these point of view, authors conducted experiment with horizontal copper plate. The sudden drop of the current value was measured, which similar to the CHF phenomenon as shown in Fig. 7 and took photographs in stepwise current value (a)-(d) and compared with the results by Ahn and Kim [21] (e)-(h) as shown in Fig. 8.

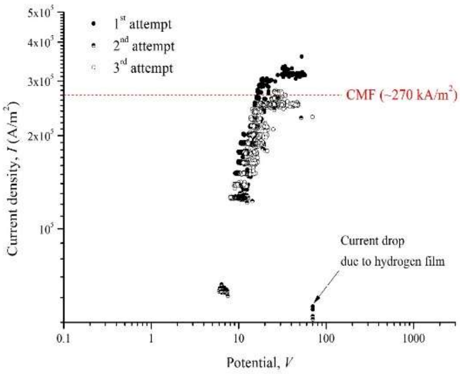
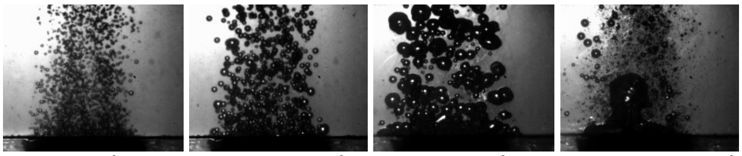
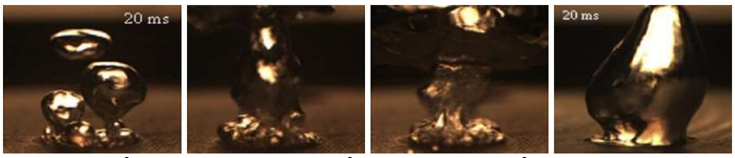


FIG. 7. Current value changes with respect to the potential increase.



(a) 10 kA/m² *(b) 50 kA/m²* *(c) 200 kA/m²* *(d) 270 kA/m²* (CMF)



(e) 100 kW/m² *(f) 400 kW/m²* *(g) 1,400 kW/m²* *(h) 1,500 kW/m²* (CHF)

FIG. 8. Bubble behavior similarity [21].

The detailed quantitative study on CHF simulation is currently performed measuring departure bubble diameter, vapor mushroom hovering time and macrolayer thickness etc. After establishing this analogy, the CHF phenomenological study on the ERVC can be performed varying geometrical parameters of the reactor insulator.

4. CONCLUSION

Several nuclear safety systems were simulated using electroplating system: IVR-ERVC situation, RCCS and CHF phenomenon. Highly buoyant conditions were tested with compact size facilities. The results were compared with the existing heat transfer results and well agreed with them. The authors figured out the basic heat transfer characteristics and flow regime of the oxide pool and mixed convection problems. Moreover, the fundamental phenomenological study have conducted using piece-wise cathode. And this electroplating system is applicable to boiling problem such as CHF phenomenon and will be developed in further works.

Although the results by the electroplating system would not be able to be accepted by the regulatory body, the authors expected that the preliminary verification of the system can be performed with the methodology.

ACKNOWLEDGEMENTS

This study was sponsored by the Ministry of Science, ICT & Future Planning (MSIP) and was supported by Nuclear Research & Development program grant funded by the National Research Foundation (NRF) (Grant code: 2014M2A8A1030777).

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ADVANCED MODELLING OF PHYSICAL PHENOMENA

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DESIGN AND DEVELOPMENT OF A RADIOECOLOGICAL DOMESTIC USER FRIENDLY CODE FOR CALCULATION OF INDIVIDUAL /COLLECTIVE RADIATION DOSES AND CONCENTRATION DUE TO RADIONUCLIDES AIRBORN RELEASE DURING THE ACCIDENTAL AND NORMAL OPERATION IN NUCLEAR INSTALLATIONS

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Abstract

Introduction of a domestic user friendly radiological dose and model that has been developed to estimate radiation exposure doses due to atmospheric and liquid discharges of radionuclides in the case of a nuclear accident and normal operation in nuclear installations. In addition, the individual doses from different pathways for different age groups, collective doses and sewerage discharge to environment as release pathway can be calculated by the developed domestic user friendly KIANA Advance Computational Computer Code for Normal and Accidental, such as follows:

1. Gaseous releases calculation.
2. Radionuclides dispersion calculation in atmosphere.
3. Calculation of liquid releases (aquatic environment).
    - 3.1. Radionuclides dispersion calculation in estuaries.
    - 3.2. Radionuclides dispersion calculation in rivers.
    - 3.3. Radionuclides dispersion calculation in little lakes and reservoirs.
    - 3.4. Radionuclides dispersion calculation in coastal waters.
4. Module for nutritional products transference.
5. Dose calculation module for critical groups.
6. Reports for each performed study at previous steps.

7. Graphics generation for each of the sector-segment previously performed studies.
8. Yield of grass and agricultural food products.
9. Harvesting and sowing time of grass and agricultural products.
10. Translocation within plants.
11. Interception.
12. Weathering from plant surfaces.
13. Dilution and Non- Dilution of radionuclide concentrations due to plant growth.
14. Uptake by plant roots.
15. Migration within the soil.
16. Plant contamination due to re-suspended soil.
17. Different livestock feeding regimes.
18. Storage times for fodder and human food products.
19. Changes in radionuclide concentrations due to food processing.

The objective of the paper is to Partial fulfilment of PhD by Research course and to create new study cases, on which calculations as well as to open, consult and modify existing study cases will be applied. Furthermore, develop a domestic user friendly dynamic radiological dose and model for normal and accidental atmospheric release of radionuclides and normal operation from a nuclear facility, which has been coupled with a long-range atmospheric transport and Gaussians dispersion model that it is first time in our country.

1. INTRODUCTION

The current Code can be coupled to any long-range atmospheric dispersion/short term model which can calculate radionuclide concentrations in air and on the ground and in the water surfaces pre-determined time intervals or measurement data. The nuclear and radioactive installations release radionuclides to the environment, affecting the atmosphere, the terrestrial surface and the surface waters, as rivers, coastal waters, estuaries, lakes and small reservoirs. The flora, fauna and human beings are directly affected by those releases, since air and water are the main flora nutrients, flora is part of the fauna diet, and flora and fauna constitute the human being feed. User Friendly KIANA Advance Computational Computer Code application is a code designed to automate the calculation of radionuclide concentrations in different environments and their impact in the nutritional chain, as well as in the human being, allowing to the researcher to center in the obtained results analysis. KIANA software tool containing many different and complex models for the simulation and evaluation of nuclear and radiological emergencies conditions can be used. The increase in functional complexity resulted in a similar increase of the complexity of the user interface. User Friendly Domestic KIANA Advance Computational Computer Code has additional requirements that have been identified as followings:

The most important requirement was found to be a simple and easy to use user interface in contrast to the another one. Ideally, the features of the user interface should be self-explaining and the intention of the input mask should be understandable even without a manual. In particular special and rarely used features should be initialized with default values and hidden in the everyday usage of KIANA Advance Computational Computer Code. Several state-of-the-art programming languages were considered in the design process. Finally, KIANA Advance Computational Computer Code was implementing using the C# programming language and the Code in this study has been designed and developed in the first time in our country.

C# is available for almost every major computer architecture and operating system especially for HP-UX, Linux and the Windows family. To support non-expert operators an internal help system has been set up based upon C# Help by PDF format [1, 2]. Type errors can be introduced easily, because there is no feedback other than evaluating the entered values with the human eye. Therefore, several features have been introduced to improve understanding and reduce the possibility to enter invalid values in KIANA Advance Computational Computer Code [1, 2]. The User Friendly KIANA Advance Computational Computer Code(software) was programmed by the Science and Research Branch of Islamic Azad University and NNSD/INRA of the AEOI under the direction and auspices of the Radiological Protection for the Public and the Environment Project which belongs to the Environmental Impact of the Radiation Monitoring Department from NNSD.The main part of Code based on S.R.S. no.19 [1] with C sharp(C#) Language but with some added improvements based on EUR 15760 [5]. The models implemented in KIANA Advance Computational Computer Code will be published by Azad University and NNSD/INRA.

2. THE CONTEXT

The objective of the paper is to develop a domestic user friendly dynamic radiological dose and model for accidental atmospheric release of radionuclides and normal operation from a nuclear facility, which has been coupled with a long-range atmospheric transport and Gaussians dispersion model. The research in this study is based on (i) atmospheric dispersion of radionuclides, (ii) dose and risk model development, (iii) validation of the model with FSAR of typically WWER-1000 Reactor. Models to represent the transport of radionuclides following atmospheric tests of nuclear weapons were developed during the 1950s and 1960s. Though radionuclides have been released into the environment during routine operational conditions of nuclear facilities, accidents and nuclear weapons tests, the KIANA Advance Computational Computer Code model that was developed for this study was planned to predict all of radiation doses and risks in the case of a nuclear accident and normal operation in nuclear installations. The novelties in this research are to couple a KIANA Advance Computational Computer Code dynamic dose and risk model with along-range atmospheric transport model to predict the radiological consequences due to accidental releases and normal operation in nuclear installations, and to perform the model simulation for NPP sites in IR. IRAN territory and with another site specification data as far as it can be acquired. Most of the mechanisms and phenomena considered in each of the existing dose and risk calculation and environmental transfer models have been compiled in the newly developed single [1, 2, 3].

KIANA Advance Computational Computer Code to lead detailed modeling. An uncertainty and sensitivity analysis can also part of the study to determine the most influential parameters and their uncertainties on the results for users (if applicable). A huge amount of data, such as radioactivity concentration in food suffs, pasture and doses, regarding the consequences of nuclear power plants' accidents and normal conditions in literature was used for development of Computer Code and its validation. We use the four method for determination of atmospheric stability classes such as the followings:

— Method of wind of fluctuations;
— Method of vertical temperature gradient and wind velocities;
— Pasquill-Turner method with IEM correction;
— The Pasquill-Gifford method.

And another novelty that had been used in the paper is use of correction allowing for the impact of the top of the atmospheric boundary layer on Plume Dispersion [1, 2]. The long-range transport model, which the Code/ software developed for this study was coupled with, was also upgraded to increase the number of pollutants modeled to provide us easiness. Besides, extensive uncertainty and sensitivity analyses associated with 96parameters have been performed for this study. The meteorological module in the existing environmental emergency response system is associated with 3-day-Domestic forecast meteorological data acquired through the State Meteorological Directorate [3]. The dispersion model that used in our case is base on IAEA safety retort series no. 19 [1] and the structures of Developed AIREM and CROM, DOZAE M, RECASS Exert Codes/model that have the capability to predict trajectories, concentration, and deposition patterns in the case of nuclear accidents and normal operations. However, doses, risks, and activities in the food chain are not calculated with the existing system in IRAN. Since the newly developed KIANA Advance Computational Computer Code for this study is compatible with the existing system's dispersion code, it can easily be integrated to it [1, 2, 3, 5].

3. KIANA ADVANCE COMPUTATIONAL COMPUTER CODE STRUCTURE

A deterministic dose calculation model called KIANA Advance Computational Computer Code has been developed for this study. For the dose assessment, all exposure pathways have been implemented as follows: Transfer of radionuclides through food chains and the subsequent internal exposures of humans due to ingestion of contaminated foodstuffs- Internal exposure due to inhalation of radionuclides during passage of cloud and from re-suspension of deposited radionuclides – External exposure from radionuclides in the passing cloud-External exposure from radionuclides deposited on the ground. The design of the KIANACode is flexible such that it can be adopted anywhere for any nuclear power plant/nuclear installations site with suitable modifications to the database [1, 2].

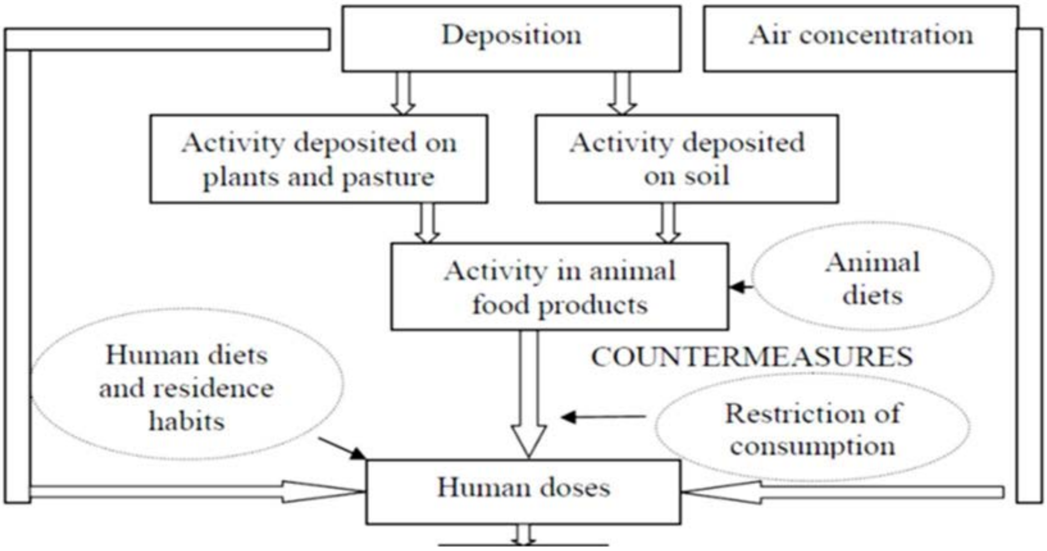

FIG. 1. Summary of Typical Code schematic Algorithms [1, 2, 5].

4. TOTAL DOSE CALCULATION

KIANA Advance Computational Computer Code calculates yearly doses for each age group and for each sector–segment after the accident and normal [1, 5]. Agricultural food products' activities are calculated at each year harvest, grass and animal products' activities are calculated on monthly basis. All aforementioned pathways are included in dose calculations as shown below:

$Dose_{total} = Dose_{inhalation} + Dose_{ingestion} + Dose_{cloudshine} + Dose_{groundshine}$, where

$Dose_{total}$ – total dose (Sv),

$Dose_{inhalation}$ – inhalation dose (Sv),

$Dose_{cloudshine}$ – ingestion dose (Sv),

$Dose_{cloudshin}$ – cloudshine dose (Sv).

A person is assumed to be as infant up to 1 year, as child up to 9 years, as teen up to 16 years and as adult up to 70 years; namely when calculating long term doses after the accident growing up of a person is taken into account in terms of his/her food consumption habits, sensitivity to doses and occupancy factors [1, 2, 5].

5. RESULT AND DISCUSSIONS

The KIANA User friendly code has dominated the process of improving the user input interfaces. Based on new recommendations and enquiries in our country and PhD by Research course of AZAD University Tessie, the formally deprecated Domestic KIANA User friendly code interface has been designed, re-engineered and implemented using state-of-the art technologies. The focus was placed on user friendliness with many additional features to support especially non frequent operators/users in the field of Radiation Monitoring and Radiation Safety issues in nuclear facilities.

The constant check of input parameters protects the operator/users from entering invalid values. The sequential lead through the input masks and the simplified entering of values gently guides the operators/users without patronizing them. All implementation as Code had been developed and performed in C# languages as well as KIANA User friendly code can be transferred to all major operating systems without any changes or recompilation. Furthermore, KIANA User friendly code can be used as an applet in a web browser thus providing a web interface for the C# based for KIANA User code.

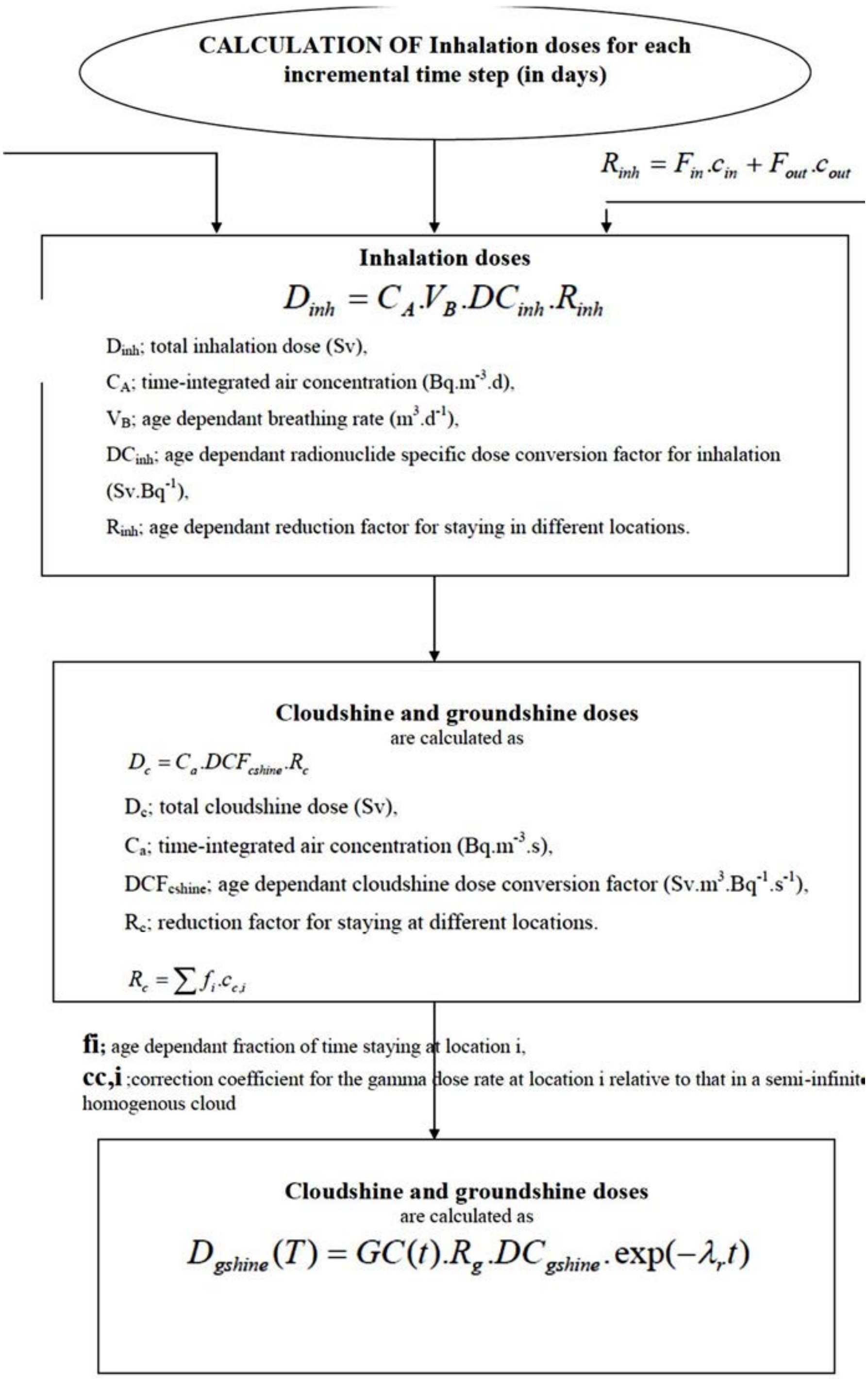

FIG. 2. Typical Flowchart and Algorithms for calculation of Inhalation doses for each incremental time step (in days) that used in Design & construction of KIANA Advance Computational Computer Code.

Calculation of Dispersion of radionuclides is also a main application area of KIANA Advance Computational Computer Code. User supplied inputs for Code calculations are as pollutant species characteristics, emission parameters, meteorological fields and output deposition definitions. The horizontal deformation of the wind field, the wind shear, and the vertical diffusivity profile are used to compute dispersion rate and concentrations. Meteorological data/formulations are required for regular time intervals. The meteorological data fields may be provided on one of the different vertical coordinate system: Pressure-sigma, pressure absolute, terrain-sigma or a hybrid absolute-pressure. The doses and time dependent radioactivity concentration values in the food products and pasture grass predicted by KIANA Advance Computational Computer Code that have been compared with those of different codes (AIREM, DOZA, CROM and RECASS Express) which participated in current validation structure assessment in KIANA Advance Computational Computer Code task, and data measured in Boshehr Nuclear plant, after severe accident had been checked by KIANA code [1, 2, 5]. Those codes are dynamic (time-dependent), and only one of them; i.e AIREM, is quasi-equilibrium [5]. Since KIANA Advance Computational Computer Code is developed as dynamic software (such as abovementioned codes), only dynamic codes' results are presented for comparison of KIANA Advance Computational Computer Code has a capability to make simulation with seven pollutants a time at most. Since some more radionuclides considered being most important in terms of their effects in the environment are used to represent accidental release of radionuclides in the literature, RECASS Express model's method source code has been used in structure of KIANA code to simulate more pollutants to provide us easiness for this study in Accidental cases.

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A DYNAMIC ASSESSMENT OF AUXILIARY BUILDING CONTAMINATION AND FAILURE DUE TO A CYBER-INDUCED INTERFACING SYSTEM LOSS OF COOLANT ACCIDENT

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Abstract

Accident scenarios with the potential to bypass containment or otherwise lead to early release of radioactive material are a focus of significant scrutiny in nuclear power plant design and safety assessment. One such scenario is an interfacing system loss of coolant accident in which the low-pressure residual heat removal system is overwhelmed by undesired physical communication with the high-pressure reactor coolant system. The pipe or heat exchanger dependent failures that may follow offer a path of contamination past containment. A cyber-induced undesired valve opening is considered that is not readily reversed from the main control room with the objective of assessing the impact of inhospitable conditions and hydrogen accumulation in the auxiliary building. Radioactive contamination and flooding in the auxiliary building may impede operators in resolving the initial valve opening event. A dynamic assessment of these uncertainties is performed using the ADAPT dynamic event tree analysis package, exploring through a structured process the likely outcomes of the combination variety of events and physical parameters of interest.

1. INTRODUCTION

In nuclear power plant (NPP) safety analysis, an early release of radionuclides represents a breakdown of the containment strategy that is designed to increase the time available for response to mitigate the consequences of an accident. The containment structure itself has numerous penetrations to accommodate ancillary systems. One class of accidents that is of concern with regards to early release in a light water reactor (LWR) is the interfacing system loss of coolant accident (ISLOCA), in which a low-pressure system beyond containment is damaged by unintended communication with the high-pressure reactor coolant system (RCS). Such events have historically been assessed to lead to large early off-site exposure [1]. The low initiating event probability of an ISLOCA often leads to it being screened out of analyses [2].

In advance of the implementation of digital instrumentation and control (DI&C) in new and existing plants [3], a great deal of research was performed for modeling the likelihood of the various failure modes of digital systems [4, 5, 6, 7]. This is necessary because the software and communications networks associated with digital systems present additional failure modes that are distinct from those in traditional analog control systems. For example, a programming error in a network device may result in spurious actuations of diverse systems in a similar fashion to a cable fire [8].

In this work, a simultaneous failure of digital controllers is considered which has the effect of opening motor-operated valves (MOVs) at full RCS pressure and endangering the lower-pressure residual heat removal (RHR) system. Plant dynamics associated with the initiating event, overpressure capacities of RHR components, and timing and success of mitigation actions by plant personnel are treated in a dynamic event tree (DET) framework which can account for complex hardware/software/firmware/process/human interactions and allows for the search of the uncertainty space in a verifiable manner for adequate completeness. Recently developed analysis tools are used to examine the resulting DET and draw insights into the importance of various events and parameters.

2. PLANT SYSTEM AND TRANSIENT

The hypothetical plant being studied in this work is a three loop pressurized water reactor (PWR) with a combined low pressure safety injection (LPSI) and RHR system located outside of containment, a configuration that exists in multiple operating plants in the USA [9]. High pressure safety injection (HPSI) and LPSI are emergency systems that can replace lost RCS inventory from external tanks in the event of a LOCA. HPSI provides a small amount of flow at pressures up to 12 MPa. LPSI can provide a larger flow rate at 2 MPa.

In the case of a sudden pressure increase, the small relief lines of the RHR system are likely to be overwhelmed. An ISLOCA subjects the RHR intake piping and the heat exchangers (HXs) to high pressure, which may cause them to fail [10]. Either type of failure has the potential to spill RCS inventory outside of containment, resulting in an early release of radionuclides and a flood in the auxiliary building.

In the event of an RHR ISLOCA, the reactor will automatically scram due to low RCS pressure. HPSI and LPSI will automatically inject water up to their operating pressures if they are available. The pilot-operated relief valves (PORVs) are automatically activated and will open and close to maintain a nominal pressure range. The effect of opening a PORV is to vent the RCS into the larger containment volume. Because of the shared RHR system, LPSI is assumed lost until recovery actions have been performed. The component cooling water (CCW) system cools the RHR HXs, RHR and HPSI pumps, and other plant systems. CCW and the systems that depend on it are assumed to be out of service in the event of an RHR HX failure, and restored once CCW is isolated from the HX.

There are actions that may be taken to mitigate the effects of the breaks. Isolating the refueling water storage tank (RWST) from the RHR system will preserve inventory for injection and lessen the extent of auxiliary building flooding. Bypassing damaged HXs restores a pathway from the RHR pumps to RHR outlet. Isolation of damaged HXs recovers systems dependent on CCW, such as HPSI. Finally, if the RHR isolation valves are stuck open the operators may open the PORVs. This action was explored in detail in the State-of-the- Art Reactor Consequence Analyses (SOARCA) project as a way to reduce the mass of water and radionuclides that blow out of containment [2].

3. DYNAMIC CASE

A DET was generated using the ADAPT driver code [11] coupled with a plant model [12] in the MELCOR light water reactor severe accident code [13]. ADAPT operates by perturbing an initial plant model according to user-supplied rules that describe possible ways the system can evolve (branching conditions) and tracking the resulting accident progression sequences [11]. The simulator (in this case MELCOR) is programmed to recognize when plant conditions indicate a branching condition has been reached. For example, if uncertainty in RHR piping failure is to be assessed, the model is programmed to stop the first time the piping is challenged by an increase in pressure. At that point ADAPT determines the reason for stopping and applies the necessary changes to a template MELCOR input file to generate at least two branches, one with no RHR piping failure and the other corresponding to the case where failure occurs. Each branch is run forward from the branching point in a new instance of MELCOR when computing resources are available until the next challenge. The process ends when all sequences reach the maximum simulation time indicated by the input.

Branches in ADAPT are assigned likelihoods that are conditional on the associated condition (e.g., ISLOCA initiation) being reached. For an uncertain parameter with a cumulative distribution function (CDF), such as the pressure capacity of piping, multiple samples may be taken from CDF to cover the uncertainty space of interest. A balance must be struck between coverage of parameter space and computational cost, as the size of the DET may grow significantly with each additional sampled value. However, it is possible to verify whether an adequate number of samples have been drawn by observing the convergence of a metric of interest (e.g., dynamic importance measures described in Section 4).

In this study, the RHR intake pipe and HXs are assumed to have uncertain pressure capacities that vary according to lognormal distributions with parameters shown in Table 1. The CDFs and initially sampled values (red stars) of these capacities are seen in Fig. 1. For illustration purposes, only the 5th, 50th, and 95th percentile values were used for each component in generating the DET. If the pressure in a MELCOR control volume exceeds the capacity at the specified CDF percentile, rupture of the represented hardware is assumed to occur. The timings for mitigating operator actions taken outside the control room are assumed to follow a Weibull distribution with parameters taken from [16], which was originally applied to the manual isolation of an auxiliary feedwater system. The CDF for the base distribution, as well as one modified with a minimum time to completion of 350s, are shown in Fig. 2. A minimum time may be set to account for unavoidable delays such as donning protective equipment and moving through the building. Values initially sampled are indicated with a red star in Fig. 2, of which the 5th, 50th, and 95th percentile values were used in this study.

A pipe break or HX rupture is assumed to initiate flow into the RHR pump room or RHR HX room, respectively, potentially bursting the room door and flooding the main hall of the building. This may delay operators, and they are assumed to be unable to complete their tasks in water levels over 1 inch. There is uncertainty as to the burst capacity of the doors; a nominal level of 5 feet is used [14], along with levels of 4 feet and 6 feet to gauge sensitivity to the burst capacity. Two mitigating actions may be taken from inside the control room: opening of PORVs and RWST isolation. These actions are branched at times of 6 minutes and 16 minutes following accident initiation, respectively, based on the SOARCA ISLOCA study [2]. Isolating the RWST from LPSI involves closing an MOV; the likelihood of failure for this action as well as opening a PORV are shown in Table 2.

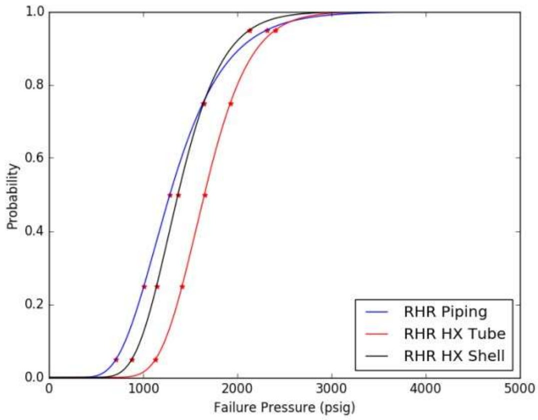

FIG. 1. CDF for RHR Component Capacities [15, 16].

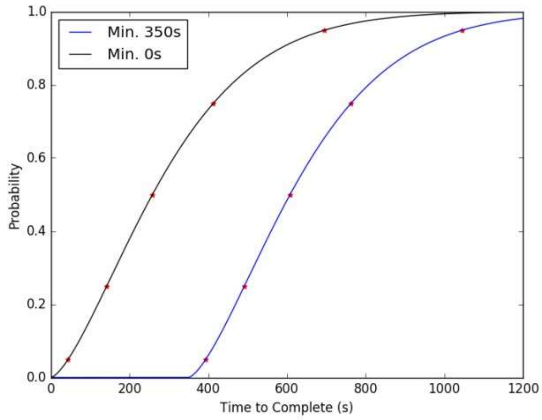

FIG. 2. CDF for time to complete actions [17].

TABLE 1. RHR COMPONENT CAPACITY

Component	Median (psig)	Log. Std. Dev.
Intake Pipe [15]	1248	0.36
Tube [16]	1650	0.23
Shell [16]	1370	0.27

TABLE 2. EVENT LIKELIHOODS

Event	Likehood (demand^{-1})
PORV fail to open [18]	$7.0*10^{-3}$
PORV fail to close [18]	$1.0*10^{-3}$
MOV fail to open or close [18]	$1.0*10^{-3}$

4. RESULTS AND ANALYSIS

The example DET was generated on a computing cluster consisting of three dual processor nodes running Red Hat Enterprise Linux 7, with capacity to run 56 branches simultaneously. This resulted in 38,000 branches and required 9 days to run. The conditional core damage probability was 2.9*10-6. The primary pressure and vessel water level are shown in Figures 3 and 4, respectively, for all sequences. Sequences that maintain a high pressure past approximately 100s are those that experienced a loss of isolation but branched into high values of pressure capacity for RHR piping and HXs. The top of active fuel is marked in Fig. 4 at 6.7m.

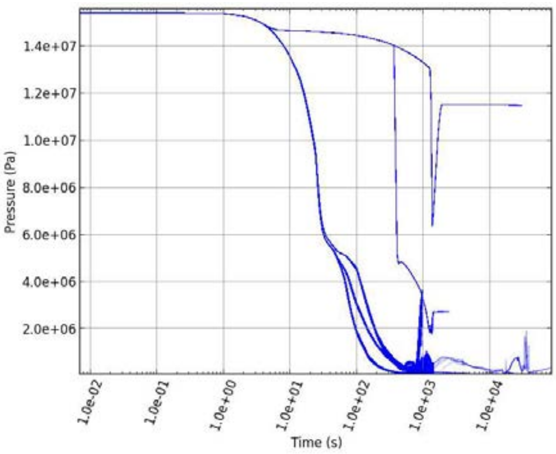

FIG. 3. Primary pressure for all sequences.

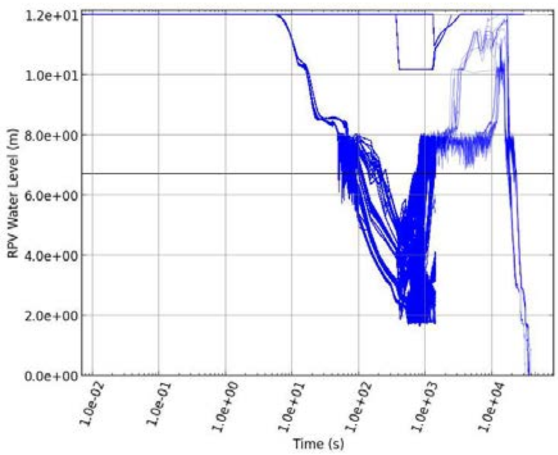

FIG. 4. Reactor Vessel water level for all sequences.

A small number of scenarios were run manually to confirm expected accident behavior. All scenarios that experienced an RHR HX break also experienced core damage. This is an expected result of the assumption that an RHR HX break disables CCW, and thus HPSI, until it is isolated. Opening the PORVs early in the scenario was observed to lessen the extent of loss of RCS inventory, as expected from the work reported in [2].

In order to determine the significance of key uncertain parameters, measures were used which compare the value of a consequence of interest (e.g., hydrogen production) for different values of the parameter of interest (e.g., time to RHR HX isolation) across a DET. Referred to as dynamic importance measures (DIMs) [19], the DIMs considered in this study are defined in Table 3, where R represents the expected value of a measure of consequence, x=0 represents the set of sequences where event x does not occur, and x=1 represents the set of sequences where event x does occur. R may be defined, for example, as the peak value of a physical measure. The value of DIM1 represents the ratio of the expected value of the consequence measure between an event occurring and not occurring. The values of DIM2 and DIM3 are functions of the value i, which is used to represent uncertainty in the extent or timing of an event x. An example calculation is given in Equation 1 for an event x that occurs in sequences A and B, and not in C. The conditional probabilities of the sequences are 0.35, 0.45, and 0.1, respectively. The values of the consequence measure R are 0.93, 0.45, and 0.55, respectively.

$$\boldsymbol{DIM1} = \frac{\frac{\sum_i^{n\ x=1} P_i * R_i}{\sum_i^{n\ x=1} P_i}}{\frac{\sum_j^{n\ x=0} P_j * R_j}{\sum_j^{n\ x=1} P_j}} = \frac{\frac{P_A * R_A + P_B * R_B}{P_A + P_B}}{\frac{P_C * R_C}{P_C}} = \frac{\frac{0.35 * 0.39 + 0.45 * 0.45}{0.35 + 0.45}}{\frac{0.1 * 0.55}{0.1}} = \mathbf{1.2} \tag{1}$$

The final value in Equation 1 is interpreted as follows: "in sequences where x occurs, the expected consequence is 1.2 times that of sequences where x does not occur". DIM1 values calculated for some relevant events are given in Table 4. Values of Inf in Table 4 indicate that there was no hydrogen generation in sequences where the event did not occur. In this case, it appears that hydrogen is produced only when an RHR HX failure occurs, and that a pipe break tends to increase it. This is expected, as an RHR HX failure also disables HPSI until the supporting CCW system can be restored.

Comparisons are made for some non-binary isolation timings in Table 5. A relationship can be seen where hydrogen production tends to increase with the time required to isolate an RHR HX shell rupture. Early isolation times are associated with decreased hydrogen generation versus cases where isolation fails, while a later isolation time is associated with increased generation. It appears that a point is reached (the location of which would be determined by further study) where late isolation is less desirable than a failure of isolation. The effects of RHR component capacities, as well as the burst capacities of the doors to the RHR pump and HX rooms are seen in Table 6. Higher levels of hydrogen production have been observed for cases with higher door capacities. Mitigating actions cannot be performed in a room while it is flooded, and bursting of the door may result in faster dryout after a leak and thus quicker recovery of affected systems.

TABLE 3. DYNAMIC IMPORTANCE MEASURES

Importance Measure	Description
$\boldsymbol{DIM1} = \dfrac{\boldsymbol{R(x=1)}}{\boldsymbol{R(x=0)}}$	Consequence ratio of occurrence to non-occurance
$\boldsymbol{DIM2(i)} = \dfrac{\boldsymbol{R(x=1_i)}}{\boldsymbol{R(x=0)}}$	Consequence ratio of value $\boldsymbol{x = 1_i}$ to non-occurance
$\boldsymbol{DIM3(i)} = \dfrac{\boldsymbol{R(x=1_i)}}{\boldsymbol{R(x=1)}}$	Consequence ratio of value $\boldsymbol{x = 1_i}$ to average of occurance

TABLE 4. DIM1 VALUES

Event (x in Table 3)	DIM1
RHR pipe break	3.8E6
RHR HX tube break	Inf
RHR HX shell break	Inf
RHR HX shell isolation	0.72

TABLE 5. DIM2 VALUES (*SEE FIG. 2 FOR PERCENTILES)

Component	DIM2
RHR HX shell isolation (5^{th*})	1.9E-6
RHR HX shell isolation (50^{th*})	1.0E-4
RHR HX shell isolation (95^{th*})	2.6E21

TABLE 6. DIM3 VALUES (*SEE FIG. 1 FOR PERCENTILES)

Component	DIM3
RHR pipe capacity (5^{th*})	1.4E-6
RHR pipe capacity (50^{th*})	8.3E11
RHR pipe capacity (95^{th*})	3.2E-7
RHR pump door (4ft)	1.0E-6
RHR pump door (6ft)	1.0E-6
RHR pump door (8ft)	9.1E17
RHR HXdoor (4ft)	1.1E-6
RHR HXdoor (6ft)	9.8E-7
RHR HXdoor (8ft)	8.8E17

5. CONCLUSION

A first-of-kind dynamic ISLOCA case was performed, considering the effects of flooding and contamination of the auxiliary building on emergency systems and the ability of operators to arrest the accident. This is an example of an accident type that has been assumed to have a low likelihood in existing analog plants, but may become more risk-significant as plants are upgraded or built with digital control equipment. The use of dynamic DET generation and post-processing tools allows for insights to be drawn that may be obscured in a traditional PRA.

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DIRECT NUMERICAL SIMULATION OF A SINGLE PHASE PTS

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Abstract

For the lifetime extension of nuclear reactors, the integrity assessment of the reactor pressure vessel (RPV) is an important issue. A Pressurized Thermal Shock (PTS) is an important transient that can threaten the integrity of the RPV during a Loss-of-Coolant Accident (LOCA). The traditional one-dimensional system codes fail to reliably predict the complex three-dimensional thermal mixing phenomena in the down-comer occurring during emergency core cooling injection. Hence, CFD can bring real benefits in terms of more realistic and more predictive capabilities. However, to gain trust in the application of CFD modelling for PTS, a comprehensive validation program is required. In the absence of detailed experimental data for the RPV cooling during emergency core cooling injection including the heat transfer into the RPV structure, Direct Numerical Simulation (DNS) databases constitute a valid alternative and can serve as a reference. The aim of this work is to perform a high quality DNS of a single phase PTS scenario, which will serve as a reference to validate lower order, more pragmatic, CFD approaches. In this regard, at NRG, an extensive research program has been carried out to design a geometrically simplified high fidelity numerical PTS set-up which will be presented in this article.

1. INTRODUCTION

Pressurized Thermal Shock (PTS) represents the occurrence of a rapid cooling of the Reactor Pressure Vessel (RPV) wall. Consequently, this sudden change in temperature prompts thermal loads on the vessel wall and under a specific set of conditions (as mentioned in [1]) it may lead to failure of the RPV. A PTS scenario exhibits a highly complex flow and the thermal fields, which cannot be predicted by commonly used one-dimensional thermal-hydraulic system codes. In this regard, Computational Fluid Dynamics (CFD) can play an important role in predicting such a complex phenomenon. However, to gain trust in the available turbulence model, their proper validation needs to be performed. For this purpose, a number of research projects [2, 3] have been carried out with a focus on the assessment on turbulence models against available experimental databases. However, due to the experimental limitations, these databases are not sufficient for a thorough validation. On the other hand, high fidelity CFD can be useful in providing an extensive database for the validation purpose.

The aim of this work is to generate a high fidelity Direct Numerical Simulation (DNS) of a simplified single phase PTS scenario, which eventually will be used to validation the available Reynolds Averaged Navier-Stokes (RANS) turbulence models. In this regards, a research program has been focused at NRG and is discussed in Section 2. In Section 3, some preliminary results of the ongoing high fidelity PTS computation are presented. This is followed by the summary.

2. RESEARCH ROUTE TAKEN AT NRG

At NRG, an extensive research programme has been carried out to perform high quality DNS computations of a simplistic single phase PTS design and is highlighted in Fig. 1. This research program consists of the three main steps:

2.1. Step 1: Code assessment to perform a true DNS

The higher order spectral element code NEK5000 [3] has been selected to perform the DNS computations. A thorough assessment of this code has been performed in order to obtain a true DNS solution. In this regard, a well-known turbulence channel flow configuration with $Re_\tau = 180$ (turbulent Reynolds number based on the wall friction velocity) was extensively reported in [2, 5, 6]. Based on this assessment, it was concluded that for the applied base mesh, a seventh or a ninth order of accuracy in space is required to achieve a high quality DNS solution. It is worth mentioning that, even for a fifth order accuracy, the obtained results were also in good agreement with the reference database. Nonetheless, the solution with N5 can be considered as an under resolved DNS (UDNS) (for details readers are referred to [5, 6]).

2.2. Step 2: DNS of a simplified single phase PTS design

This step is further divided into two parts, i.e. a PTS scenario (i) without and (ii) with buoyancy effects. Due to the complexities of the problem, a PTS without buoyancy effects has been considered as a starting point.

2.2.1. PTS without buoyancy effect

As a first step, a base geometry, based on the ROCOM test facility [2] was selected. An extensive RANS study was performed to calibrate this base geometry along with the boundary conditions (for details see [2]). The final geometric design (also shown in Fig. 2: Left) was further assessed to check its feasibility, with respect to the available computational resources, in order to perform the DNS computations. Although, the details of this PTS configuration are discussed in [5], however, a few important parameters are highlighted in Table 1.

As a next step, a high quality mesh for the simplified PTS design was generated for the selected spectral element code. It consists of 0.75 million spectral elements and hereafter is indicated as a base mesh. It is worth mentioning that the overall number of grid points depends on the selected order of accuracy. For example, for N3 (i.e. third order of accuracy in space) the total mesh corresponds to 20 million grid points. Similarly, for N5 and N7, the overall mesh corresponds to 95 and 275 million grid points, respectively. Hence, an enormous amount of computational power is required to perform the computations.

Nevertheless, in order to achieve a high quality DNS solution, this base mesh needs to run with spectral elements of seventh order of spatial accuracy (i.e. N7). However, due to the computational constraints, it has not been possible for NRG yet to run this case with N7. Nonetheless, as highlighted in Fig 1., the starting simulation was performed on N3 in order to understand the flow physics and to learn about the statistical convergence. This N3 PTS computation was performed on 1000 processors at Centrum Informatyczne Świerk (CIS) in Poland [7]. Due to the complex nature of the flow field, the simulation has run for 5 million core hrs (i.e. 208 days on 1000 processors) in order to achieve a fully converged statistical solution [6]. This means that, if we perform this PTS case with N5 and N7, it would require approximately 42 and 158 million core hours, respectively. This corresponds to a significant amount of computational power. Hence, as a next step, the solution obtained from PTS N3 is used an initial condition to perform the second run with N5. Although, N5 case does not really correspond to a true DNS, however, based on the available computational resources this simulation has been pursued. This in turns will provided a step forward to confirm the convergence towards a true DNS, from N5 to N7. In addition, it can also serve as an intermediate validation database (as shown in Fig. 1). This PTS N5 simulation is ongoing and some preliminary results are presented in Section 3.

2.2.2. PTS with buoyancy effect

Following the path of PTS without buoyancy, the calibration of the PTS case with buoyancy effects will be performed. Accordingly, high fidelity simulations, starting from pre-cursor N3 till N7 will be performed to obtained a good quality DNS database.

2.3. Step 3: Validation of RANS models

The last step of this research program is to perform an extensive validation of the available RANS models against the generated database.

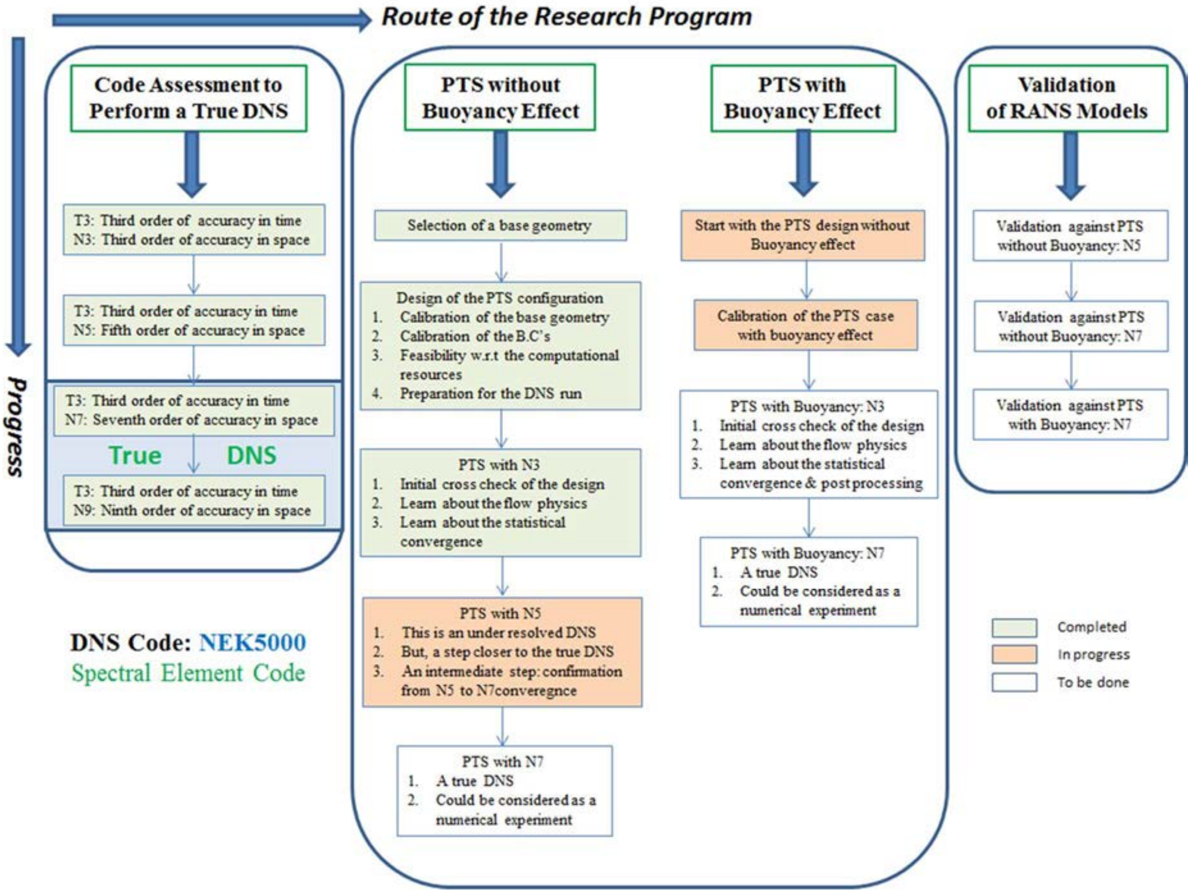

FIG. 1. A schematic of the research route taken at NRG.

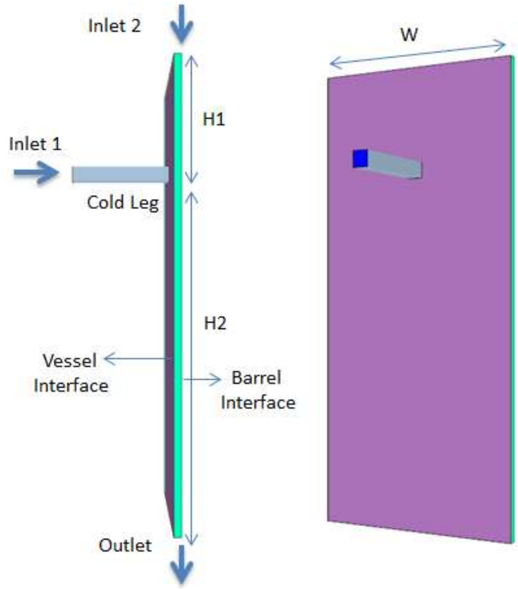

FIG. 2. Geometric design of a simplified PTS scenario without buoyancy effects.

TABLE 1. LIST OF THE PARAMETERS USED FOR THE PTS DESIGN, AS SHOWN IN FIG. 2.

Parameters	Description
Inlet 1	Instead of a round pipe, a duct is considered for the inlet section. In addition, the flow parameters are calibrated to match $Re_\tau = 180$ (for details see [2])
Inlet 2	In the absence of the buoyancy effects, an article inlet 2 is imposed to force the inlet 1 flow towards the down-comer.
Cold Leg	The length of the cold leg is 6 times the equivalent diameter (D) of the duct.
H1	The upper height of the domain is 10D.
H2	The lower height of the down-comer is 23.3D.
W	The width of the domain is 20D.
Vessel Interface	Instead of a conjugate heat transfer, two boundary conditions are imposed (i) iso-thermal (2) adiabatic
Barrel Interface	Two boundary conditions are imposed (i) iso-thermal (2) adiabatic
Outlet	Pressure outlet

3. RESULTS AND DISCUSSION

The section presents some preliminary results of an on-going PTS simulation with N5. As mentioned in the previous section, it consists of 95 million grid points. This simulation is running on 5000 processors at CIS [7]. Figure 3 displays a zoom close to the cold leg region. It is clearly noticeable that the cold leg duct shows a fully developed turbulent flow field, which represents a well-defined inlet boundary condition (see Fig. 3: Left). It is worth reminding that a fully developed turbulent velocity profile at inlet 1 is obtained by recycling the flow field in an extended duct upstream of the inlet 1. The length of this extended duct is 4 times the equivalent diameter of the duct [6]. This well-defined fully developed inlet boundary condition is very important for the validation phase in order to reduce the level of uncertainties. Furthermore, the flow from the cold leg hits the barrel wall and disperses in all direction, exhibiting a very complex three-dimensional flow field. To visualize it more clearly, a cross-section at the mid of the down-comer is selected to present the flow pattern appearing in this PTS design (see Fig. 4).

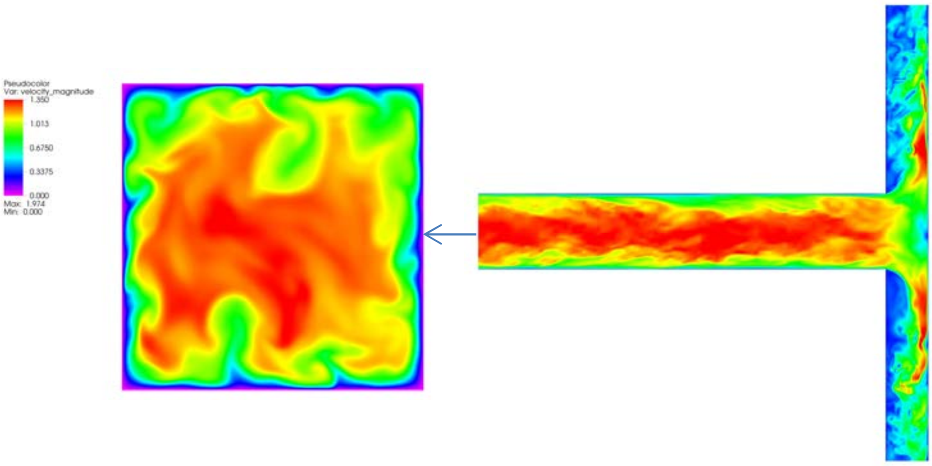

FIG. 3. Iso-contours of the instantaneous velocity field: (Left) at the inlet section (Right) a zoom close to the cold leg and the down-comer.

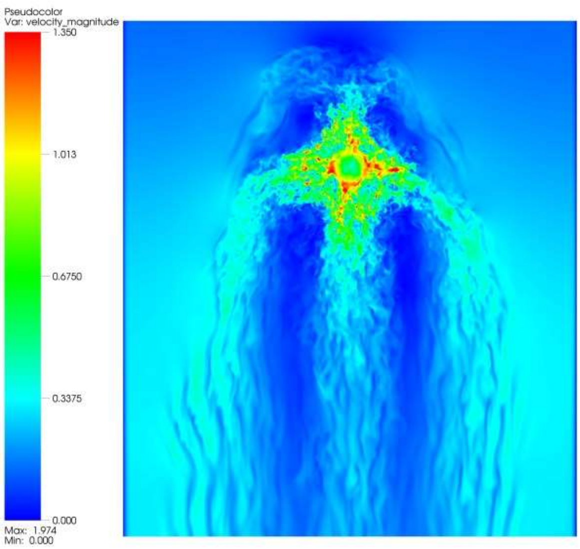

FIG. 4. Iso-contours of the instantaneous velocity field in a mid cross-section along the down-comer.

In addition to the flow field, iso-contours of the thermal fields corresponding to two selected boundary conditions are also shown in Fig. 5. These boundary conditions are carefully selected to represent different temporal stages of a transient PTS scenario. The iso-thermal condition entails the existence of a substantial temperature gradient within the wall thickness: representing early phases of the emergency core cooling (ECC) injection. In the later stages, the temperature distribution reaches to a stationary condition and such temperature gradient diminishes. This situation can be represented by an adiabatic boundary condition. In a nutshell, the PTS N5 simulation is ongoing and hopefully will yield a nice database for an intermediate validation phase, as highlighted in Fig. 1.

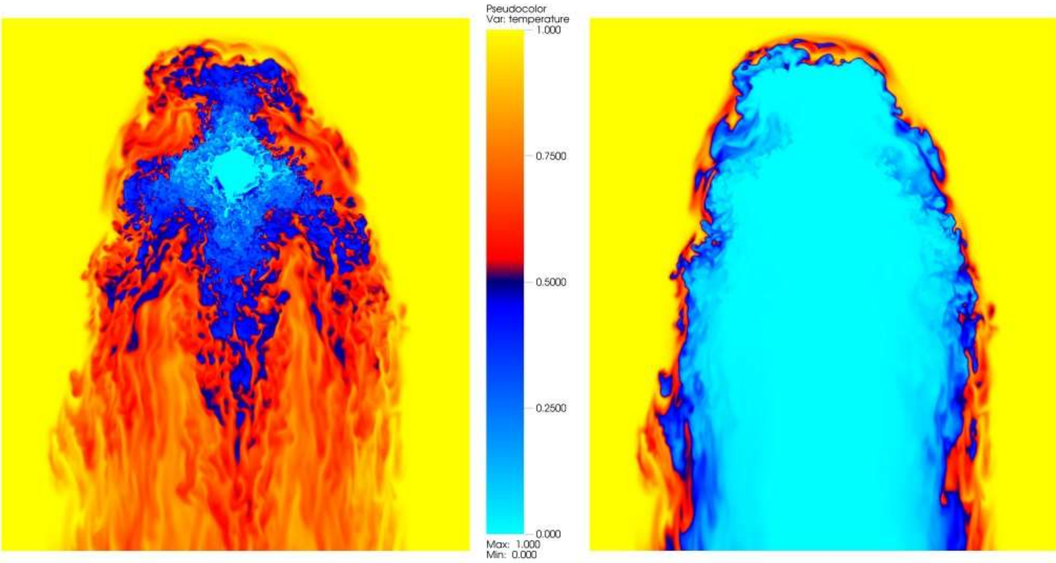

FIG. 5. Iso-contours of the temperature field along the mid cross-section corresponding to (Left) iso-thermal and (Right) adiabatic boundary conditions.

4. SUMMARY

An extensive research program has been carried out at Nuclear Research and Consultancy (NRG), with an aim to generate high fidelity DNS databases for a simplified single phase PTS scenario with and without buoyancy effects. In this regard, an enormous effort has been put forward for the design of a simplified geometric design for a single phase PTS scenario without buoyancy effects. In addition, a thorough assessment of the code NEK5000 has been done in order to perform a true DNS for a PTS like scenarios. Accordingly, a systemic progression towards the generation of the DNS database, for a PTS scenario without the buoyancy effects, is ongoing and seems to result in a successful campaign. This will provide a one of a kind reference DNS database to validate available RANS turbulence models.

ACKNOWLEDGEMENTS

The work described in this paper is funded by the Dutch Ministry of Economic Affairs. The DNS computations presented in the paper are performed at Swierk Computing Centre within the framework of the EU and MSHE grant no. POIG.02.03.00-00-013/09.

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COMPUTER CODES FOR SAFETY ASSESSMENT

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ANALYSIS OF PWR SEVERE ACCIDENT SEQUENCES INCLUDING MITIGATIVE MEASURES TO PREVENT OR DELAY THE FAILURE OF SAFETY BARRIERS WITH THE SEVERE ACCIDENT CODE ASTEC

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Abstract

After the Fukushima accident, several international research initiatives started to investigate all safety-relevant phenomena occurring during severe accidents. In this frame, a re-evaluation of the safety features of operating nuclear power plants was performed to identify the strengths /weaknesses of different designs. As a consequence, severe accident cooperative research was worldwide intensified and the Fukushima accidents are being evaluated. The obtained data will be of paramount importance for code validation. In Germany, the WASA-BOSS Project started in 2013 to evaluate the capability of severe accident codes for LWR-severe accidental sequences including accident management measures. Similarly, the European CESAM project that gathered many European research institutions has been launched to enhance the simulation capability of the European Reference Code ASTEC to simulate the behaviour of different plant designs e.g. PWR, BWR, VVER and PHWR in case of severe accidents. At KIT, investigations have been performed in the frame of the CESAM Project to evaluate the ASTEC capabilities to describe PWR severe accident sequences e.g. MBLOCA, SBLOCA and SBO including SAM measures. For this purpose, ASTEC was firstly validated using experimental data and later on applied to analyse different accidental sequences e.g. SB-LOCA including SAM measures. In this paper, the influence of several SAM measures on the in-vessel accident progression and on the integrity of the reactor pressure vessel (RPV) investigated with ASTEC for a Small Break LOCA and for a Station Black-out (SBO) sequences will be presented and discussed in detail.

1. INTRODUCTION

In the last two decades, a significant progress on Severe Accident Management strategies through the elaboration of plant specific SAM Guidelines (SAMGs) has been observed. Despite of this significant progress, the severe accidents at Fukushima (2011) have highlighted that further improvements of SAMGs are still necessary. For the development and optimization of SAMGs, a solid basis of risk-relevant analyses must be produced using severe accident simulation tools. The ASTEC code, developed by the *Institut de Radioprotection et de Sûreté Nucléaire* (*IRSN*) and the *Gesellschaft für Anlagen und Reaktorsicherheit (GRS)*, is becoming the reference integral SA code in Europe. The code is able to simulate the progression of a SA from the initiating event till the release of radioactive material from the containment. Within the European CESAM project [1], ASTEC capabilities regarding the assessment of SAM measures are being extended through model development, code validation and reactor calculations. In the frame of such project, KIT-INR is devoted to the verification and extension of SAM measures in a German Konvoi PWR using ASTEC, considering the lessons learnt from the Fukushima accidents [2].

Within this work, the ASTECV2.0 (rev3) is used to investigate the impact of selected SAM measures (e.g. secondary and primary bleed and feed procedures) on the progression of postulated SBLOCA (with SBO) and SBO sequences, which are relevant for the risk of core damage and containment failure according to the outcomes of the Probabilistic Safety Analysis (PSA) for the Konvoi PWR [3]. First of all, the ASTECV2.0 model of the Konvoi PWR is introduced. Then, the reference SBLOCA and SBO sequences are analysed. Finally, the influence of SAM measures is discussed and general recommendations concerning SAM are proposed.

2. THE ASTEC CODE

The ASTEC code simulates complete severe accident sequences for water-cooled nuclear reactors. The structure of ASTEC is modular, each module representing a zone of the reactor (primary loop, secondary loop, safety systems, vessel, etc.) or a set of physical phenomena. Two modules are of interest for the current work: CESAR and ICARE. Further details about all ASTECV2.0 modules and their modelling features can be found in [4]. The CESAR module calculates the thermal-hydraulics in the core region up to the beginning of core degradation, and in the rest of the primary and the secondary circuit. CESAR makes use of a 5-equation approach to solve the thermal-hydraulics in 1-D. ICARE describes the thermal-hydraulics in the core region after the onset of core degradation. For this purpose, the module uses the same approach to solve the thermal-hydraulics as CESAR. The 1-D approach is valid as long as the axial velocities are predominant over the radial ones. However, when significant blockages have been formed in the core region, a 2-D resolution of the thermal-hydraulics is necessary. The reflooding model used in the current work is devoted to bottom reflooding assuming that the core geometry is sufficiently intact so that it can be treated with a 1-D approach.

3. GENERIC ASTEC MODEL FOR A KONVOI PWR

The reference plant is the four loop German Konvoi PWR. The ASTECV2.0 model of a generic German Konvoi PWR used in the current work has been derived from a more detailed one used by GRS in previous studies [5]. The geometry of the reactor domains, the physical phenomena considered during the in-vessel phase, the automatic actions taken by the Reactor Control Protection System and the most relevant safety systems are identical to the ones described in [6, 7]. In the following, a description of additional safety systems and key information from the aforementioned reference is given.

The four loop PWR is represented by two loops: the loop B (containing the pressurizer) and the loop A (containing the other three loops). Simplified sketches of the primary and secondary circuits of the loop B and the RPV are depicted in Fig. 1.

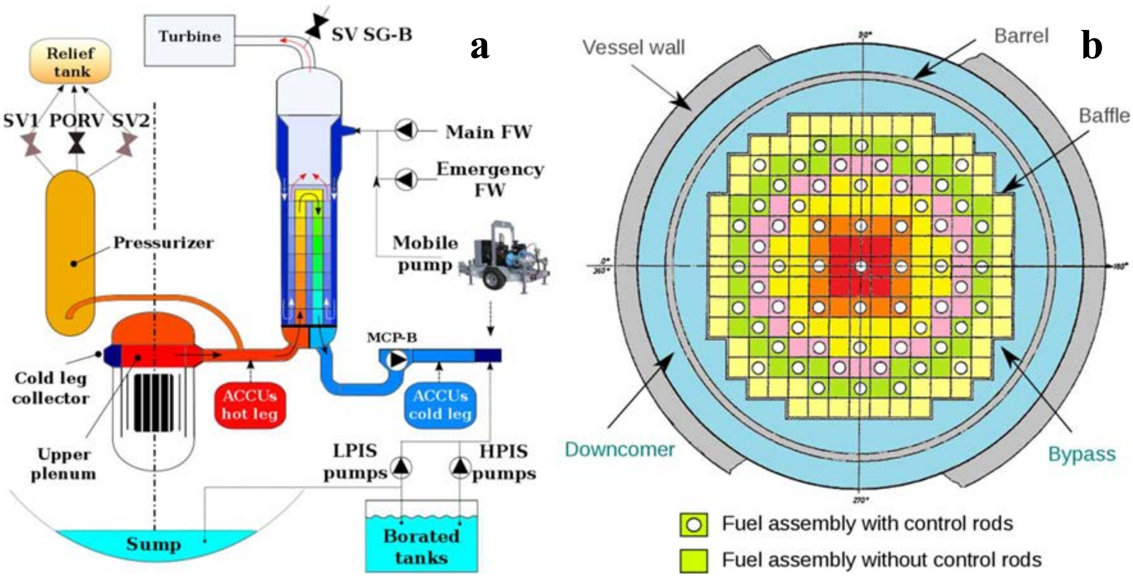

FIG.1. Sketch of (a) the primary and secondary side of loop B (b) the RPV for a generic German Konvoi PWR by ASTECV2.0 [6].

The RPV is divided in six rings for the core region, the bypass and the downcomer. The pressurizer is equipped with a Pilot Operated Relief Valve (PORV) and two Safety Valves (SV1 and SV2). Each SG is equipped with a Safety Valve. The Emergency Core Cooling System (ECCS) consists of accumulators, the High Pressure Injection System (HPIS) and Low Pressure Injection System (LPIS). The last two take water from four borated tanks or from the sump. Mobile pumps have been included in the model to simulate an external injection into the primary or the secondary side.

Regarding the SAM procedures used in a German Konvoi PWR, secondary bleed and feed (SBF) is initiated when the liquid level of all SGs is lower than 4 m or when the plant has been for more than 20 min without any AC power supply [8]. The action is modelled through the opening of the SV SG-B and the injection by means of mobile pumps upon the fulfilment of such criteria. Primary bleed and feed, also named Primary Side Depressurization (PSD), is initiated when the Core Exit Temperature (CET) exceeds 400 °C or the core liquid level falls below *min3* [8]. This action is modelled through the opening of all pressurizer valves upon the fulfilment of the first condition (CET > 400 °C).

4. SEQUENCES WITHOUT SEVERE ACCIDENT MANAGEMENT

The current section describes the behaviour of the risk-relevant sequences in a German Konvoi PWR [3] by means of ASTECV2.0 simulations. In particular, a Small Break LOCA in the main coolant line, which is relevant for the risk of core damage (PSA-1), and the total Station Blackout, which is relevant for the risk of fission product release to the environment (PSA-2), have been considered. For both sequences, an automatic intervention of the systems introduced in section 0 is assumed. Additionally, the following assumptions are made: (1) No failure of the surge line due to high temperatures; (2) No seal LOCA in the Main Coolant Pumps, (3) No thermal-induced SG Tube Rupture and (4) No automatic 100 K/h cooldown of the reactor through the SGs. The calculations are limited to the in-vessel phase and hence, they are terminated when RPV failure is calculated.

Concerning SBLOCA, it is considered that a 10 cm^2 break occurs in the cold leg of the pressurizer loop at 0.00 h. Furthermore, the sequence assumes the total loss of AC power supply 1 h after SCRAM and the availability of 3 out of 4 trains of the HPIS. On the other hand, the Station Blackout sequence postulates the total loss of AC power supply at 0.0 h without any kind of break in the main coolant line. The pressure evolution in the primary circuit as a function of the time after SCRAM is depicted in Fig. 4.

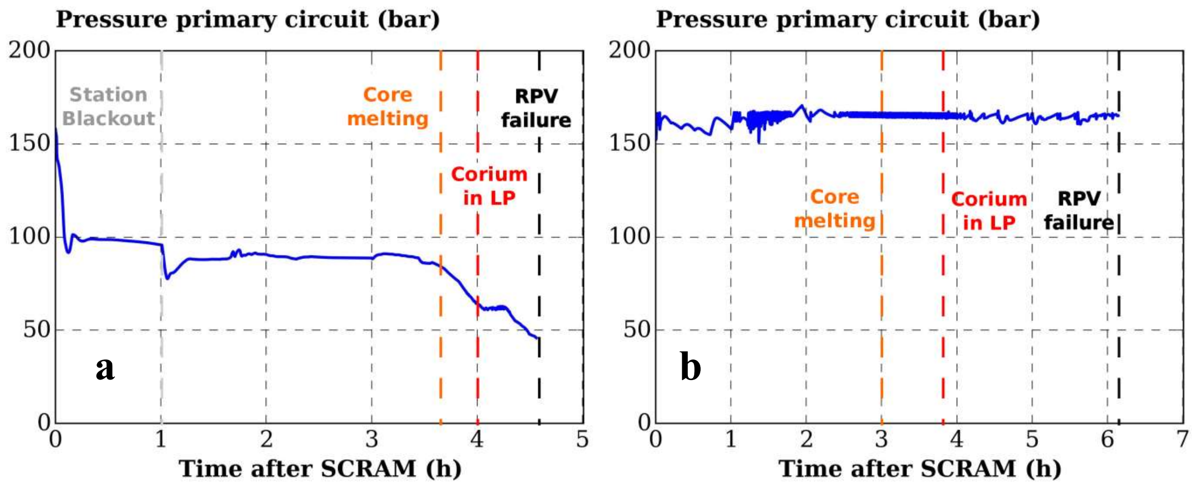

FIG. 2. Evolution of the primary pressure during the postulated (a) SBLOCA (b) SBO sequences as a function of the time after SCRAM in hours. Vertical lines mark the occurrence times of major events in each simulation.

For the considered SBLOCA sequence (Fig. 4-a), the pressure falls down at the beginning of the transient, since the water is leaving through the break. A few seconds later, when the pressure falls below 110 bars, the HPIS injects the inventory of the borated water tanks into the circuit. Consequently, the pressure remains at about 100 bars up to 1 h after SCRAM. At that time, there is a total loss of AC power, which automatically cuts off the HPIS. Afterwards, no active system can provide coolant to the reactor. Initially, the pressure decreases because the water is leaving the circuit at the cold leg. However, after some minutes, the water in the hot leg reaches saturation and starts to evaporate, which contributes to the increase of the pressure. This pressure balance is kept up to about 3.50 h, when the steam evacuation from the circuit compensates the overpressure of the gases exiting the circuit. In contrast to SBLOCA, where RPV failure occurs at medium pressures, the SBO sequence (Fig. 4-b) leads to high pressure sequences just after the loss of AC power. This happens because the water inventory of the SGs is rapidly depleted (in about 1.20 h) and hence, the heat sink of the primary side is lost. As a result, the temperature of the water and the pressure in the primary circuit starts to increase up to the set point of the pressurizer safety valves, where it remains until RPV failure.

5. SEQUENCES WITH SEVERE ACCIDENT MANAGEMENT

In the previous section it was that, if no Accident Management measures are considered, the postulated accidents lead to core melting, corium relocation to the lower plenum and, eventually, RPV failure, which also occurs at medium-high pressures. Within this section, several accident management measures are put in place to prevent, mitigate or delay core melting and the failure of the RPV. In this context, primary bleed and feed is investigated as a mean to stop the progression of the SBLOCA, whereas secondary bleed and feed is the selected measure to stop the progression of the SBO. The influence of both actions is shown in separated diagrams (FIG. 3 and FIG.4) in the form of timelines. The bar located at the top represents the timeline of the sequence without any accident management, whereas the vertical dashed lines mark the occurrence times of other relevant events of that simulation.

As far as the SBLOCA sequence is concerned, FIG. 3 shows that PSD significantly delays core melting and vessel failure if it is performed with a maximum delay of 20 min after the detection of a CET = 400 °C. On the other hand, a 60-min delayed PSD cannot delay RPV failure with respect to the reference sequence. This happens because at least 20 tons of corium has been heating up the upper metallic layer in the lower plenum and the vessel walls component in contact with it. Therefore, the water supplied by the accumulators cannot cool down the metallic layer, this eventually leading to vessel failure. In between, the 40-min delayed offers the longest grace time to vessel failure. This occurs because the water inventory from the accumulators covers the corium in the core region and hence, ASTECV2.0 shifts the corium relocation to a later time point. However, the validity of this finding should remain in question until further analyses with the ASTECV2.1 version [9], which incorporates new models for degraded core reflooding, would be performed.

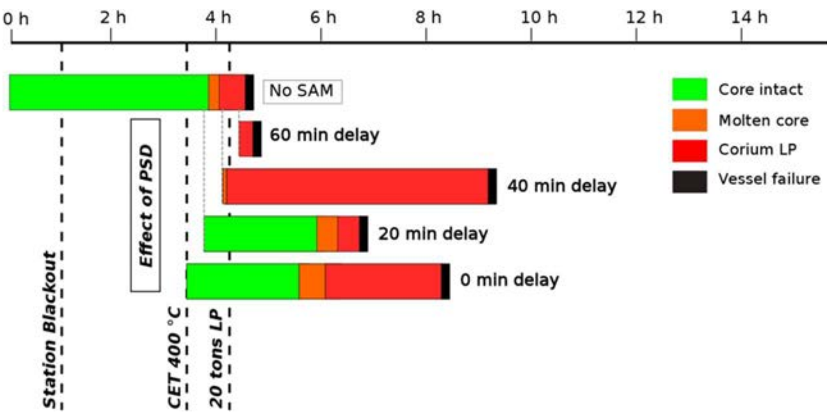

FIG. 3. Impact of primary bleed and feed on the progression of the SBLOCA sequence. Vertical lines mark the occurrence times of major events for the SBLOCA sequence without any SAM measure.

The timelines associated to the SBO sequence (Fig.6) differ with respect to the ones of SBLOCA in certain aspects: first of all, the delay of the measure is evaluated after the secondary bleed and feed conditions are fulfilled; second, each bar (except the one located at the top) has two attributes: the mass flow rate injected by the mobile equipment into the depressurized SGB and the aforementioned delay in the performance of secondary bleed and feed. It can be observed that SBF is effective regarding the prevention of core melting if it is performed before the detection of CET = 400 °C providing that a mass flow rate higher than 10 kg/s is provided into the depressurized SGB. Beyond such time, SBF cannot prevent vessel failure, but may delay it if the supplied mass flow rates are above 10 kg/s. For all cases, an injection rate below 5 kg/s does not affect the time of vessel failure, which occurs at similar times as the reference case without SAM actions.

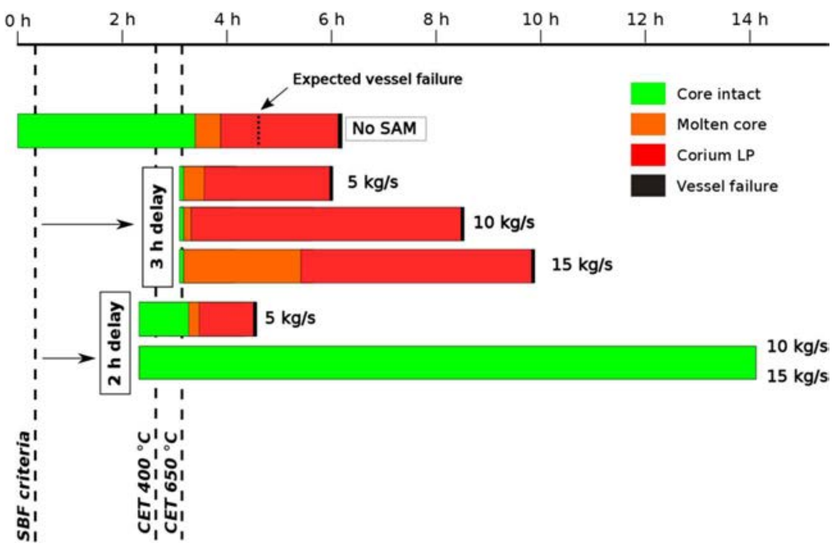

FIG.4. Impact of secondary bleed and feed on the progression of the Station Blackout sequence. Vertical lines mark the occurrence times of major events for the SBO sequence without any SAM measure. Flow rates are referred to the injection rates into the depressurized SG i.e. SGB.

6. CONCLUSIONS

The elaboration of SAMGs requires a solid database of deterministic analysis performed with validated severe accident codes. Within the EU CESAM project, KIT-INR is involved in the extension of the technical basis for the verification, improvement and development of SAM measures for a German Konvoi PWR using ASTEC. The analyses are based on the findings of the PSA for the German Konvoi PWR and the lessons learned from the Fukushima accidents. Within this work, the impact of selected SAM measures (e.g. primary and secondary bleed and feed) on the progression of the risk-relevant SBLOCA and SBO sequences was studied. The results show that secondary bleed and feed should be initiated before detecting a CET=400 °C with an injection rate of at least 10 kg/s to prevent core degradation (according to SBO), and that primary bleed and feed should be initiated with a maximum delay of 20 min after the detection of CET=400°C in order to significantly delay core melting and vessel failure. Future investigations will deepen in the parametrization of the aforementioned measures, with a special attention to the reflooding of overheated cores during SBO conditions, similarly to the analysis performed in [6]. The verification of the findings concerning degraded core reflooding will be also addressed with the new ASTECV2.1 version [9].

ACKNOWLEDGEMENTS

The authors would like to thank GRS-experts for providing the Konvoi input deck and the ASTEC team at IRSN in Cadarache for continuous support. Additionally, the authors would like to acknowledge the financial support from the EU CESAM consortium and the NUSAFE research program of KIT.

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COUPLING OF THE THERMAL-HYDRAULIC CODE (RELAP5) AND THE MONTE-CARLO NEUTRONICS CODE (SERPENT)

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Abstract

In the paper, external (loose) coupling between a thermal-hydraulic system code (RELAP5) and a Monte-Carlo reactor physics code (Serpent) is performed to investigate potential improvements in prediction capability comparing with stand-alone codes simulation. The coupling is performed for simulation of a single fuel assembly based on the OECD-NEA/NRC PWR MOX-UO_2 Core Transient Benchmark. The procedure is based on driving scripts written in MATLAB and Python as a tool to exchange data between input and output files of the two coupled codes. Different coupling schemes and different convergence criteria are implemented and tested. Convergence is attained after 10 iterations, and it is shown that simulation starting with RELAP5 using a volume-weighted average for temperature feedback gives consistent results.

1. INTRODUCTION

The need for increasing computational prediction capability is continuously growing for high fidelity design of nuclear power plants. In order to correctly capture the different physics involved in a nuclear reactor, multi-physics approaches are commonly used. The ultimate goal of these approaches is to analyze relevant physical phenomena in a nuclear reactor. This includes: thermal-hydraulics, neutron-kinetics, material behaviour etc. There are several multi-physics computational platforms currently under development, but none of them seems to achieve sufficient certainty for general release [1].

For code coupling performance, several aspects need to be considered to achieve less computational time and better prediction capability. Some of these aspects are: the convergence criteria and the update process of the cross section data. Several convergence criteria have been adopted in literature; for instance, Volkan [2] used eigenvalue and flux values for a convergence criterion, while others [3, 4, 5] chose local fuel temperature as the convergence parameter. Another convergence criterion implemented by Kotlyar [6] was based on the local power distribution. Moreover, the uncertainties in Serpent calculations were used for a new convergence criterion adopted by Wu [7]. Temperature dependence of the cross-section data in a coupled simulation must be taken into account. Different methods have been used to accurately capture the strong dependence of cross-section data on temperature [1]. The present work describes external (loose) coupling between the thermal-hydraulic safety analysis code, namely RELAP5 and the Monte Carlo reactor physics code, the Serpent. In this study, the codes coupling is performed to investigate potential improvements in prediction capability comparing with stand-alone codes simulations.

2. BENCHMARK SPECIFICATIONS

In this work coupling is performed for simulation of a UO_2 fuel assembly based on OECD/NEA and U.S. NRC PWR MOX/UO_2 core transient benchmark [8]. This benchmark was designed to offer the ability to evaluate modern reactor kinetic codes in terms of their ability to predict the transient response of a core partially loaded with MOX fuel.

The fuel assembly used in this study is based on a 17×17 Westinghouse fuel design (Fig. 1), with 4.2% Uranium enrichment. Each fuel assembly has 264 fuel rods and 25 guide tubes, 24 of them for control rods and one for instrumentation cell. The schematic configuration of assembly is shown in Fig. 1 and more detailed information of the benchmark design dimensions and boundary conditions are listed in Table 1. Material compositions inside the fuel assembly are shown in Table 2. Thermal-Physical Properties provided by Finnemann and Galati are used [9].

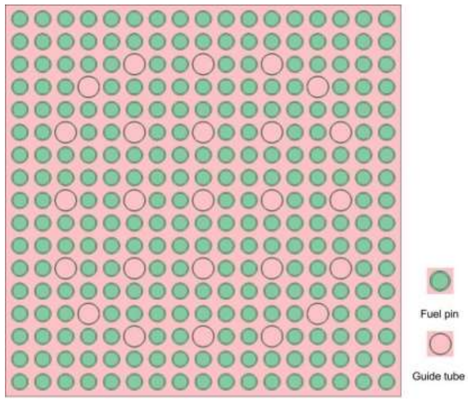


FIG. 1. UO_2 assembly configuration.

TABLE 1. BENCHMARK DESIGN DIMENSIONS AND BOUNDARY CONDITIONS

Property	Value
Number of fuel assemblies	193
Fuel lattice, fuel rods per assembly	17x17, 264
Number of guide tubes	25
Pin pitch (cm)	1.26
Assembly pitch (cm)	21.42
Active fuel length (cm)	365.76
Fuel rod inner radius (cm)	0.3951
Fuel rod outer radius (cm)	0.4583
Guide tube inner radius (cm)	0.5624
Guide tube outer radius (cm)	0.6032
Inlet temperature (K)	560
Core inlet pressure (MPa)	15.5
Total active core flow (Kg/s)	15849.4
Power level (MWth)	3565

TABLE 2. MATERIAL COMPOSITION

Material	Density (g/cm^3)	Composition
UO_2	10.24	U^{235}:4.2 wt%, U^{235}:95.8 wt%
Cladd	6.504	Zr: 98.23 wt%, Sn: 1.5 wt%, Fe: 0.12 wt%, Cr: 0.1 wt%, N: 0.05 wt%
Coolant	0.75206	H_2O at 560 K and 15.5 MPa

3. COUPLING METHODOLOGY

3.1. Models and Meshing

- RELAP5 Model

The thermal-hydraulic model consists of coolant flow in a single pipe with internal flow area of 2.5604 cm^2 and a height of 3.6576 m. This length is divided into 24 axial cells, 15.24 cm each. The flowing coolant (water) is initially pressurized to 15.5 MPa and exhibits an initial temperature of 560 K. The power from nuclear fission is added into the coolant flow by means of a "decoupled" heat structure. The heat structure has the same height as the pipe, and divided equally into 24 axial cells. Further, the heat structure is divided into ten radial meshes (nine intervals), six intervals of the fuel region, one interval of the gap, and two intervals of Zircaloy cladding. The output of this RELAP5 model contains temperatures and densities of the coolant along the 24 axial cells of the pipe.

- Serpent Model

In Serpent, the fuel assembly is modelled based on a pin-wise approach (no major approximations on the geometry of the assembly). For simplicity, control rods and instrumentation cells are ignored and their material is replaced with cladding tubes filled with water. Serpent updates cross section data automatically using a method called the pseudo-material method [1]. For a Serpent run, 1 million neutrons per cycle were used with 1000 active cycles and 100 inactive cycles (total of 1.1 billion neutrons per run). This insures sufficient degree of certainty in simulation results.

- Coupling Scheme

External coupling between the Serpent and RELAP5 was performed to obtain the behaviour of the system in a single fuel assembly model. Python and MATLAB scripts were used to exchange data between the codes and to perform convergence evaluation. Two schemes are applied in the present work to perform the external coupling of RELAP5 and Serpent:

— Scheme I: Serpent First
— Scheme II: RELAP5 First

For the first scheme, coupling starts with executing a Serpent input file with initial estimations of temperature and density profiles. Data exchange between the two codes followed upon the completion of each run of either code. Results from this scheme are not satisfied and not presented in this paper. The second scheme started with executing RELAP5, with a flat power distribution as an initial guess. The scheme algorithm is shown in Fig. 2. This scheme is performed for two different calculation methods for the average feedback temperatures based on radial mesh point's values:

— Method I: Fuel temperatures are calculated as the arithmetic mean of the 7 inner meshes and clad temperature as the arithmetic mean of the 3 outer meshes.
— Method II: Weighted-average temperatures are calculated based on volume fractions for the fuel and the cladding regions.

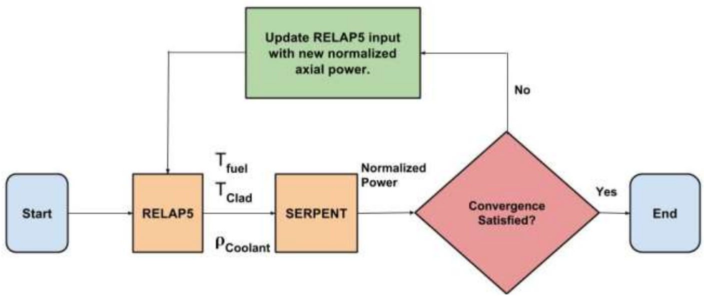


FIG. 2. Algorithm for the second coupling scheme.

- Convergence Criteria

Three convergence criteria are investigated in this study. The main criterion is based on the uncertainties in Serpent calculations, while the other two were based on relative change in fuel and cladding temperatures. In this work, convergence criteria were not implemented to get results. Rather, results are used for making comparison between the three convergence criteria. In other words, coupling is performed for ten iterations (convergence is guaranteed from literature), and then the results of this coupling is used to compare different convergence criteria.

3.1.1. Monte-Carlo Inherent Uncertainty

Monte-Carlo neutronic codes have inherent statistical uncertainty. The uncertainty in the axial power distribution calculated by Serpent is propagated to the thermal-hydraulic parameters calculated by RELAP5 when both codes are coupled [7]. Therefore, it is reasonable to implement a convergence criterion based on these uncertainties rather than using some arbitrarily chosen tolerance.

- Relative Change in Fuel and Cladding Temperatures

These two criteria are based on relative change in the volume weighted-average of temperature (fuel or cladding) at each cell along the 24 axial cells. Convergence is achieved when the relative change in temperature is less than the value of a convergence parameter ϵ which is expressed in the following equation:

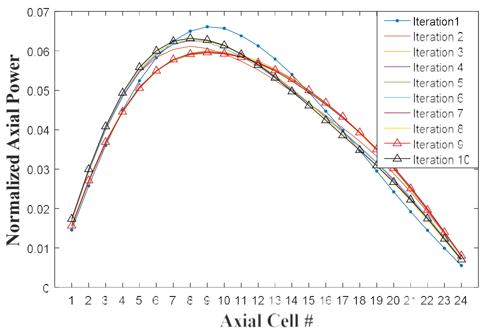
$$\left|\frac{T_j^{current} - T_j^{previous}}{T_j^{current}}\right| \leq \epsilon \quad (1)$$

The subscript j refers to fuel or cladding. The convergence parameter ϵ is chosen to be 0.0055 for fuel temperature and 0.001 for cladding temperature.

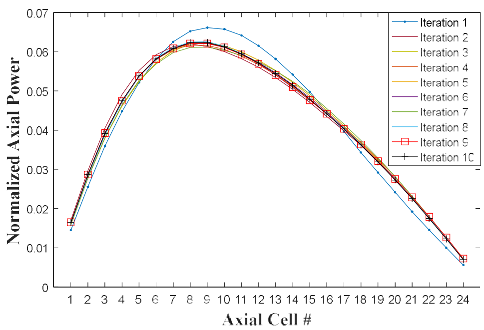
4. RESULTS AND DISCUSSION

Calculations were carried out on Linux platform using a cluster with 3.47 GHz Xenon X5690 series CPU and 45 Gb RAM. Calculations were done in parallel mode on a single node with 24 cores. Each run (iteration step) required 5 hours of execution.

To analyse the behaviour of the coupling scheme, the normalized power distribution is described upon the completion of each iteration. Fig. 3 shows the normalized power distribution for two different treatments of temperature feedback (Methods I and II as discussed before). It can be seen that Method II yields more exact results and has faster convergence.



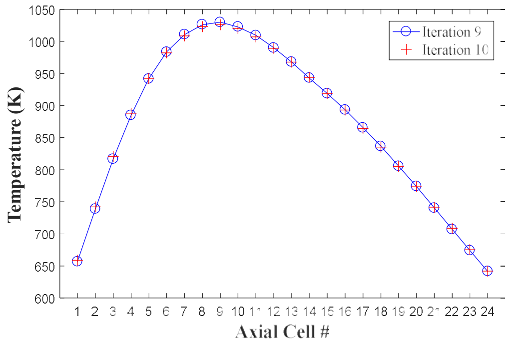
(a)



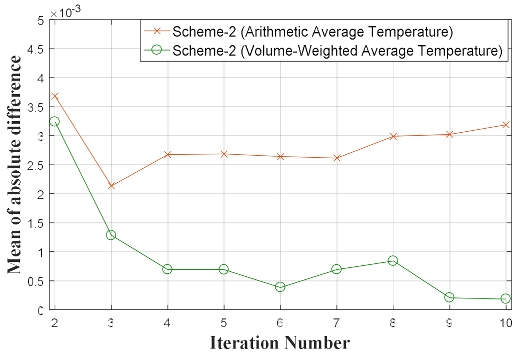
(b)

FIG. 3. Convergence of power distribution (a) for Scheme I, (b) for Scheme II.

It is shown that the system is converged within nine iterations if the convergence criterion based on Monte-Carlo uncertainty is used. The normalized axial power distribution for the last two iterations and the convergence behaviour for the two methods are shown in Fig. 4. Based on the other two convergence criteria (relative change in fuel and clad temperatures), the scheme is converged within ten iterations.



(a)



(b)

FIG. 4. Coupling convergence (a) normalized axial power for last two iterations (b) convergence behavior.

5. CONCLUSIONS AND FUTURE WORK

In this project, the coupling between the thermal hydraulic code RELAP5 and the Monte-Carlo neutron-kinetic code Serpent was successfully performed on the OECD-NEA/NRC PWR MOX-UO2 Core Transient Benchmark. Coupling has been carried out using Python, BASH shell, and MATLAB scripts. Different coupling schemes were used, and different convergence criteria were implemented. The influence of the way temperature is updated was tested. It was found that starting with RELAP5 with a temperature feedback based on a weighted-average showed best results in terms of consistency and convergence. This work can be improved and continued by using different mechanisms for updating fuel temperature. Comparing the results of such experiments can reveal an optimal mechanism for a faster and more rigorous coupling scheme. Another way of improving this work is to implement the available coupling schemes and techniques to different benchmark problems. Such a study would determine the limits of this work and its applicability to different types of problems with different geometry and compositions.

ACKNOWLEDGEMENTS

Gratitude is duly expressed to the Jordan Atomic Energy Commission (JAEC) and their Jordan's National Supercomputing Center (IMAN1) for offering the computation resources needed for this work. This work would have not been possible without their kind support and assistance.

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COMPARATIVE ANALYSIS OF WWER-440 REACTOR CORE WITH PARCS/HELIOS AND PARCS/SERPENT CODES

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Abstract

Currently predominantly deterministic 2D lattice codes (HELIOS, CASMO, WIMS) are used for few group macroscopic cross-section libraries generation for further reactor core 3D static and dynamic performance analysis by nodal diffusion codes, like PARCS, DYN3D etc. However recent developments in high performance computing and new generation of Monte Carlo codes makes it possible to get rid of all shortcomings of 2D lattice codes and generate cross-section using 3D geometry of fuel assembly. In this work the homogenized constants were applied to the WWER-440 core model developed by PARCS, using the deterministic lattice code HELIOS and the Monte Carlo code Serpent. Several parameters, such as k_{eff}, axial and radial power were compared.

1. INTRODUCTION

2D deterministic lattice codes like HELIOS, CASMO, WIMS, TRITON are the major mature analytical tools widely used both by industrial and regulatory communities for generation of the few-group macroscopic cross-sections and associated neutronics data (yields, fractions of delayed neutrons, precursor decay constants, inverse velocities etc.) for nodal diffusion codes. Availability of high performance computational infrastructure, capacity of handling big memory footprints and massive parallelisation feature of Monte Carlo method allowed to create new generation of Monte Carlo codes for reactor physics applications that have some fundamental advantages over deterministic lattice codes:

- use of continuous energy neutron-nucleus interaction cross-section libraries instead of multi-group cross-section libraries in deterministic codes. This is a big advantage, because the self-shielding effects are automatically accounted for, without relying on various approximations used by deterministic codes [8];
- no-geometric limitations, allowing modelling of any 2D and 3D fuel assembly, group of assemblies (super-cell), reflector or entire reactor core;
- more precise modelling of the isotopic composition, especially close to the reflector regions. In deterministic analyses to calculate isotopic compositions in the fuel assembly due to axial burnup distributions, set of 2-D calculations are carried out for discrete burnup values. However, this method is acceptable only for the fuel nodes far from axial reflectors where there is no significant axial gradient of the neutron flux. Moreover, the neutron flux spectrum is softened near to the top and bottom nodes of the fuel assembly, due to predominantly leakage of fast neutrons and reflecting back of thermal neutrons, and this effect is not captured by 2D codes [1].

To evaluate maturity of the Monte Carlo reactor physics codes 2D deterministic HELIOS-2 [2] and 3D Monte Carlo Serpent codes [3] were used for the generation of the 2-group macroscopic cross-sections library and associated neutronics data for WWER-440 fuel. Selection of the HELIOS-2 code is due to its wide verification and validation for WWER type of fuel by both us [4] and internationally [5]. Cross-sections generated by HELIOS and Serpent were applied to the WWER-440 reactor core model.

2. COMPARATIVE ANALYSIS OF THE CROSS-SECTIONS

As mentioned earlier Monte Carlo reactor physics codes requires high computational power and big memory footprint, therefore the burnup of the WWER-440 fuel assembly was limited up to 20 MWt*day/kgU with following discretization: 0, 0.15, 1, 3, 6, 10, 12, 14, 16, 18, 20. To properly take into account axial moderator density gradient influence on the cross-sections 3 history cases were set up corresponding to the WWER-440 reactor core coolant inlet, average and outlet temperatures at steady state operation (see Table 1). For each history 27 branching points were used (see Table 1):

TABLE 1. CROSS-SECTIONS PARAMETRISATION

Moderator density, g/cm^3	Boron concentration, ppm	Fuel temperature, K	Moderator temperature, K
	History		
0.74424	500	900	561.5
0.69729	500	900	561.5
0.7823	500	900	561.5
	Brunches		
0.74424	500	900	561.5
0.50000	500	900	561.5
1.00000	500	900	561.5
0.50000	0	900	561.5
0.50000	1000	900	561.5
0.50000	2000	900	561.5
0.74424	0	900	561.5
0.74424	250	900	561.5
0.74424	750	900	561.5
0.74424	1000	900	561.5
0.74424	15000	900	561.5
0.74424	2000	900	561.5
1.00000	0	900	561.5
1.00000	1000	900	561.5
1.00000	2000	900	561.5
0.50000	0	450	561.5
0.50000	0	2500	561.5
0.74424	0	450	561.5
0.74424	0	2500	561.5
1.00000	0	450	561.5
1.00000	0	2500	561.5
0.50000	0	900	420
0.50000	0	900	600
0.74424	0	900	420
0.74424	0	900	600
1.00000	0	900	420
1.00000	0	900	600

In the HELIOS-2 (version 2.0.01) model 1/12 symmetry of the WWER-440 fuel assembly with uniform 3.6% enrichment was taken into account (see Fig. 1). The CCCP (current coupling and collision probabilities) neutron transport equation solution method of HELIOS-2 was used in the model. 49 group neutron and 18 group gamma group cross-sections library [2] was used in HELIOS calculations.

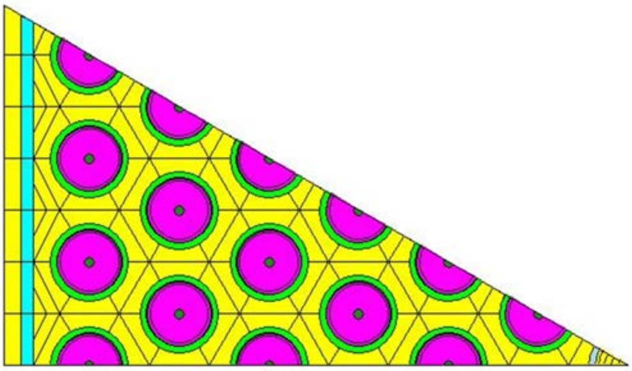

FIG. 1. WWER-440 fuel assembly model developed by HELIOS-2 code.

Fuel assembly model consists of space elements that are coupled with each other and with the boundaries by interface currents, while the properties of each space element are obtained independently. Furthermore, space elements subdivided into flat-flux regions and their periphery subdivided into flat-current straight-line and arc segments. Inside these regions, the cross-sections are assumed to be constant. K=3 coupling order was used in the model which is the best practice among WWER HELIOS community [6].

In Serpent, a full model of the WWER-440 fuel assembly was developed (see Fig. 2). A total of 891 Monte Carlo criticality simulations were run, using the automated burnup sequence of Serpent, with twenty million neutron histories per case.

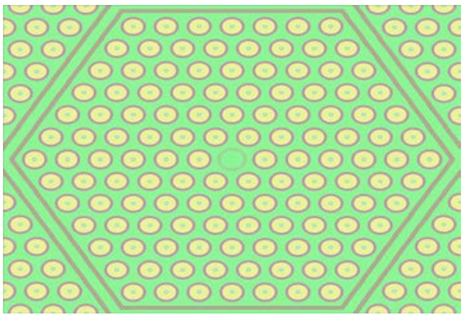

FIG. 2. WWER-440 fuel assembly model developed by Serpent code.

The Serpent calculations were run using the continuous energy ACE format cross-section library based on ENDF/B-VII data files. The B1 methodology was adopted in order to obtain cross-sections consistent with HELIOS. In this work 2.1.28 version of Serpent was used for the two-group cross-section data generation.

In both models a) the thermal cutoff energy is 1.84 eV to ignore upscattering; b) fission yields and delayed neutron data were calculated according to the recommendations of the [7] to be consistent with PARCS code; c) the cross-sections were generated using reflective boundary conditions.

In the Table 2, as an example, differences between HELIOS-2 and Serpent calculated B1-corrected k_{inf} values for Hystory-1 reference case are shown. The maximum difference is 250 pcm. For brunches difference varies between 0.67pcm to 587pcm.

TABLE 2. DIFFERENCES IN K_{INF}

Burnup, MWt*d/kgU	B1_Kinf-Serpent	B1_Kinf-Helios	Difference, pcm
0.15	1.23965	1.23582	250.00
1	1.22566	1.22259	204.87
3	1.2047	1.20213	177.46
6	1.17141	1.16875	194.29
10	1.12919	1.12702	170.51
12	1.11013	1.10781	188.65
16	1.09189	1.08965	188.27
18	1.07434	1.07236	171.86
20	1.05826	1.05583	217.48

Fig. 3-4 show percentage of relative differences between HELIOS and Serpent calculated 2-group transport, absorption, fission, ν-fission, κ-fission macroscopic, Xe and Sm microscopic cross sections and ADFs at different burnups for Hystory-1 reference case.

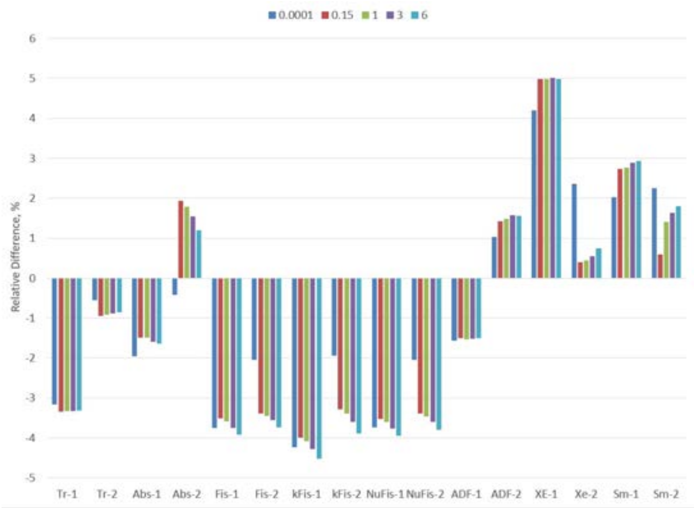

*FIG. 3. Percentage of relative differences between HELIOS and Serpent calculated cross-sections and ADF at 0, 0.15, 1, 3, 6 MWt*d/kgU burnups.*

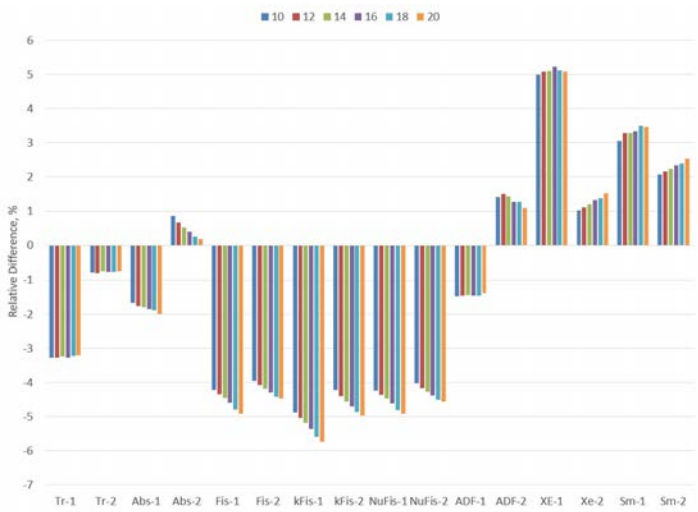

*FIG. 4. Percentage of relative differences between HELIOS and Serpent calculated cross-sections and ADF at 10, 12, 14, 16, 18, 20 MWt*d/kgU burnups.*

As we can see, for the cross-sections relative difference varies within 1-6%. Differences in the transport cross-sections is due to simplified approach (out-scatter approximation) [8] used in Serpent in calculating transport cross-section which is prone to significant errors for light nuclides like, hydrogen bound in water. For macroscopic absorption, ν-fission, κ-fission, fission and microscopic Xe and Sm cross-sections there is clear tendency of increasing discrepancies between HELIOS and SERPENT with increasing of the burnup. Here potential reason could be absence of leakage (B1) correction to the transmutation cross sections during burnup calculation [8]. This could significantly influence on isotopic composition of burned fuel since fuel depletion is strongly dependent on spectral changes.

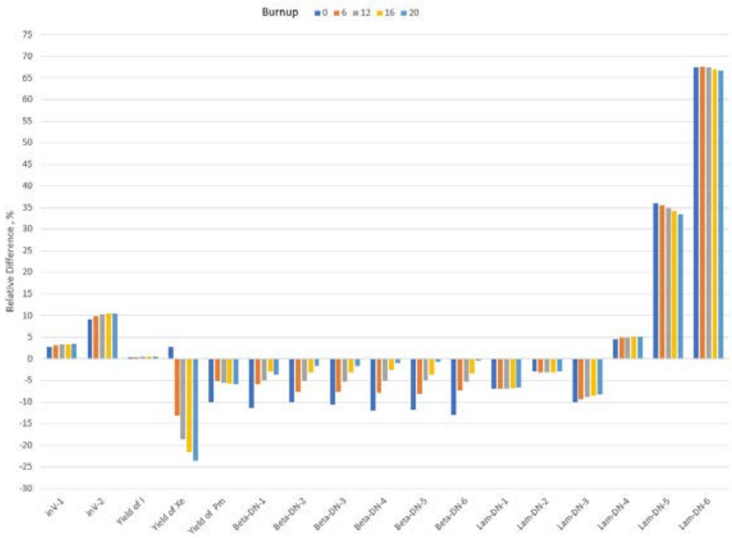

FIG. 5. Percentage of relative differences between HELIOS and Serpent calculated auxiliary neutronics data.

k_{inf} differences could be explained by approximations [8] used in B1 leakage correction in case of using reflective boundary condition in Serpent as well as by above mentioned reasons.

Fig. 5 shows percentage of relative differences between HELIOS and Serpent calculated inverse velocities, fission yields of I, Xe and Sm as well as delayed neutron fractions and precursors decay constants.

Differences in delayed neutron fractions and precursors decay constants mainly are due to methodological differences of their calculations. It's obvious that application of Serpent for transient analysis could bring significant errors due to big discrepancies, especially for Xe fission yield and delayed neutron data.

3. TEST OF CROSS-SECTIONS WITHIN FULL CORE MODEL

HELIOS and SERPENT cross sections were applied to the WWER-440 reactor core model. Since WWER-440 core has 60-degree rotational symmetry in the PARCS [9] model 61 fuel types (59 fuel and 2 reflector types) were modeled. Fuel types' radial configuration is presented in Fig. 6.

White nodes correspond to fuel assemblies, yellow and red nodes to control assemblies, blue nodes to reflector. Each of fuel types axially subdivided into 43 axial nodes (compositions). Top and bottom axial nodes belong to corresponding top and bottom reflectors.

PARCS model coupled to the RELAP model via PVM. The coupling of the PARCS neutronic nodes to RELAP thermal-hydraulic nodes was accomplished through the assignment of mapping weights between the respective nodes. These mapping weights, with values between 0 and 1, determine the distribution of neutronic power in the thermal-hydraulic and heat structure components, as well as the calculation of thermohydraulic feedback in the neutronic nodes. Mapping weight consists from radial and axial weights. Since in the RELAP and PARCS core models the same radial nodalization has been used for each axial level, in the core radial weights are equal to 1 for all nodes. Taking into account axial nodalization of PARCS and RELAP models axial weights were calculated as each RELAP node coupled to 4.1 PARCS nodes. For top and bottom reflectors, axial weights equal to 1 has been used. Entire set of mapping weights is constructed by multiplying each radial weight by each axial weight.

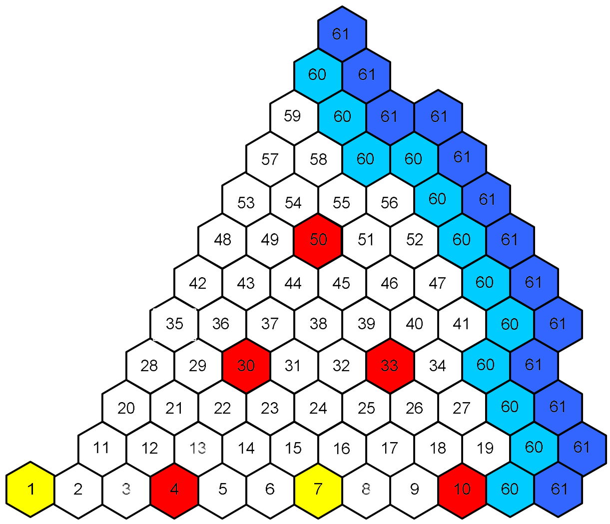

FIG. 6. Assembly types used in ANPP core model.

Radial burnup distribution of the core is shown in the Fig. 7.

PARCS calculated k-eff values using HELIOS and SERPENT generated cross-sections are in good agreement: difference is 161.7 pcm.

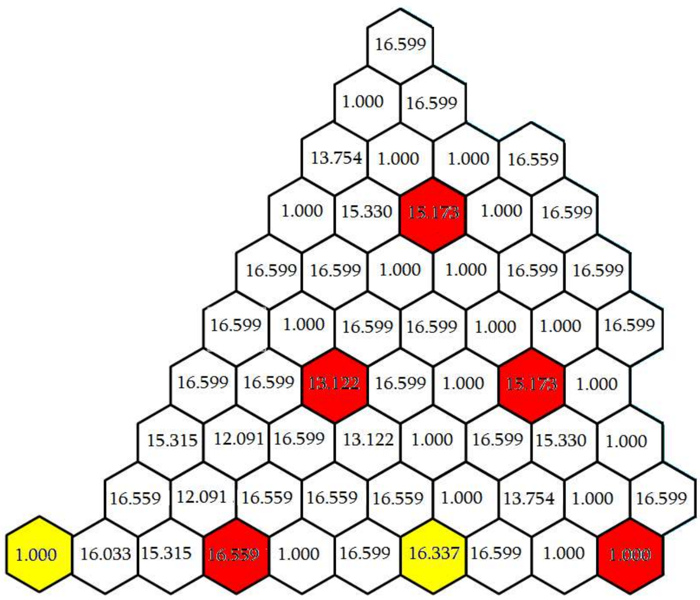

FIG. 7. Radial burnup distribution of the core.

4. CONCLUSIONS

Helios and Serpent 2-group cross sections shows fair agreement, so SERPENT generated could be used for WWER-440 reactor core steady state analysis. However due to large discrepancies in delayed neutron data and fission yields of important poisons application of the Serpent cross-sections for RIA analysis could be questionable. Full core calculations have shown good agreement on k_{eff}.

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ADVANCED MODELLING AND ANALYSIS CODES

Chairperson

S. MONTI
IAEA

A SAFETY FUEL ELEMENT ASSESSMENT BY USING NEW MATERIALS AND ADVANCED MODELLING TOOLS

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Abstract

After the accident occurred at Fukushima Daiichi nuclear power plant (NPP), a widespread awareness emerged in the nuclear field community that revealed the urgent need to develop fuel designs much more resistant to extreme accident conditions and capable to limit the magnitude of its harmful consequences. In order to fulfil these requirements, a worldwide initiative is underway, aimed to develop and implement the needed Accident Tolerant Fuel (ATF) concepts. The BaCo code was developed to simulate the nuclear fuel rods behaviour under irradiation and dry storage conditions including statistical analysis and 3D extensions. Research on new fuels and cladding materials properties based on *ab initio* and Multiscale Modelling of Materials (M^3) are developed to be included in the BaCo modelling. This work studies the whole life of fuel elements, during and after irradiation, and the benefits of implementing the M^3 calculations in the behavioural code. The evaluation of an ATF with a silicon carbide (SiC) cladding is the considered example with a focus on the analysis on the fuel safety.

1. INTRODUCTION

As a consequence of the accident produced in the Fukushima Daiichi nuclear power plant, leading to huge radioactive release in 2011; it was considered essential to implement mayor improvements in nuclear safety not only during operational conditions but also in case of a less frequent accident occurrence. In order to do that, the Accident Tolerant Fuels (ATF) initiative arose as an answer to develop more resistant fuel elements to support the first steps of a severe accident.

Simulation and modelling of the fuel behaviour at different operational conditions are essential in order to improve fuel designs and nuclear security. They allow evaluating critical conditions without expending too much time and money in the performance of expensive experiments. First principles methodology, based on density functional theory is implemented through the Quantum Espresso code, as a theoretical complement to support these calculations. It is also meant to determine the predictive ability of this code to adequately describe the studied material's mechanical and thermal properties.

This paper aims to show how a computational tool as the BaCo code could help improve the everyday management of a nuclear power plant (NPP).

2. THE BACO CODE

The BaCo code ("BArra COmbustible" -Spanish expression of "Fuel Rod"-) was developed at CNEA ("Comisión Nacional de Energía Atómica" – Atomic Energy National Commission of Argentina) to simulate nuclear fuel rods behaviour under irradiation conditions [1, 2] and at dry storage conditions [3, 4]. BaCo focus on PHWR fuels as the CANDU [4, 5] and Atucha [6] ones but we keep a full compatibility with PWR, BWR, WWER and PHWR MOX, plus advanced, experimental, prototypes and/or non-usual fuels. At present CNEA are developing the CAREM reactor where the fuel element has a hexagonal array as the WWER fuels. BaCo was strongly involved in the original design and test of that innovative Argentinean PWR fuel and advanced PHWR fuels (as PHWR MOX and the CARA ones).

BaCo assumes azimuthal bi-dimensional symmetry in cylindrical coordinates for the FR ("Fuel Rod") [1]. Although angular coordinates are not considered explicitly, angular dependent phenomenon, as well as radial cracking, are simulated through the angular averaging method [7]. Also, axial pellet cracking and relocation are included in BaCo. The hypotheses of axial symmetry and modified plane strains (constant axial strain) are used in the numerical modelling. The fuel rod is separated in axial sections in order to simulate its axial power profile dependence. Rod performance is numerically simulated using finite time steps (finite differential scheme). The modular structure of the code easily allows the description of phenomena observed in the UO_2 pellet and the Zry cladding behaviour. The current version of BaCo can be applied to any geometrical dimensions of cylindrical fuel rods mainly with UO_2 pellets (either compact or hollow, with or without dishing) and a Zry cladding.

In addition, it is suitable for the inclusion of advanced structural and fuel materials of the fuel rod, at least for the present version of the code, with an illustrative and comparative purpose. At present, the modelling of materials as metallic uranium, uranium carbide, uranium nitride (for pellets) and silicium carbide, FeCrAl (for claddings) are under development [8, 9].

The details of the mechanical and thermal treatment and the pellet, cladding and constitutive equations are available in reference (1) and an extended description of the code is included in reference [2].

The BaCo code was a participant of a series of Coordinated Research Projects ("CRP") of the International Atomic Energy Agency ("IAEA") generically named FUMEX ("Fuel Modelling at Extended Burnup"). The D-COM and the CRP FUMEX I, II & III were a series of comparison among experimental irradiations and code calculations. At present, we are involved in the IAEA's CRP ACTOF (CRP on "Analysis of Options and Experimental Examination of Fuels for Water-Cooled Reactors with Increased Accident Tolerance").

The code includes additional tools as the software package for finite elements 3D calculations and the statistical analysis for advanced fuel designs by taking into account the as fabricated fuel rod parameters and their statistical uncertainties. BaCo allows the calculation of a complete set of irradiations as for example the calculation of a full reactor core. BaCo 3D tools [10], statistical analysis [11], full core calculations [6] and graphical data post-processing improve the code performance and the analysis of the calculations [2].

Although the BaCo code uses a quasi-two-dimensional approach, the use of several three dimensional (3D) finite element features allow a complementary analysis of 3D properties, as for example the stress-strain state at a specific period of time during the irradiation [10]. The BaCo code results were enhanced by using "ad hoc" tools developed at the MeCom and SiM³ Divisions (Bariloche Atomic Centre, CNEA) [12].

For a better understanding of the uncertainties and their consequences, the mechanistic approach must, therefore be enhanced by the statistical analysis [11]. BaCo includes a probability analysis within their code structure covering uncertainties in fuel rod parameters, in the code parameters and/or into the fuel modelling taking into account their statistical distribution. As consequence, the influence of some typical fabrication parameters on the fuel cycles performance can be analyzed. It can also be applied in safety analyses and economics evaluation to define the operation conditions and to assess further developments. These tools are particularly valuable for the design of nuclear fuel elements since BaCo allows the calculation of a complete set of irradiations.

3. FIRST PRINCIPLES METHODOLOGY

For non-traditional fuels, in particular ATF and Generation-IV fuels, available data for the development of new materials can be obtained through the Multi-Scale Model of Materials (or M^3), a methodology that Provide the theoretical approximation to the modelling of the properties of the Materials through ab initio methods, molecular dynamics, Monte Carlo Kinetic and finite element calculations.

Silicon carbide is presented as a large family of crystalline structures called "polytypic ". This manifests the ability of this compound to crystallize in numerous modifications that can be described as different stacking sequences of the same unit layer. Most of the polytypic are modifications of the alpha phase (hexagonal structure). Nevertheless, for reasons of nuclear application, we were interested in starting with the study of the β phase (cubic structure) that forms below 1700°C.

3.1. M^3 Methodology applied to β-SiC

Elastic Constants

The silicon carbide beta phase (β-SiC) corresponding to the 3C polytypic has the cubic zinc blende crystal structure (F-43m space group).

The elastic constants determine the elasticity and mechanical stability of the crystals. For small deformations, within the elastic range, one expects a quadratic dependence of the energy of the crystal E with the deformation (Hooke's Law).

Table 1 compares the results of the constants and elastic modulus obtained in our work in comparison with those previously published by other authors.

TABLE 1. COMPARISON OF THE ELASTIC CONSTANTS OBTAINED BY DIFFERENT AUTHORS. WITH THE EXEPTION OF THE POISSON DIMENSIONLESS COEFFICIENT (V), YOUNG'S MODULUS (Y), SHEAR MODULUS (G) AND BULK MODULUS (B) ARE EXPRESSED IN UNITS OF GPA

	C_{11}	C_{12}	C_{44}	B	Y	G	ν
Present work	376.35	121.38	257.65	206.37	317.15	194.26	0.142
Ref. [13]	390.1	142.7	191.0	225.1	313.6	123.7	0.268
Ref. [14] (CASTEP)	420	132	267	228		208	
Ref. [15] (FP-LMTO)	420	126	287	223			0.231
(Exp.) Ref. [16]	390	142	256				
(Exp.) Ref. [17]	379	141	252				

3.2. Thermal properties

3.2.1. *Phonon density of states*

The phonon dispersion curves have no imaginary frequencies, showing its dynamic stability, in agreement with the results obtained by the analysis of the elastic constants. In Fig. 1, it can be observed that at low frequencies (below 20THz) the partial density of phononic states is dominated by the Si atoms as the acoustic modes originate mainly from the heavier elements, and above 20THz the phononic modes originate mainly by C atoms. It can also be noticed the presence of a phonon gap around 18-22 THz which is in good agreement with previous results [18].

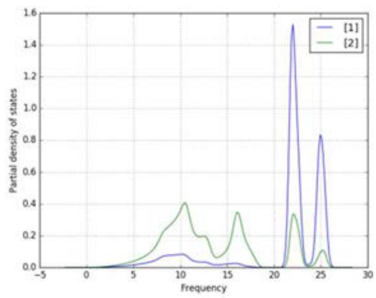

FIG.1. Partial phononic density of states. [13] C atom, [14] Si atom. Frequency in THz units was calculated from the primitive unit cell.

3.2.2. *Specific heat*

For many purposes, an efficient method to calculate the temperature dependent properties of an anharmonic crystal is the quasi-harmonic approximation (QHA). Within this approach, the anharmonic lower order corrections are considered, allowing the interatomic force constants and phonon frequencies to depend on the volume. In order to calculate the specific heat at constant volume, only the harmonic frequencies are needed, which were determined for the lattice in equilibrium.

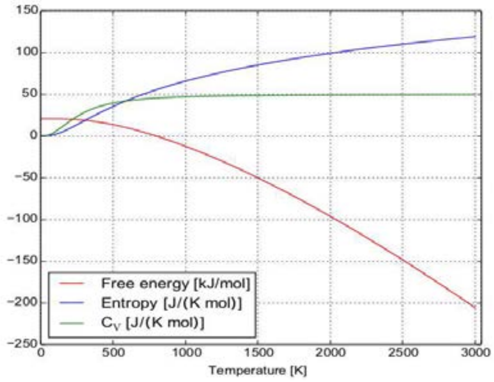

FIG. 2. Entropy (SV), Helmholtz free energy (FV) and specific heat (CV) as a function of temperature.

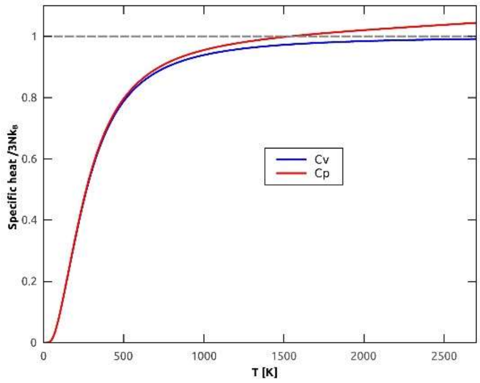

FIG. 3. Specific heat calculated as a function of temperature. The red line corresponds to the dimensionless quantity CV/3Nk$_B$ while the blue line CP/3Nk$_B$, where N is the number of Avogadro and k$_B$ is the Boltzmann constant. The horizontal line indicates the Dulong-Petit limit.

Fig. 2 and 3 show the calculated thermal properties C_V, S_V and F_V, using the harmonic approximation, while in order to calculate C_P the quasi-harmonic approximation (QHA) was used. The results obtained for the vibrational entropy and specific heat at constant pressure are in good agreement with the work Refs. [18], [19], [20] and [13], but the work of Lee et al. [19] who used the VASP code, and Thakore et al. [20] who used the Quantum Espresso code, have a difference between the specific heats that appears to be a little more pronounced. Since the solids are assumed practically incompressible, values for C_P and C_V are almost equal and therefore a single value for the specific heat is used. However, with increasing temperature, their difference is no longer small.

3.2.3. *Linear thermal expansion coefficient*

The linear thermal expansion coefficient, denoted α_L, is obtained theoretically as:

$$\alpha_L = \left(\frac{d\, lna}{dT}\right)_P, \tag{1}$$

where T is the temperature and “a” is the lattice parameter. Therefore, a very good approximation is given by:

$$\alpha_L = \frac{1}{a_0}\left(\frac{da}{dT}\right)_P, \tag{2}$$

where a_L is the lattice parameter and a_0 is the lattice parameter at the reference temperature T_0 =300K.

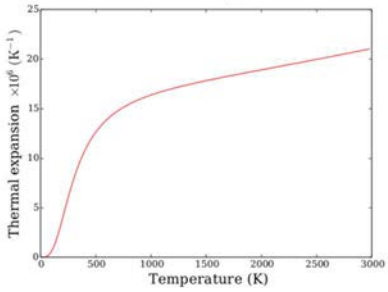

FIG. 4. Temperature dependence of β- SiC phase linear thermal expansion coefficient vs. temperature.

4. BASIC DEVELOPMENT OF ATF’S (ACCIDENT TOLERANT FUELS)

Currently, the nuclear industry has begun to require the development of new fuels that are resistant to severe accidents and do not imply a reduction in their performance. New materials and new designs of fuel elements converge in the ATF initiative. We analyse the physical properties of the proposed materials for fuels and claddings. In addition, the parameters of these materials are obtained by "ab initio" and M³ methods, in particular for the SiC, which are incorporated into the BaCo and BaCo3D codes for a first approximation of the ATF simulation under irradiation.

4.1. Performance Analysis of an ATF

The relevant calculations were performed with a version of BaCo which includes the properties of SiC calculated by ab initio methods in the previous section in order to obtain a new approximation about its use as cladding material (i.e. a type SiC/UO_2 fuel rod) and continue the comparison of the behaviour relative to Zry-4/UO_2 and FeCrAl/UO_2. The results obtained in this new version with the support of the basic physics associated with the modelling, are equivalent with the parameters from bibliography.

As an example of these calculations, in Fig. 5, is represented the central temperature of the fuel pellet of UO_2 as a function of the burnup, where is reached the maximum temperature recorded in the FE ("Fuel Element") at a steady irradiation at 250 W/cm. The reduction in temperature for the case of Zry-4 from ~3 MWd/kgU is due to the PCI ("Pellet-Cladding Interaction") which improves the thermal conductivity and the transfer of heat from the pellet to the coolant. In Fig. 6. we see the evolution of the pellet and clad radii using a SiC cladding, obtaining PCI to EOL ("End Of Life").

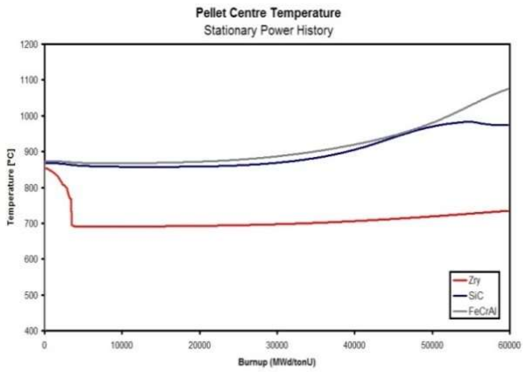

FIG. 5. Central temperature of the UO_2 fuel pellet under steady state irradiation (200 W/cm) using three different cladding materials.

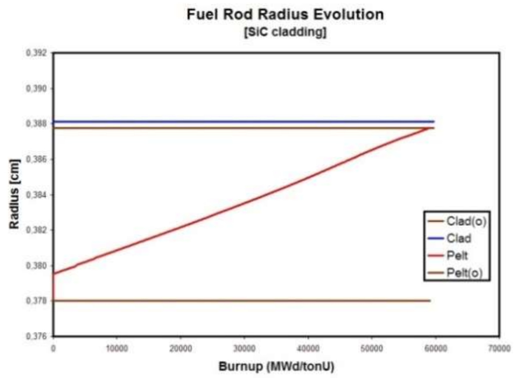

FIG. 6. Evolution of the outer radii of a UO_2 pellet and internal of a SiC cladding.

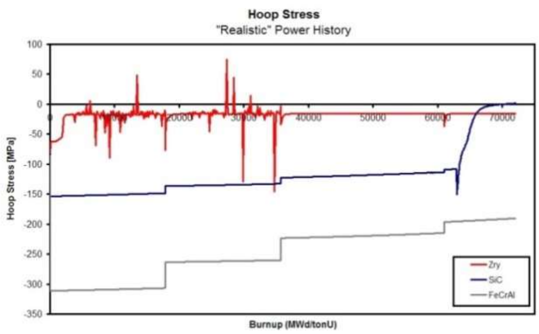

FIG. 7. "Hoop stress in Zry, SiC y FeCrAl claddings. CRP FUMEX II, Case 27(2d), provided by FANP.

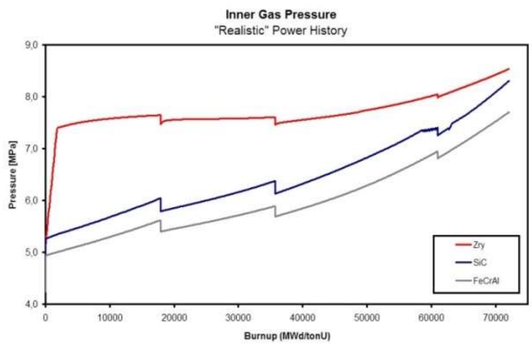

FIG. 8. Internal pressure produced by the fission gases in the fuel rods with Zry, SiC y FeCrAl claddings.

A more realistic example corresponds to the "hoop stress" in Fig. 7, where the irradiation corresponds to a realistic simplification (generic, conservative PWR power history, already very high burnup) of a commercial irradiation that was provided by FANP ("Framatome Advanced Nuclear Power, Inc.") for its study (Case 27.2d) within the FUMEX II (Coordinated Research Project on Fuel Modelling at Extended Burnup II) of the IAEA, where BaCo participated [21, 22]. The modelling starts at an average power of 350 W/cm followed by 4 power cycles at a decreasing level. The irradiation time is from ~1700 days to an extraction burn of ~70 MWd/kgU. Fig. 8 shows the curves of the internal pressure of the free gases in the sheath (filling gases -He and gaseous fission products, Xe and Kr-). Here we have coupled all the aspects of calculation with the code BaCo, ie, mainly, the thermal and mechanical aspects because the temperature and evaluation of the free space inside the sheath are relevant. The highest-pressure value is obtained with the use of Zry due to the PCI that eliminates the free spaces between the pellet and the cladding mainly due to the creep down of the cladding and thermal expansion of the fuel pellet.

To illustrate, we repeat the simulation calculation of a fuel under dry storage conditions by using a WWER 440 fuel rod type used as a control case in the FUMEX II CRP (Case 9), irradiation in the Kola 3 NPP, BC 007 of the FA 222) [22] and assuming a SiC cladding. Fig. 9 shows the same agreement with the previous calculations and the same general observations. It is evident that, if using a SiC cladding, it will be necessary to ensure that its gap remains open, to avoid PCI. In order to prevent a failure of a solid tube of SiC we use a cladding type SiC/SiC (inner solid tube covered with fibres). Fig. 10 shows the evolution of the gap in which PCI is reached to EOL and we find the gap opened when passing to storage conditions.

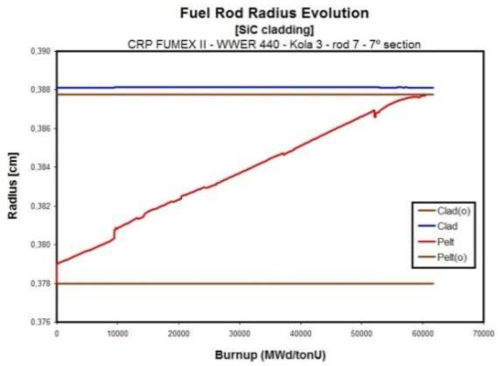

FIG. 9. *Evolution of radii of the pellet and the cladding in the most required axial section for a SiC fuel rod cladding.*

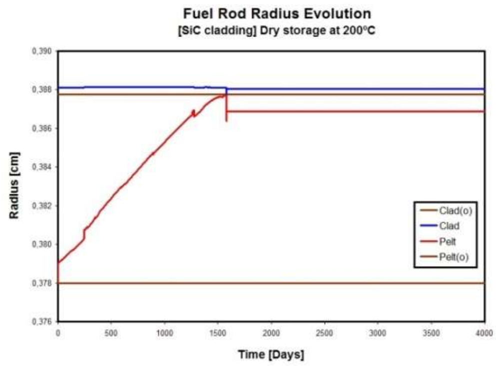

FIG. 10. *Evolution of radii of the pellet and the cladding during irradiation and in dry storage conditions at 200ºC. SiC cladding. Case 9 of the CRP FUMEX II of IAEA.*

The Fig. 11 shows the gas pressure inside the fuel rod during the irradiation. In order to enhance the previous analysis of the behaviour under dry storage conditions it is included in the Fig. 12 the pressure outside the fuel rod. That pressure is the coolant one during the irradiation at the nuclear reactor. The fuel rod after irradiation is stored at normal pressure (1 atm) and a temperature of 200ºC is estimated. The gas pressure is lower than the pressure of the coolant during all the time of the irradiation as it is required for safety reason.

The calculations plotted on Figures 5 to 11 were performed by using the data input corresponding to the nominal or as fabricated parameters of the fuel rod and the reactor. The as fabricated tolerances of the fuel parameters usually are not included in the code calculations. BaCo allows the calculation including the statistical dispersion produced during the fuel fabrication that is compatible with the tolerances defined for the fuel designers among other parameters. Several running codes are automatically programed into BaCo in order to perform a statistical evaluation of the fuel behaviour. Each running code is using a random input data set. Each parameter of each individual data set is estimated by take into account the as fabricated tolerance and its statistical dispersion. Then each individual data set is representing a possible and real fuel rod.

Fig. 12. is including the previous nominal calculation and the statistical evaluation. The set of points plotted in Fig. 12. and its density shows the dispersion of the calculation. A smooth increment on time of the gas pressure was found including the stage at dry storage due to of the diffusion and release of the fission gases retained in the fuel pellets. The main difference is that the external pressure (the coolant pressure) during irradiation is compressive and a tensile state is done at dry storage conditions. The tensile stress on the SiC cladding could produce an unexpected crack or failure during storage.

Fig. 13. shows the hoop stress of the cladding (tangential stress at the inner surface of the cladding). A compressive and conservative tensile stress is found during irradiation and stress reversal is done at EOL. The positive value of stresses at dry storage shows a non-desirable condition.

Fig. 14 is the statistical evaluation with BaCo of the pellet centre temperature. The big dispersion at high burnup up to EOL is done due to the PCI or non PCI event during that step time.

The calculations were using the most demanding conditions and it is expected a more conservative result during an experiment environment.

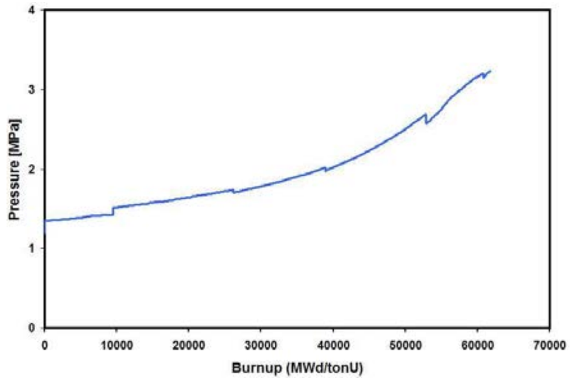

FIG. 11. Gas pressure inside the fuel rod during the irradiation.

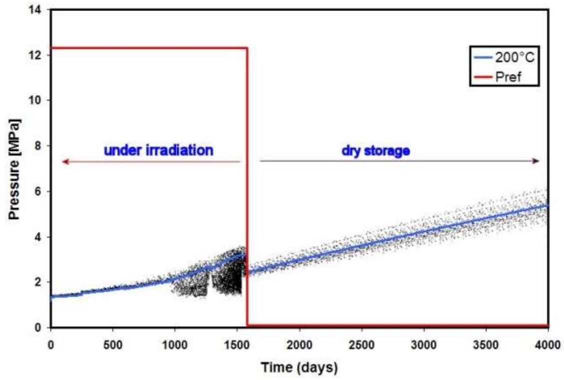

FIG. 12. Statistical analysis of the gas pressure calculation in the fuel rod including the boundary conditions (coolant pressure and ambient conditions) plus the nominal calculation and dispersion.

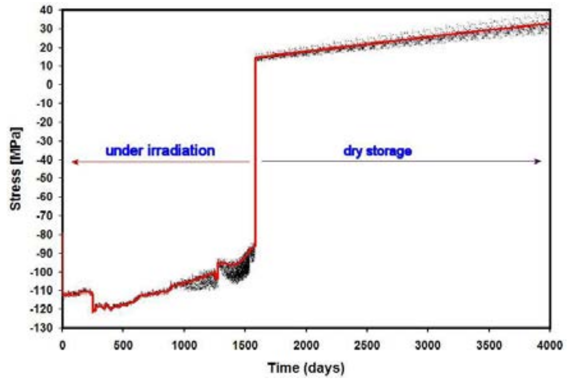

FIG. 13. Statistical analysis of the Hoop stress (tangential stress at the inner surface of the cladding). Nominal value and dispersion are plotted.

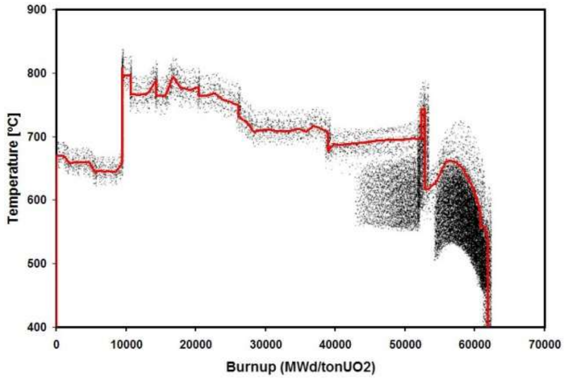

FIG. 14. Statistical analysis of the pellet centre temperature at the most demanding axial segment of the fuel rod. Nominal value and dispersion are plotted.

5. FINITE ELEMENT METHOD (FEM) EXTENSIONS

5.1. BaCo (~2D) and BaCo3D (FEM)

Optimization of a WWER fuel pellet with a central hole

One of the characteristics of the WWER fuel pellets is the presence, by design, of a central hole. This hole produces a reduction of tensions, which induces a reduction of the radial deformations and, finally, a reduction of the stresses between the pellet and the cladding. Finally, a conservative design result is obtained. On the other hand, this design causes a decrease in the inventory of fissile material which can approximately be compensated for by a somewhat larger pellet radius or possibly with a slight increase in enrichment (if this is possible and, the extra technical problems that this would entail).

Fig. 15 includes the comparison of the radial deformation axial profile of a fuel pellet. The reduction of this deformation is verified experimentally in Fig. 16, where it is shown how effectively the presence of that central hole causes a decrease in the radial deformations. It is interesting to remark that the data provided by the Halden Reactor Project used for the validation is not a specific experimental irradiation [23].

Currently, designers of fuel element of the WWER have the possibility of producing solid pellets, without a central hole and with a "dishing" [24]. This design change is still under discussion for the security reasons already discussed. Another parameter that is taken into account is the proposed diameter for the central hole of this type of pellets (see Fig. 15). Fig. 17 includes a set of maps representing the radial deformation of a WWER fuel pellet as illustrative purposes of the BaCo3D capabilities.

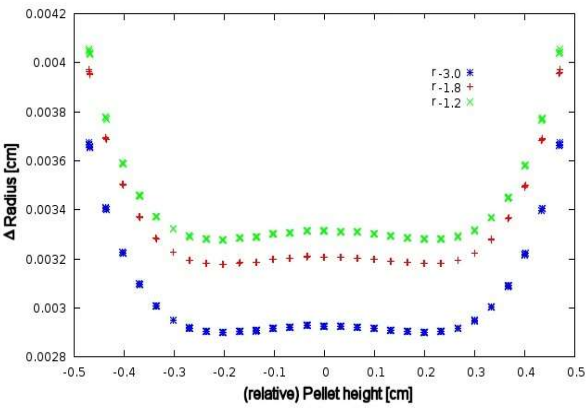

FIG. 15. Radial deformation of irradiated WWER type pellets in equivalent situations by varying the diameter of their central hole (Ø = 1.2, 1.8 and 3.0 mm).

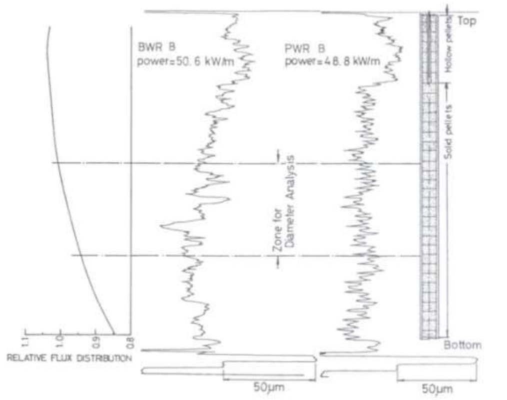

FIG. 16. The power profile of an experimental FR of the Halden Reactor. At the top of the FR are the pellets pierced with thermocouples. (HRP-305/8, [23]).

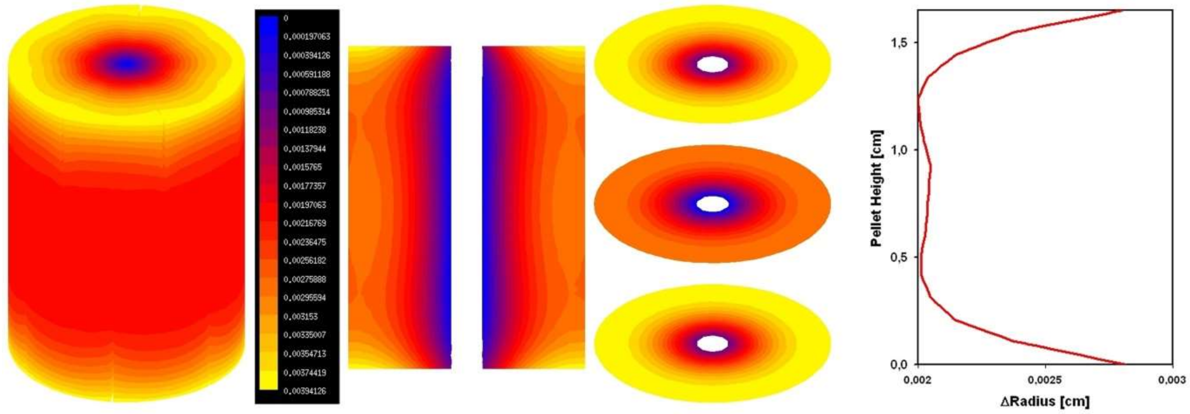

FIG. 17. Radial deformation of a WWER fuel rod with a central hollow in the pellet. Different graphical ways to represents them with an "ad hoc" software developed in junction with BaCo3D.

6. NEUTRONIC CONSIDERATIONS ABOUT ATF DEVELOPMENT

In agreement with previous results, all new materials proposed for the ATF initiative need to revaluate its neutronic behaviour. First studies performed at unit cell level, showed that the reactivity of the core is highly reduced when using FeCrAl alloy for example [25]. A more detailed analysis of this subject needs to be done to estimate reactivity coefficients and a complete core calculation to see which other parameters are affected.

7. CONCLUSIONS

Different capabilities of the BaCo code to perform a complete analysis of the fuel element were shown. The importance of these types of analyses relies on the feedback that an operation authority could have in case of different engineering problems that could appear during the life plant.

In accordance with the analysis, the methodology of M³ is essential to support behavioural codes to obtain proper values of the properties of new materials. The experimental data needed for fuel simulation and ATF designs can be partially supported by using M³. The results obtained for the SiC has enough accurate as to be included in the BaCo modelling. As a conclusion of the presented calculations of SiC it was shown the needs to prevent PCI and the formation of microcracks during irradiation which could grow and present fission products leakage during storage. In the same way, it was clearly shown how a fuel pellet designed with a central hole will reduce stresses and deformations in the pellet. Those results were validated by using data provided for the Halden Reactor Project and it is interesting to mention that the experimental data were extracted from an experiment of irradiations not related with the main issue of our study.

Improvements on the tools that support BaCo code, as FEM method, statistical analysis and the inclusion of material parameters obtained with M^3, could increase the information that it can be obtained and can help to visualise in a simpler manner the fuel rod and the fuel element as a hole during irradiation and dry storage. Once all these tools are tuned and benchmarked, new improvements can be done and new designs that can enhance a safe operation of the NPP.

8. AKNOWLEDGEMENTS

To the IAEA that give us the opportunity to present our work, Halden Reactor Project that kindly provided the valuable data used in this paper and the MeCom Department of CNEA that provided the basic engine for the FEM calculations included in BaCo3D.

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GERMAN EXPERIMENTAL ACTIVITIES FOR ADVANCED MODELLING AND VALIDATION RELATING TO CONTAINMENT THERMAL HYDRAULICS AND SOURCE TERM

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Abstract

Severe accidents in light water reactors are governed by thermal hydraulic processes influencing safety relevant phenomena which may challenge the containment integrity and define the radiological source term. Among the most relevant phenomena coupled with containment thermal hydraulics are the formation and distribution of flammable hydrogen/air mixtures and the fission product (aerosols, iodine) behavior. International efforts are focused on further development of computational methods and tools to simulate these phenomena to provide a reliable background for Severe Accident Management. In this context, essential contributions are made by German institutions JÜLICH in co-operation with RWTH Aachen University, Becker Technologies, and AREVA in the framework of ambitious national and international projects. The facilities involved provide high quality data bases for advanced code development and validation. Current experimental activities include investigations on gas distribution with particular emphasis on light gas stratification formation and break-up by natural convection flows. Furthermore, the operation of passive auto-catalytic recombiners (PARs) which play a central role in the mitigation strategy of flammable gas mixtures is studied under specific severe accident conditions. In order to enhance the database for source term assessment, test programs are focusing on iodine multi-compartment behavior and pool scrubbing. The paper provides an overview of the German experimental programs and the involvement in ongoing national and international projects. The discussed topics will be concluded with future perspectives.

1. INTRODUCTION

Severe accidents in light water reactors are governed by thermal hydraulic processes influencing safety relevant phenomena which may challenge the containment integrity and define the radiological source term. The formation and distribution of flammable hydrogen/air mixtures and the fission product (aerosols, iodine) behavior are among the most relevant phenomena which are coupled with containment thermal hydraulics. In order to provide a reliable background for Severe Accident Management, international efforts are focused on the further development of computational methods and tools to simulate these phenomena. German institutions JÜLICH in co-operation with RWTH Aachen University, Becker Technologies, and AREVA contribute to ambitious national and international projects by providing advanced experimental facilities with high quality data bases for advanced code development and validation.

2. GAS DISTRIBUTION AND THERMAL HYDRAULICS

The containment of large pressurized water reactors (PWRs) has a volume of up to 70,000 m³ and is subdivided into different compartments, which house the components of the reactor cooling system and auxiliary systems. During the course of a design and beyond design-basis accident, large quantities of steam and hydrogen are released from the break location into the containment within a short time. Detailed knowledge of the spatial distribution of gases and the atmospheric conditions is essential, e.g. for the design of safety systems and accident management, and provides relevant information for subsequent analyses of possible hydrogen combustion loads and the radiological source term.

2.1. THAI

The technical-scale THAI (Thermal hydraulics, Hydrogen, Aerosols and Iodine) test facility is operated by Becker Technologies in close co-operation with AREVA, Erlangen, and GRS, Cologne. Details on the facility and on THAI experimental programs are given in [1, 2]. The main components of the THAI+ facility are two cylindrical stainless steel vessels connected by DN500 piping at top and bottom, the THAI Test Vessel (TTV: 60 m³, 9.2 m height, and 3.2 m diameter) and the new vessel called Parallel Attachable Drum (PAD: 17.7 m³, 9.73 m height, and 1.6 m diameter), with sump compartments at the lower end of each vessel (Fig. 1). Design boundary conditions of the test facility are 14 bar at 180 °C and its unique experimental features include use of hydrogen, iodine tracer I-123 and differential wall heating/cooling. The gas distribution experiments in THAI deal with build-up/dissolution of stratification and natural convection phenomena in containment atmospheres. In the experiments using steam, two-phase phenomena are introduced which influence the flow behavior by condensation and/or evaporation in the gas atmosphere or on the vessel walls. The analytical work and code benchmarks performed based on THAI containment thermal-hydraulics and gas distribution experiments resulted in an improvement of the LP and CFD codes and provided valuable experience to code users [1, 2]. As an example, in the HM experiments, transferability of helium results to hydrogen distribution problems in containments was successfully demonstrated.

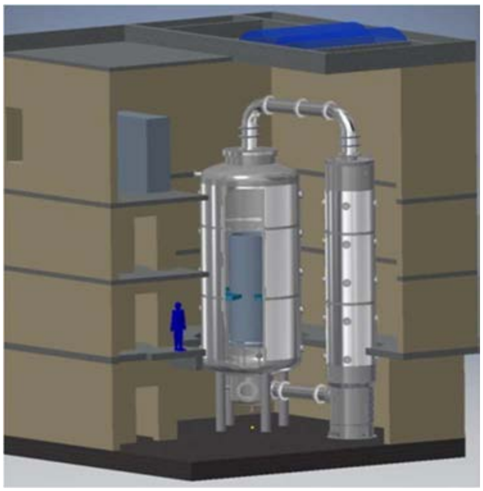

FIG. 1. THAI^{+} test facility.

2.2. SETCOM

In the context of buoyant flows, condensation plays a major driving role in containment thermal hydraulics. The SETCOM test facility (Fig. 2) at JÜLICH was designed to provide data for the development and validation of wall condensation models for coarse-mesh CFD (Computational Fluid Dynamics) application [3]. The facility is a closed loop which consists of a rectangular flow channel (0,44 x 0,44m) with three adiabatic walls and one cooling plate with a length of 6 m. The whole SETCOM facility can be steplessly inclined from horizontal to vertical orientation at any preferable angle and allows investigating wall condensation under flow velocities from nearly zero up to 5 m/s. Besides classical point-wise measurements of local wall heat flux or temperature, the facility has a flexible optical access to investigate transport phenomena inside the transpired boundary layer by means of PIV (Particle Image Velocimetry) and LDA (Laser Doppler Anemometry) as well as combined Raman-Rayleigh Scattering. This data is used to develop dedicated wall functions, which can be applied for effective and reliable CFD modeling of containment flows. Recently, a first test series on forced convection conditions has been concluded and confirmed the measurement and post-processing concept. Currently, a comprehensive investigation on condensation under mixed convection conditions is on-going.

FIG. 2. SETCOM test facility.

3. PAR OPERATION UNDER SEVERE ACCIDENT CONDITIONS

Passive auto-catalytic recombiners (PARs) are installed inside the containments of nuclear power plants worldwide in order to remove hydrogen that may be released during a loss-of-coolant accident and to avoid possible threats related to fast hydrogen deflagration. Besides the removal of hydrogen, PARs contribute to the containment thermal hydraulics by inducing heat and flow patterns promoting atmosphere mixing. Recently, research is focusing on PAR operation under challenging severe accident conditions in order to further enhance the assessment of hydrogen mitigation efficiency.

3.1. REKO platform

JÜLICH operates several REKO facilities at different scales in order to study the processes inside PARs to provide a database for the validation of detailed mechanistic PAR models. The goal is to further enhance the reliability of numerical models describing the operational behavior of PARs under relevant boundary conditions [4]. By this, the REKO platform complements existing integral experimental programs, e.g. THAI (see chapter 3.2). The REKO-1 facility enables catalyst performance tests, e.g. start-up tests of samples contaminated with aerosols like cable fire soot. Furthermore, homogeneous gas phase ignition on hot catalyst sheets is investigated. The REKO-3 facility represents a full scale segment of a PAR catalyst section to be investigated under forced flow conditions. The gas mixture which is fed into the flow channel consists of different components well defined by means of mass flow controllers, pre-heater and steam generator. The comprehensive REKO-3 database for validation of numerical PAR models includes experimental data on hydrogen conversion and catalyst temperatures for a wide range of boundary conditions. In order to study the interaction of the catalyst section with the chimney, the REKO-4 facility consists of a 5.5 m³ vessel. The facility may be operated at a pressure of up to 2.3 bar at 280°C. The flow velocity field of the PAR inflow is measured by means of Particle Image Velocimetry (PIV) in order to avoid disturbing the inflow by measurement installations. The GRART facility enables the investigation of the effect of humidity on the start-up delay of catalyst sheets by determining the adsorption isotherms of water condensate on the catalyst surface. The on-going experimental program aims at solving open issues related to PAR modeling including PAR ignition, carbon monoxide conversion and poisoning, efficiency under oxygen-lean conditions, and the impact of aerosols [5].

3.2. THAI

The focus of PAR experiments conducted in THAI test facility is to provide a comprehensive database on the operational behavior of PARs under typical severe accident conditions. The experiments are generally conducted using commercially available PARs, e.g. from vendors AREVA, AECL (now CANDU Energy Inc.) and NIS in 60 m³ THAI test facility, and more recently in the extended 80 m³ two-vessel configuration of THAI$^+$ (Fig. 1). The large THAI vessel allows PAR operation with unrestricted natural convection which includes interaction of PAR performance and vessel atmosphere distribution. Therefore, THAI PAR experimental database is suitable for PAR specific modelling as well as for accident simulation by using containment codes. In addition to the data already gained from other facilities, e.g. BMC, H2PAR, KALI, REKO, experiments conducted in the THAI extended the knowledge base under severe accident conditions, such as PAR start-up behavior, PAR performance (H_2 recombination rate, H_2 depletion efficiency), influence of oxygen lean atmosphere on PAR performance, interaction with fission products (aerosols, iodine) and PAR induced ignition. The variation in test parameters included: Initial pressure between 1.0 bar and 3.0 bar, initial gas temperature between ambient and 117 °C; atmosphere steam content of 0-60 vol.%; variation in O_2 concentration between 0.5 and 21 vol.% (air); and PAR overload by high hydrogen concentration until ignition [1, 2]. In general, application of the THAI PAR experimental database allowed analysts to set-up or improve PAR models implemented in various codes based on LP, CFD and mechanistic approaches [6]. Based on THAI experiments, conditions under which a PAR acts as an ignition source for the combustible gas mixture present in the PAR environment (respectively in the test containment volume) could be defined. Recently, the experimental database has been extended for PAR performance under counter-current flow conditions and interaction between two operating PARs. Tests with carbon monoxide are foreseen in the near future.

4. IODINE AND AEROSOL BEHAVIOR

With regard to severe and beyond-design-basis accidents and potential consequences in terms of land contaminations, aerosol issues are of high relevance. In order to prevent long-term consequences for the environment and population as a result of severe accidents, filter containment venting systems (FCVS) are used in several nuclear power plants, mainly in Europe. One of the most relevant processes being conductive to source term mitigation is pool scrubbing, the particle retention in water pools.

4.1. THAI

THAI experiments related to iodine and aerosol issues cover a wide spectrum of accident scenarios and provide relevant data for validation and development of iodine and aerosol models implemented in the severe accident analysis codes. The adsorption and desorption of iodine on steel and painted surfaces under different thermal-hydraulic conditions in single- and multi-compartment volumes, the mass exchange of iodine between gaseous and liquid phases, interaction between iodine and airborne aerosol particles, dry and wet resuspension of aerosols from surfaces and water pools, iodine wash-off from wet paint, iodine and aerosols interaction with an operating PAR, formation of iodine oxides by reaction between iodine and ozone, and iodine release from a flashing jet are the phenomena and processes which have been investigated in the THAI tests conducted so far [7]. Based on THAI experimental results, important progress has been demonstrated during EU-SARNET2 code-benchmark exercise on modelling of iodine behavior in multi-compartment geometry. The analysis results highlighted the need for detailed thermal-hydraulic modelling to calculate the correct relative humidity and atmospheric flow rates. THAI multi-compartment iodine tests including painted surfaces have been recently conducted. In one of the tests, multi-compartment iodine behavior is studied in the presence of high relative humidity (without wall condensation) and with silver aerosol. All THAI tests are accompanied by model improvement and validation work e.g. in COCOSYS/AIM. Another focus area of THAI tests is on iodine and aerosol release from water pools under different thermal hydraulics, chemical boundary conditions and pool convection (stagnant/agitated) states. Experimental investigations cover both re-entrainment of pre-mixed aerosols/iodine from water pool and aerosol decontamination due to pool scrubbing related phenomena and processes. Re-entrainment of soluble and insoluble aerosols from boiling sump has been investigated in the previous THAI tests and recent tests extend the range of test parameters, e.g. effect of depressurization induced boiling as expected during operation of a filtered containment venting system. Experimental investigations related to aerosols pool scrubbing with specific focus on measurement of decontamination factor and bubble hydrodynamics are foreseen in the near future.

4.2. SAAB

FIG. 3. SAAB test facility.

The SAAB (Severe Accident Aerosol Behavior) test facility at JÜLICH provides a new database for validation and optimization of pool scrubbing models by performing systematic experimental research of decontamination factors and on the influence of different parameters on aerosol retention [8]. The test vessel consists of seven segments with an inner diameter of 1.5 m (Fig. 3). The aerosol can be injected into the lowermost segment in three different ways: upward, sideward and downward. Five identical segments with 1 m height each can be installed above the injection segment and filled with water. Each segment includes three inspection glasses (0°, 90°, 180°) enabling the observation of bubble formation and bubble rise velocities for the investigation of hydrodynamics with a high speed camera. The topmost segment is a hood to collect the remaining particles with minimized particle losses for taking isokinetic samples with ELPI+, SMPS, APS and filters to determine particle number concentration and particle mass concentration. Before entering the test vessel, the particles are mixed with gas inside a mixing chamber to generate aerosols consisting of different gas mixtures (nitrogen, helium, steam and air) and different particles (SnO_2, CsI, CsOH and Ag). By this, differences in retention of soluble and insoluble particles with condensable and non-condensable gases can be investigated. Furthermore, the effect of particle size and size distribution as well as different fluid dynamics conditions can be studied.

5. CONCLUSIONS

The German institutions JÜLICH in co-operation with RWTH Aachen University, Becker Technologies, and AREVA provide essential contributions to ambitious national and international projects in order to enable advanced modeling of containment thermal hydraulics and source term. The facilities involved provide high quality data bases for advanced code development and validation with increasing level of complexity.

The ongoing OECD/NEA THAI-3 project aims to further investigate issues specific for water cooled reactors under severe accident conditions related to fission product behavior, PARs and flame propagation in compartmentalized geometry. Source-term relevant experiments, like fission product release from hot water pool involving "pool scrubbing" related phenomena and resuspension of pre-deposited fission products by hydrogen deflagration ("delayed source term") will provide data on issues considered to be of high priority at international level in the light of the Fukushima accident.

In the framework of the SAAB II project, systematic investigation of particle retention in a water pool will be performed. The foreseen test matrix includes the experimental investigation of single effects and the validation of reliable correlations for the assessment of particle retention. Furthermore, the investigation of the effect of cable fire aerosols on the operation of PARs will be continued. The current SETCOM program includes a comprehensive investigation on wall condensation under mixed convection conditions.

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ENHANCED NUCLEAR ENGINEERING SIMULATORS

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Abstract

Engineering Simulation is a sophisticated multi-purpose technology allowing the users of simulators to run various engineering activities thanks to the possibility of representing the behavior of the plant under normal and adverse conditions, and of modifying the simulated plant architecture and components, adjust a huge set of parameters, test alternative operational solutions. Engineering Simulators have been developed worldwide for performing plant design, integrated safety analysis, verification and validation of systems and components and human factors studies. These tools also play a role in developing and maintaining key nuclear skills, as knowledge repositories for training at various levels of expertise. The 'ENES' Strategic Project (proposed by ENEA within NUGENIA framework) aims at designing in detail the most urgent improvements to the current generation of Engineering Simulators for Gen II – Gen III NPPs in order to define the key characteristics of a new generation of Enhanced Simulators that can respond adequately to the various issues raised by the Fukushima accident and to other key targets of the international agenda for nuclear plants safety.

1. INTRODUCTION AND OBJECTIVES

The 'ENES' Strategic Project (proposed by ENEA in the NUGENIA framework [1] and recently (October 2016) proposed as a Research & Innovation Project under the EURATOM call [2]) is to design in detail the relevant and agreed most urgent improvements to the current Engineering Simulators for Gen II – Gen III NPPs in order to promote the development in the short-medium term of a new generation of 'Enhanced' Engineering Simulators that respond adequately to the issues raised by the Fukushima accident and other key targets of the international agenda for nuclear plants safety. This objective will be achieved through the definition of an innovative simulation architecture including both 'classic' models categories and 'innovative' software models such as those for the simulation of extreme natural events and advanced tools for predicting the dispersion in the environment of radionuclides released in case of severe accidents and their potential effects on population. For each model category a few alternative solutions will be identified for giving a remarkable flexibility and applicability to the new architecture. In particular ENES will examine in detail the possibility of using - in each simulation area – one more complex and one simpler software solution, for allowing future users to select the heavier codes only for the activities where this is necessary, while keeping the calculation burden not excessive through the use of the simpler fast running options. The domain of analysis of the simulators will grow from the current main systems and buildings of one reactor, to the whole NPP extension and the surrounding territory for describing with a three-dimensional approach the interactions with the environment. The innovative concept will be turned into a first prototype of enhanced simulator, that will be demonstrated through case studies based on a real LWR under operation. The demonstration will be performed on the Pisa University hardware platform NUTEMA (that is basically a knowledge management and simulation system, see Fig. 1) and with support from the HPC (High Performance Computing) CRESCO system at ENEA. One specific relevant motivation for the development of enhanced simulators derives from the analysis of the Post Fukushima Stress Tests and the linked ENSREG recommendations focusing on a deeper consideration of extreme natural events in the design and safety verification of NPPs. ENES will therefore include new models to be integrated in the engineering simulation environment, as solutions to problems so far dealt with in simple ways, if not neglected, especially those related to extreme weather events (e.g. hurricanes, tornadoes, floodings).

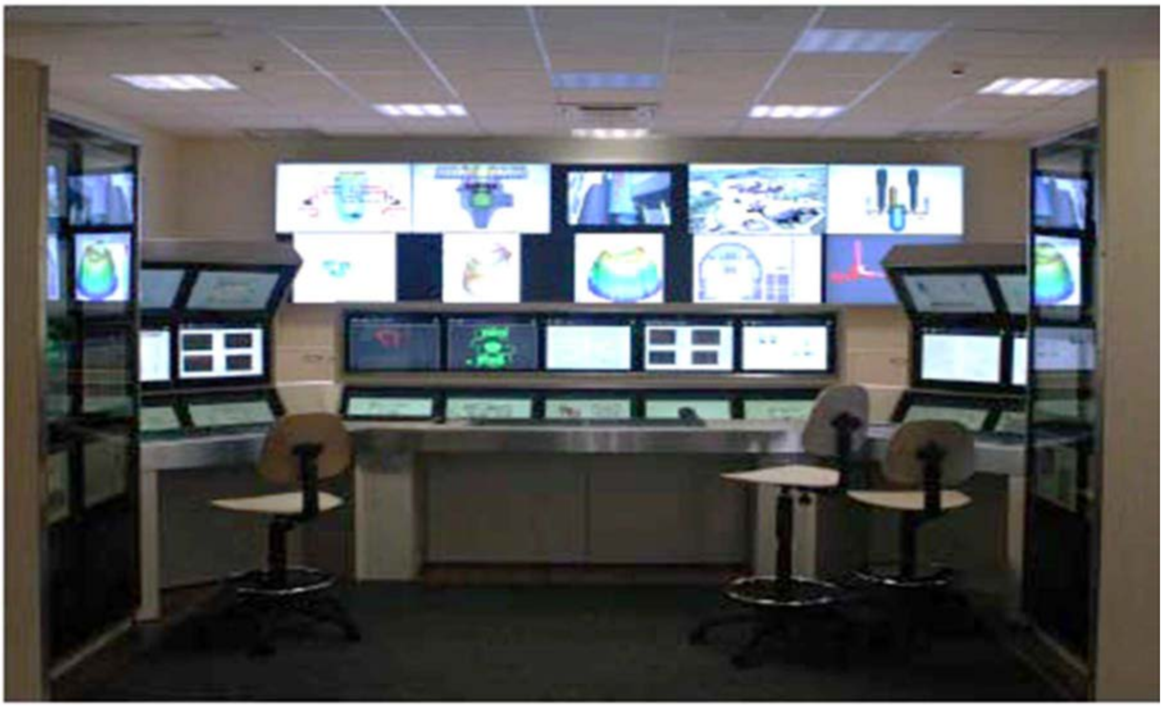

FIG. 1. NUTEMA simulation and knowledge management platform.

Finally, increasing computation speed beyond real time will also allow using ENES-based future simulators as a Decision Support Tool during plant life, when operators' decisions can be eased by fast and accurate analysis of alternative action consequences.

2. OVERALL CONCEPT AND METHODOLOGY

The overall objective of the ENES Project [1], [2] is to design in detail the desirable improvements to the current generation of Engineering Simulators (ES) for Gen II – Gen III NPP. The project will identify the R&D needs within the NUGENIA Roadmap that can be at least partially satisfied through a new generation of Engineering Simulators. Topics dealing with design and plant upgrading, probabilistic and deterministic safety assessment, analysis of accidents originated by single or multiple extreme external events are among those to be benefited by future simulators. The analysis of the expected user needs will go in parallel with the identification and analysis of the ongoing developments for the currently used software models in existing simulators and of the 'innovative categories' of models that can be introduced in the short term (e.g. those concerning Extreme Natural Events). The result of this extensive comparison between needs and possible solutions coming through simulators innovations will be the detailed definition of an Enhanced Simulation Architecture (ESA) that will include both the macroscopic aspects (e.g. the agreement of the 'simulation areas' to be covered in the new tools, or what will be the extension of scope of the new tools) and more 'microscopic' detailed aspects such as the definition of the best alternative software models to be used, the specific metadata to be exchanged among these software tools, the relationships between some of these models and a Geographic Information System (GIS) or with a High Performance Computing (HPC) platform.

The definition of the Enhanced Methodology will allow the selection of a variety of alternative software tools available worldwide for covering the extended functionality of the new simulators. ENES project doesn't aim at defining a 'unique' very specific architecture that can be implemented only with a given set of software models. And it would be unrealistic to try to materialise the future ESA by means of 'whatever' tools will be made available in the market for covering a certain 'area of simulation'. So, the keyword in the identification of 'alternative tools' compatible with the ESA will be 'balance', by avoiding both an excessive architecture rigidness and a not manageable hyper-adaptability to any type of software models dealing with the identified simulation areas. The below reported preliminary scheme of a possible software architecture (Fig. 2) reflects the key idea of identifying systematically the 'classic' and 'innovative' simulation areas, and consider the best fitted software models in each area for defining ‘alternative solutions’ for implementing the same flexible innovative simulation architecture for future enhanced ES. The scheme highlights the identification of key simulation areas in core modelling (CORE-SIM), Primary and secondary cooling (RCS and SEC), auxiliary and emergency systems (AUX, EMER), containment and severe accidents (CONT, SA), External Events and NPP Impacts (EXTEVE and IMP), electric systems (ELSYS).

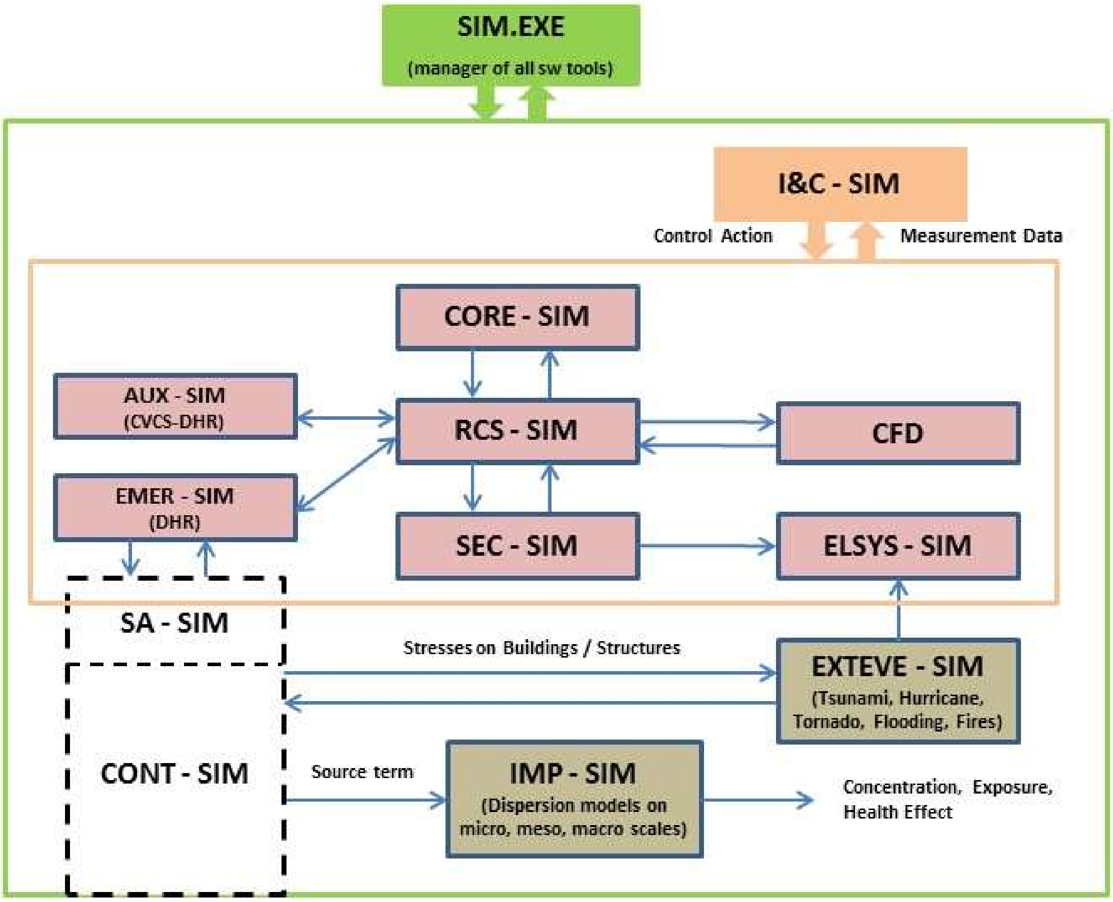

FIG. 2. First trial architecture for enhanced engineering simulators.

3. EXPECTED IMPACTS

ENES will produce various kinds of impacts concerning different categories of end-users of project outcomes. Here the differentiated impacts are shortly described as a function of the identified stakeholders [2].

3.1. Impacts on Research Organisations

ENES consortium Research Organisations are active in the simulation sector and in particular in the realisation of advanced simulators in the energy sector and the nuclear field in particular. The technical-scientific achievements planned in ENES will grow significantly the expertise and the competitiveness of these partners in the context of the R&D sector, thus easing the creation of new funding opportunities both at national and international level.

3.2. Impacts on utilities operating NPPs

The availability in the next years of an enhanced generation of engineering simulators will allow NPP utilities to better satisfy the requirements for the new NPPs (EUR requirements or more specific utility requirements). Good examples can be a better design of safety enhancement systems and an assessment of the potential effects of extreme natural events on the integrity of key plant systems and components.

Improved simulators will support the optimisation of non-electric applications of NPP (e.g. for integrated production of electricity, feeding of district heating systems or provision of steam to nearby industries).

Enhanced engineering simulators will improve the training capabilities of all kind of utilities engineers (designers, operators …), will increase the opportunity of using these simulators as a Decision Support Tool during plant life (for analysing the effects of alternative actions in delicate or crucial situations), and will represent a strong basis for developing or improving full scale Training Simulators. Therefore, ENES will enhance innovation capacity within utilities, and will strengthen their competitiveness.

3.3. Impacts on simulator vendors

ENES Euratom project proposal involves two simulator vendors of international value. ENES results will give to these two commercial companies building and selling engineering simulators since many years a set of innovative solutions for developing enhanced simulators for their future customers. Notwithstanding their evident competence and competitiveness, ENES most relevant results will enhance their innovation capacity and help them to create new market opportunities thus triggering the growth of these companies. Moreover, the strong dissemination and exploitation effort planned in ENES will benefit in these terms also the other actors in the international community of simulators vendors.

3.4. Impacts on Training Capacity

The outcomes of ENES will affect the training capacities of organizations beyond the utilities, such as Technical Safety Support Organizations (TSO) or National Safety Authorities (NSA). This additional impact meets the following expectation from the Amended Nuclear Safety Directive: “In order to ensure that the proper skills are acquired and that adequate levels of competence are achieved and maintained, all parties should ensure that all staff having responsibilities relating to the nuclear safety of nuclear installations and to on-site emergency preparedness and response arrangements, undergo a continuous learning process. That can be achieved through the establishment of training programs and training plans, procedures for periodic review and updating of the training programs as well as appropriate budgetary provisions for training”.

3.5. Impacts on nuclear industry

ENES project sees industrial organisations involved both in the consortium and in the ‘advisors and users board’. Such entities will gain from the ENES results the possibility to have in their portfolio of Design and Safety Assessment tools a variety of solutions for coping with various specific problems and the possibility to make use of integrated enhanced engineering simulators for better design, improved safety assessment, use of future simulators as DST in current and near future NPPs. These effects will enhance their innovation capacity and will strengthen their competitiveness and facilitate the growth of such companies.

3.6. Impacts on involved SMEs

ENES project involves also a few SMEs either as partners or as planned subcontractors. These companies operate in specific technical sectors that are of high strategic value for ENES (e.g. nuclear engineering, software development, EC projects management, impacts of energy systems). Participation in ENES will increase innovation capacity and will create market opportunities and chances for companies growth.

3.7. Impacts on the society as a whole

Better simulators for designing and assessing at best the new generations of NPP will contribute to the ‘sustainable development’ of nuclear energy, thus contributing to the limitation of emissions of greenhouse gases, but also protecting air quality from polluting and health-damaging emissions from fossil burning plants. By supporting the initiatives and guidelines on the safety of NPP, the project will also give its contribution to the limitation of the probability and severity of possible future NPP accidents, also considering the risks connected with the more frequent and heavier extreme weather events that could be in future the ‘initiating events’ of new severe nuclear accidents: a new generation of reactors designed with a new generation of simulators, in which the consideration of extreme natural events will be the rule and not the exception, will represent an additional protection against this sometimes underestimated but growing risk for NPPs.

The architecture of the ENES Euratom proposal is shown in Fig. 3.

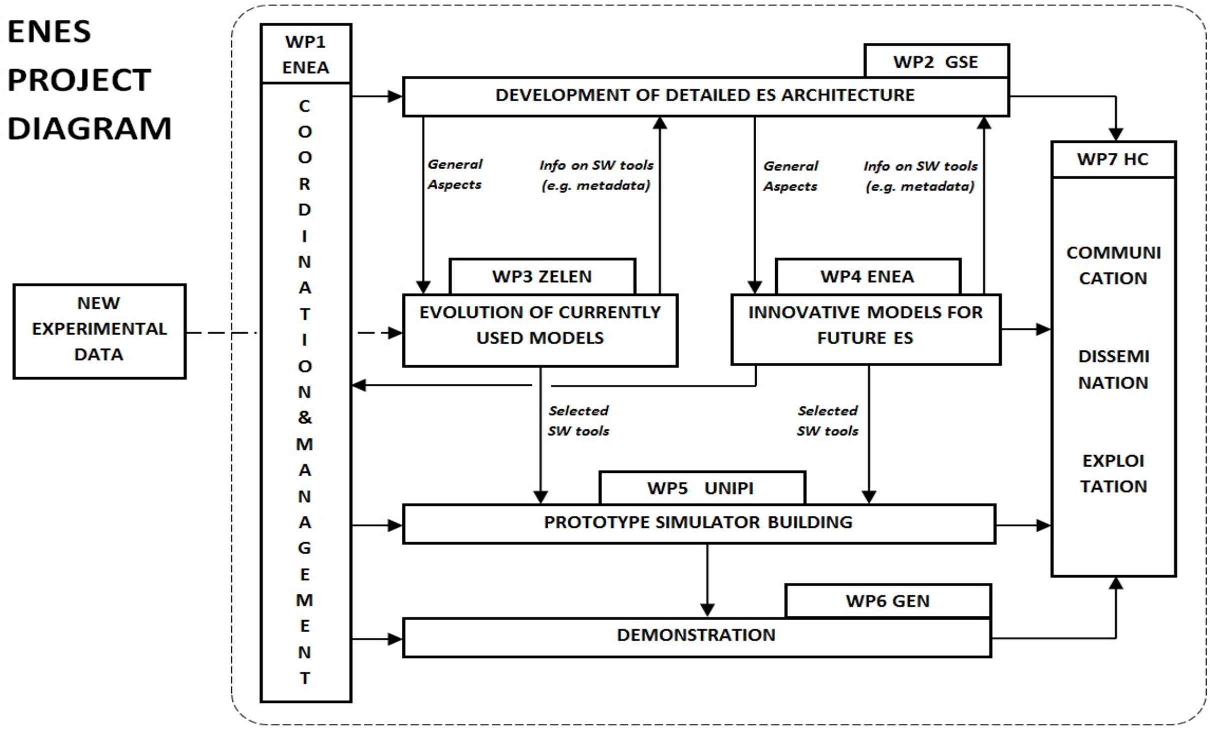

FIG. 3. Organisation of the ENES project work packages.

ACKNOWLEDGEMENTS

Acknowledgements are here made to the organisations having cooperated within the development of the ENES Project idea under NUGENIA association, and-or the preparation of the first ENES proposal under the Euratom call 2016-17 (as either partners or members of the advisory board), namely Tecnatom (E), Gen-Energija and Zel-En (Si), AREVA and Heich Consult (D), Ansaldo Nucleare and NINE srl (Italy), GSE Systems (S), UJV (Cz), L3 (Can). The authors hope these organisations will continue supporting the development of the ENES Project.

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THE NUCLEAR SIMULATION CHAIN OF GRS

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Abstract

In order to simulate all relevant phenomena within a nuclear power plant during normal operation, incidents, accidents and severe accidents, GRS uses different largely self-developed and validated methods and computer codes. These codes are forming the so called nuclear simulation chain covering phenomena of neutron kinetics, thermal hydraulics within the cooling circuit and containment as well as structural mechanics. By developing most of the codes by its own, GRS gets an enhanced understanding of the related physical phenomena and remains independent of other (commercial) organizations. In parallel, GRS also uses third party codes and develops interfaces to its own codes. Due to decreased funding, GRS cooperates also with other organizations when developing its codes. In order to stay and effect on the current state of science and technology, GRS takes part continuously in national and international projects, especially in experimental programs. The codes can be licenced by interested organizations in order to increase the worldwide nuclear safety standards. This paper shows first an overview about the whole simulation chain, while secondly the motivation of the further development of the nuclear simulation for advanced light water reactors (ALWR) is described.

1. INTRODUCTION

In Germany, the Atomic Energy Act (AtG) [1] is the legal basis for the construction, operation, modifications as well as decommissioning and dismantling of nuclear power plants (NPP) and other nuclear facilities. The required safety is regulated by laws, regulations, technical rules and policies. These safety requirements are adapted to the latest developments and insights; an important measure for this is the current state of science and technology [2]. In the nuclear licensing and supervisory procedures, the compliance with these requirements is assessed, confirmed and monitored during the entire life time. Safety assessments of NPP and other nuclear facilities are extremely complex. Technical, as well as organizational, issues have to be considered. The technical complexity comes up because the plants are very different themselves, built at sites with different conditions, and are equipped with various types of reactors and components.

The independent and nonprofit Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH is the main Technical Support Organisation (TSO) in nuclear safety for the German federal government and active in all of the above-mentioned activities. Additionally, GRS is a major research organization. Over 60 technical experts are developing and validating amongst others a nuclear simulation chain, which allows the simulation and assessment of all relevant phenomena for the analysis of operational states, incidents, accidents and severe accidents in NPP and in other nuclear facilities. The scientific basis for the code development and validation activities is built by new insights on physical phenomena, reliable plant and experimental data, as well as information gained from operational occurrences or accidents. Because GRS operates no test rigs, monitoring and evaluation of the results of national and international reactor safety research network projects, and especially the participation in experimental programs, are an essential part of the work. Through its national and international research and expert activities, GRS is able to consider, in this context, the current state of science and technology.

2. NUCLEAR SIMULATION CHAIN

Today, a comprehensive, historically grown scientific code system is available at GRS. In general, GRS develops, as far as possible, its own codes, because this approach leads to an improved understanding of the relevant physical phenomena. This approach allows GRS to be independent of the interests of commercial software developers and therefore to improve selected codes to respond faster and more flexible to current events.

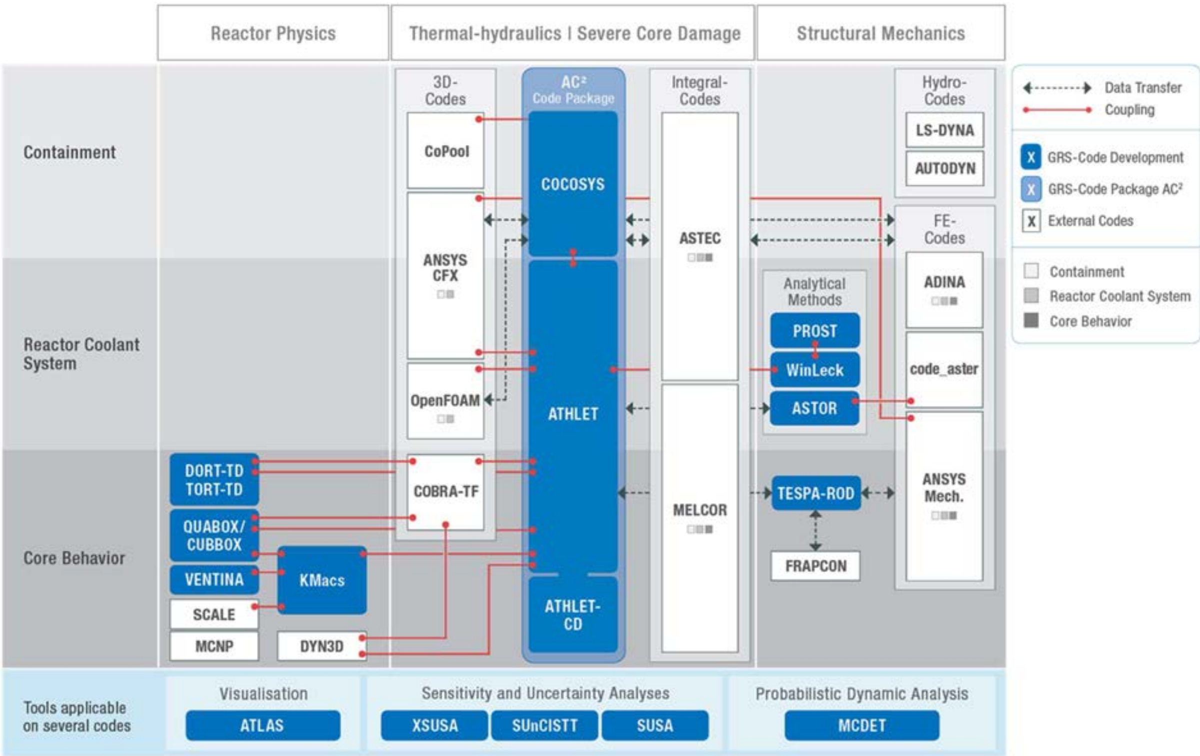

FIG. 1. Nuclear Simulation Chain of GRS [3].

The structure of this nuclear simulation chain is depicted in Fig. 1 [3]. It consists of GRS' own developments (deep blue boxes) and third party codes (white boxes). Many codes can be coupled simply for data transfer (indicated by dotted lines in Fig. 1) or in a more complex way through interfaces (indicated by red lines in Fig. 1). The latter option requires the development of appropriate interfaces. The advantages of coupling will be discussed more detailed later in this section.

The codes are assigned to main research fields: reactor physics, thermal-hydraulics/severe core damage and structural mechanics (columns in Fig. 1). The systems/components: reactor core, reactor coolant system (RCS), containment which can be simulated with the codes are arranged in rows and correspond to the respective fundamental safety functions control of reactivity, core cooling and enclosure of radioactivity. In addition, there is a fourth row, which contains other codes (e.g. for visualization, sensitivity, uncertainty and probabilistic dynamic analysis). The purposes of the main codes developed and/or used by GRS and sorted by research fields are explained in the following:

- reactor physics

— DORT/TORT: solution of time-dependent neutron transport equations for 2D/3D transients;
— QUABOX/CUBBOX: 3-D neutron kinetics core model;
— KENOREST: prediction of the characteristics of irradiated light water reactor fuels;
— VENTINA: calculation of bun-up nuclide inventories;
— KMacs: modular adaptable core simulator;

- thermal-hydraulics/severe core damage

— AC^2: code system consisting of the codes ATHLET, ATHLET-CD and COCOSYS;
— ATHLET: lumped parameter code for analysis of leaks and transients in the RCS;

— ATHLET-CD: extension of ATHLET for severe accident analyses in the RCS including core meltdown and fission product release;
— COCOSYS: detailed LP code for analysis of conditions within the containment and buildings of NPP in case of accidents and severe accidents;
  - structural mechanics
— PROST: probabilistic analysis of structural reliability of piping/vessels;
— WinLeck: analysis of leak areas and discharge flow rates based on geometry, material, medium;
— ASTOR: simplified procedures for the integrity assessment of reactor pressure vessels (RPVs) and piping loaded under internal pressure and high temperatures concerning failure times;
— TESPA-ROD: strain and burst behavior of a fuel rod under of a LOCA conditions;
  - others
— ATLAS: analysis simulator for pre-processing, visualization of results and interactive control of the simulation of several computer codes;
— SUnCISTT: determination of sensitivities/uncertainties in criticality, inventory, source term tool
— SUSA: uncertainty/sensitivity analyses;
— XSUSA: nuclear cross section uncertainty/sensitivity analysis;
— MCDET: Monte Carlo event tree for probabilistic assessment of consequences of (severe) accident scenarios.

The advantage of the nuclear simulation chain is that selected modules and/or codes can be coupled if necessary. Below considerations for couplings of codes within a research field or research field overarching are discussed. For example, in Fig. 1 it can be seen that three different methodologies exist for the calculation of the thermal-hydraulics in the RCS. These are the system code approach (AC^2), the CFD code approach (ANSYS/CFX or OpenFOAM) or the coupled system and CFD code approach (AC^2 and ANSYS/CFX or AC^2 and OpenFOAM). The selection of the appropriate approach and code for the respective issue is based on the necessary spatial resolution. With the latter mentioned options, it is possible to determine the overall RCS behavior by the LP code while relevant parts being of special interest can be calculated in a detailed 3D solution. In general, the same considerations can be applied also for the containment. In this research field, work is ongoing to couple COCOSYS with a 3D water pool model CoPool [4].

The starting point of the research field overarching consideration is the simulation of a design basis accident (DBA) with the ATHLET code (now integrated in AC^2). Further codes can be coupled to consider detailed phenomena/processes such as QUABOX/QUBOX for the detailed simulation of 3D neutron kinetic in the core or CFD codes like ANSYS/CFX or OpenFOAM for the simulation of 2D/3D velocity, temperature and concentration fields in the RCS or the containment. If the DBA is progressing into a beyond design basis accident (BDA) or a severe accident, in which for example core melt and melt relocation occurs, the user can pass over to the ATHLET-CD (also included in AC^2). ATHLET-CD can be operated in coupled mode again with COCOSYS (also included in AC^2) for whole plant analysis including fission product behavior and release into the environment. In the near future, it is also intended to couple AC^2 with ASTOR to consider structure mechanics aspects of RCS behavior under steady state and dynamic loads or fluid/structure interactions [5].

All applied codes are systematically validated, e.g. the thermal-hydraulic codes based on a well-balanced set of integral and separate effect tests e.g. derived from CSNI code validation matrices.

3. MOTIVATION FOR FURTHER DEVELOPMENT OF THE NUCLEAR SIMULATION CHAIN UNDER CONSIDERATION OF THE CURRENT GERMAN AND EUROPEAN ENERGY POLICY

After the Fukushima nuclear disaster the German federal government decided to terminate the use of nuclear energy latest in 2022 (13th amendment of the atomic energy law in March 2011) [6]. The government decided to shut down permanently 8 NPPs (GKN-1, KKB, KKK, KKP-1, KKU, KWB-A, KWB-B, KKI-1). Nine NPPs have been continuing operation, but the time points when the operation permission of the respective NPP expires has been fixed (2015: KKG, 2017: KRB-B, 2019: KKP-2, 2021: KBR, KRB-C and KWG, 2022: GKN-2, KKE, KKI-2). Under these special political conditions it is important, to illustrate our numerous national and international partners, how the further development of the nuclear simulation chain of GRS continues.

The further development of the GRS nuclear simulation chain takes into account two (a national and an international) aspects. The national aspect is based on the common understanding of all German stakeholders, that in the remaining operating time of the German NPP, the high safety standard shall be maintained and further improved. During the structured phasing-out safety must be guaranteed in line with the latest developments in science and technology. This requires besides extensive training at universities and colleges, for the next generation of scientists and personal working in utilities, approval authorities and TSO also the further development and validation of the nuclear simulation tools.

In Europe, national government policies differ on the further use of nuclear energy for electricity generation. Many countries (e.g. Russia, UK, Finland, Hungary, Poland) are planning to build new NPPs or at least maintain and/or extend their operating time. Currently 27% of all electricity consumed in the European Union (EU) is generated by NPPs. All scenarios of the Energy roadmap 2050 of the European Commission (EC) include the reliance on nuclear power. This implies both an increasing role for long-term operation and the construction and grid connection of new builds [7]. The projection in the latest Nuclear Illustrative Programme (PINC) forecasts a stable nuclear capacity in Europe between 95 and 105 GWe from 2030 onwards. At this time, roughly 80-90% of the installed capacity would be new builds [8], mainly Advanced Light Water Reactors (ALWR) considering important lessons learnt from the Fukushima event (such as the control of long term station blackouts) and the EU Stress Tests. For the reception of legitimate nuclear safety and/or security interests German authorities require in this context own and independent expertise for the safety assessments of NPPs and other nuclear facilities in our neighbourhood on an international level of science and technology.

Reactor technology has been developed and improved for almost five decades. The ALWR designs are now ready to solve the future EU energy supply shortfall problem. They incorporate passive safety features (PSFs) which do not require any active controls or operational intervention to manage accidents. The PSFs work according to basic laws of physics such as gravity and natural convection and are automatically initiated. By combining these PSFs with proven active safety features (ASFs), the ALWR can be considered to be amongst the safest equipment ever made [9]. In Gen III+ reactors, DBA can be solely controlled by PSF.

Current nuclear rules and regulation as well as evidence tools were largely developed for Gen II NPPs relying on an ASF. Although they are generic and reactor-design independent, it must be ensured whether they can treat all aspects of PSFs reasonably. This includes inter alia the mutual influence between ASFs and PSFs, of the PSFs with each other, of different trains of PSFs with each other and the application postulates (e.g. failure of trains of systems and the single failure concept). The widely different definitions for PSF currently used, cause difficulties in applying nuclear rules and regulations. The most common definitions originate e.g. from the IAEA, EPRI and the German Safety Requirements for Nuclear Power Plants. They differ e.g. concerning the degree of passivity, the supply of auxiliary energy to initiate the system start. Even today, numerous still open questions addressed in [10], have to be answered prior to the implementation of PSFs:

- Which initial and boundary conditions do PSFs require for their operation? Is it ensured that these conditions will be present in case of challenge?
- How to test PSFs? How to evalute its reliability, if the PSF cannot be inspected or periodically tested after installation? Are damages leading to failure of a PSF detected reliably? How to evaluate ageing materials? How do PSFs behave under deviating conditions?
- Which redundancies do PSFs require? Must a single error be postulated? How to assess the elimination of human actions (human errors are excluded, but PSFs cannot be used for AM).

Current discussions on these topics, are taking place at the International Project on Innovative Nuclear Reactors and Fuel Cycles of the IAEA, the Working Group on the Regulation of New Reactors of the Western Nuclear Regulators Association (WENRA), the Committee on Nuclear Regulatory Activities (CNRA) of the Organisation for Economic Cooperation and Development / Nuclear Energy Agency (OECD / NEA), the Working Group Safety Fluid Systems of the European TSO Network (ETSON) and NUGENIA. The results of these discussions are continuously monitored by GRS and implemented in nuclear simulation chain. Selected examples will be presented in the following GRS contribution on the qualification of AC^2 [11].

4. CONCLUSIONS AND OUTLOOK

Despite the termination of the use of nuclear energy, the German Federal government will support the further develop and validation of the nuclear simulation chain consisting of methods and computer codes covering all relevant phenomena of (neutron kinetics, thermal hydraulics within the cooling circuit and containment as well as structural mechanics) of NPPs and other nuclear facilities also in future. This is done basically for two reasons:

first to guarantee the safe operation of the German NPPs during the structured phasing-out for the next the 5 years which require the preservation of nuclear simulation tools with the latest developments in science and technology and second to support the German authorities at the reception of legitimate national nuclear safety and/or security interests (e.g. for the life time extension of existing NPPs and new builds) in our neighbourhood.

Even under the new German energy policy framework GRS remains a stable partner for all topics of code development, validation, application, code transfer and user support and training and will continue to participate in all relevant national and international projects. The (codes of the) GRS nuclear simulation chain, are being – even beyond 2022 – further developed and validated for (all already existing as well as existing then upcoming) issues and will thereby substantially contribute to increase the worldwide nuclear safety standards.

ACKNOWLEDGEMENTS

The development of the GRS nuclear simulation chain is currently funded by the German Federal Ministry of Economic Affairs and Energy (BMWi) within several projects (essentially RS1507, RS1518, RS1520, RS1526, RS1532, RS1535, RS1540, RS 1538, RS1543, RS1546, RS1547).

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USE OF ADVANCED MODELS AND CODES

Chairperson

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NATURAL CIRCULATION TEST CAMPAIGN ON HERO-2 BAYONET TUBES TEST SECTION

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Abstract

Innovative nuclear systems, such as the SMR and Gen-IV reactors, require relatively new approaches to accomplish the function of heat removal to achieve the goal of safety and economics, both in operative and accidental conditions. The heat exchanger compactness and operability in natural circulation are of great importance for such systems. ENEA in collaboration with POLIMI and SIET is leading some studies on innovative heat exchangers within the framework of a National Research Program funded by the Italian Minister of Economic Development. The SIET laboratories in Piacenza, leader in testing and development of innovative components and systems, hosted an experimental campaign to characterize a DHR system working in natural circulation with HERO-2 test section; a bayonet tubes heat exchanger composed by two parallel tubes. The natural circulation tests conducted, even in presence of non-condensable gas, have allowed the creation of a valuable database for the characterization of the heat exchange capability in passive accidental conditions, useful for the qualification of computer codes supporting the design and safety analysis of innovative reactors. The paper presents the test campaign carried-out on HERO-2.

1. INTRODUCTION

In the frame of a National Research Program funded by the Italian Minister of Economic Development, the National Agency ENEA, in collaboration with SIET laboratories and the Polytechnic University of Milan, has carried out an experimental campaign to characterize bayonet tubes for heat exchanger applications [1] at PWR SMR conditions. Such tubes have been tested, after adaptation, in the IETI facility previously used to test and characterize steam generators with helical tubes by POLIMI [2].

The test section HERO-2 consists of two parallel bayonet tubes, each composed of an inner pipe which conveys the incoming liquid and an outer tube electrically heated. The generation of steam occurs in the interspace between the two tubes. In view of future test campaigns, the design pressure of the component is 180 bars, but the current facility is able to operate at a pressure of 70 bars and a flow rate up to 0.1 kg/s per tube. The plant is able to feed the test section with subcooled or saturated water.

In the first experimental campaign, the component HERO-2 has been tested in open circuit in order to characterize the heat exchange (with a single active tube) and to detect and quantify thermalhydraulic instabilities of the tubes under specific operating conditions (both tubes). ENEA and POLIMI conducted both the pre-test [3] and post-test analysis [4] with RELAP5 system code. The model of HERO-2 driven by boundary conditions gave quite good results even in reproducing certain instability conditions [5].

The objective of the present test campaign is the characterization of the behaviour of HERO-2 in natural circulation conditions typical of a DHR (Decay Heat Removal) system for PWR SMR. To accomplish the new task, the loop has been closed around the HERO-2 realizing the heat sink through a tube submerged in a small pool, 11 m on top of the test section. In this configuration, several steady-states have been recorded at different filling ratio and test section power: 19 single tube tests, 21 double-tube tests and 8 tests with controlled injection of non-condensable gas.

The present paper introduces the main achievements of the test campaign together with some preliminary considerations on the general behaviour of the facility.

2. FACILITY DESCRIPTION

The instrumented test section HERO-2 (Heavy liquid mEtal pRessurized water cooled tube #2) has been designed and supplied by ENEA in Brasimone, taking advantage of the present development and testing of this solution for heavy liquid metal GEN-IV applications [6].

The component is connected to the IETI facility as shown in the sketch of Fig. 1 in which are reported also some details of a bayonet tube. HERO-2 consists of a couple of parallel bayonet tubes with an overall length of about 7.3 m and external tube diameter of 1”, while the overall height of the facility is about 18 m. Each bayonet tube is made up of three concentric tubes, a slave pipe which conveys the incoming water, an inner tube to create a sealed gap filled by air in order to reduce the thermal flux to the downward water flow, and an outer tube electrically heated. The steam generation occurs in the annular space between the inner and outer tubes. At the inlet of each tube (top part), a structure is placed to housing the interchangeable orifices for water flow stabilization.

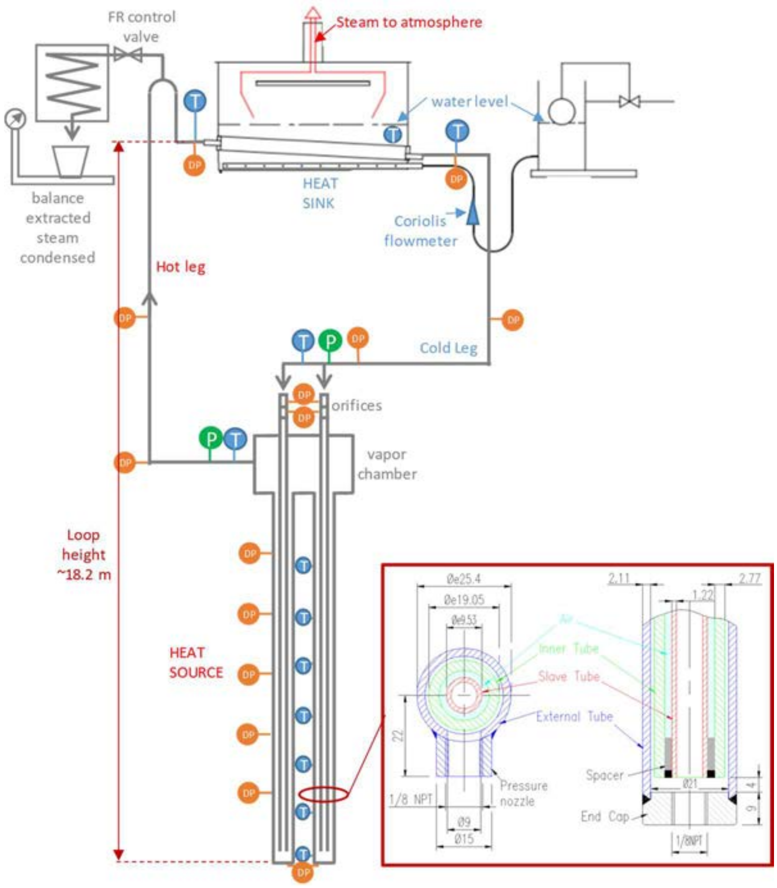

FIG. 1. Sketch of the HERO-2 facility for natural circulation studies.

The heating is realized by a total of 210 electric resistors that surround the two outer pipes for their entire length. Each electrical heater supply 240 W at 100 V, meaning that each tube is powered with up to 25.2 kW. The maximum temperature to safely operate the heaters is 350°C. In first approximation, the power generation could be considered linear even though this type of heaters leads to a certain discontinuity in the stream of power supplied, due to possible edge effect in each heater and the spaces left to give room to five pressure nozzles in each tube. The tubes, as well as the piping and the pool are thermally insulated with rock wool.

A hot leg 3/4” pipe of about 16 m connects the HERO 2 Test Section with a pre-existing condenser submerged in a small pool, which works at ambient pressure (1 x 0.4 m and height 0.6 m). The condenser tube is a near-horizontal pipe (inclined of 3°) of 2” diameter in AISI316 stainless steel. The cold leg piping of 3/4” of about 18 m closes the loop.

Since the facility works in natural circulation, no mechanical components are present. The parameters that completely define every steady-state are: the electrical power supplied through the bayonet heaters and the Filling

Ratio (FR), the latter is calculated as $(M_{max} - M_{extracted})/M_{max}$, where M_{max} is the maximum water mass at cold conditions (~19.5 kg) in the loop and $M_{extracted}$ is the water mass subtracted before the tests to reach the desired FR.

The scheme of Fig. 1 shows also the system to maintain the water pool level during the operations, where the refilling water mass flowrate is measured with a Coriolis flowmeter, and the system to reach the required FR, where the vapor extracted from the loop is condensed and weighed.

Although the facility is well instrumented in terms of temperature (T), pressure (P) and differential pressure (DP), there is no direct measurement of mass flowrates in the loop, which can be derived from the pressure drop through the orifices once the flow factor Kv is known. The characterization made with the experimental data of the previous test campaign gives an average value of the Kv=0.122. The mass flowrate is:

$$\dot{m} = \dot{m}_{tube1} + \dot{m}_{tube2} = K_v \rho \left[\sqrt{\frac{\Delta p_{DP11}}{\frac{\rho}{\rho_0}}} + \sqrt{\frac{\Delta p_{DP21}}{\frac{\rho}{\rho_0}}} \right] \quad (1)$$

where ρ/ρ_0 is the relative density with water at about 20 °C.

Another derived quantity that can be evaluated is the vapor quality at the exit of the test section:

$$x = \frac{\dot{Q}_{el} - \dot{m}(h_{ls}(P_{out}) - h(T_{in}, P_{in}))}{\dot{m}(h_{vs}(P_{out}) - h_{ls}(P_{out}))} \quad (2)$$

where $\dot{Q}_{el}$ is the electrical power supplied, $\dot{m}$ total mass flowrate, $h_{ls}(P_{out})$ outlet liquid enthalpy, $h_{vs}(P_{out})$ outlet vapour enthalpy, $h(T_{in}, P_{in})$ inlet enthalpy, with 'in' and 'out' referring to the inlet and outlet measurements of HERO-2 test section.

The net power can be evaluated through the difference between the electrical power supplied $\dot{Q}_{el}$ and the estimated overall heat losses $\dot{Q}_{loss}$, in turn obtained considering the difference between the vapor enthalpy that leave the pool $h_v(T_{pool})$ and the enthalpy of the replenishing water to maintain the level in the pool h_w:

$\dot{Q}_{loss} = \dot{Q}_{el} - \dot{m}_{refill} \cdot (h_v(T_{pool}) - h_w)$.

3. HERO-2 TEST CAMPAIGN

The test campaign consists of series of steady-state at different power and FR as follows:

- 21 double-tube tests (DTs) at FR 0.69, 0.64, 0.5, 0.43, 0.32 and power ranging from 5.0 to 50.0 kW;
- 19 single-tube tests (STs) at FR 0.72, 0.65, 0.56, 0.45, 0.35 and power ranging from 5.5 to 22.5 kW;
- 8 double tube tests with mass of non-condensable (N_2) of 4 and 7 g, FR 0.50, power from 11 to 50 kW.

The main results obtained in the test campaign in the double and single tube tests are shown in the following Fig. 2 and Fig. 3 where the vapor chamber pressures and mass flowrates versus the net power and FR are respectively reported for every tested steady-state.

As expected, the facility tends to pressurize increasing the power at constant FR, and the increase in pressure becomes more evident increasing the FR due to lower amount of compressible volume.

Also, the mass flowrate increases when the FR increases, except for the higher FR where the trend stops. At constant FR, the flowrate is characterized by a local minimum value when the power increases, resulting from the balance of two conflicting phenomena. The first is the increase of the vapor quality with the power that, reducing the hot leg average density, increases the pressure losses. The second is the pressurization, which determines an increase in the hot leg density with a consequent reduction of the pressure losses.

The ST tests are characterized by lower power than the DT ones, but a relative greater hydraulic resistance due to the exclusion of a heated bayonet tube, therefore, higher saturation pressures are recorded in STs for the same total power. Moreover, while the mass flowrate is almost halved respect the DT tests, the system behavior becomes more variable.

The Fig. 4 shows the behavior of the estimated vapor quality at the exit of the test section and the evaluation of the global heat transfer coefficient of the condenser U in the double tube tests. The global heat transfer coefficient is defined through $\dot{Q}_{cond} = S\, U\, \Delta T_{ml}$, where $\dot{Q}_{cond}$ power exchanged by the condenser, S exchanging surface area and ΔT_{ml} logarithmic mean temperature difference. Increasing the FR, the slope of the exit vapour quality function of power decreases due to the higher pressurization levels, as well as the global exchange coefficient decreases due to the liquid fraction increase in the condenser.

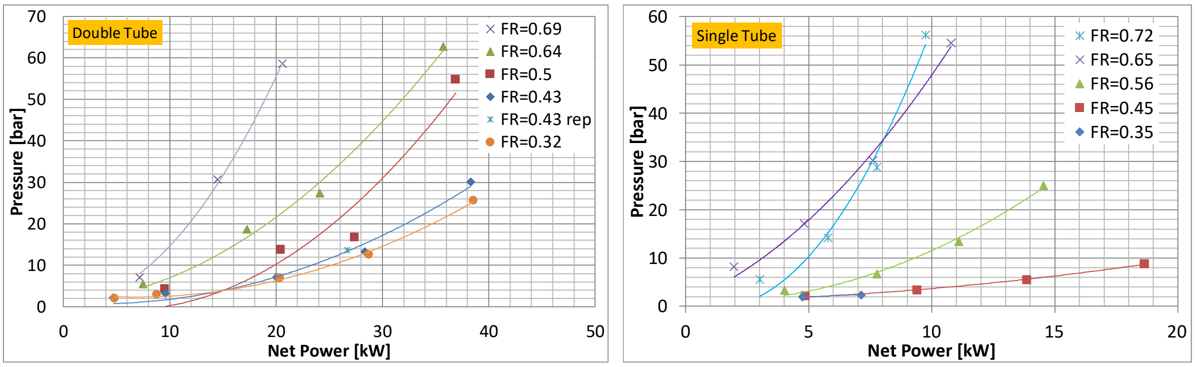

FIG. 2. Pressure vs net power and FR in single and double tube tests.

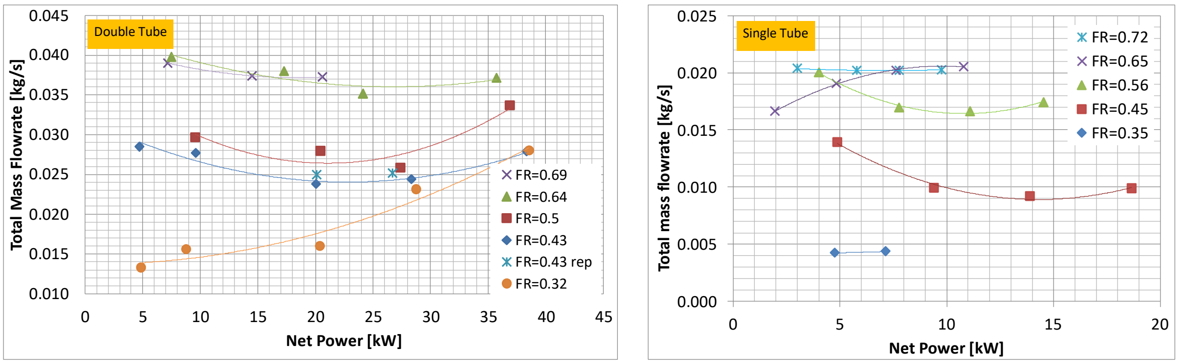

FIG. 3. Total mass flowrate vs net power and FR in single and double tube tests.

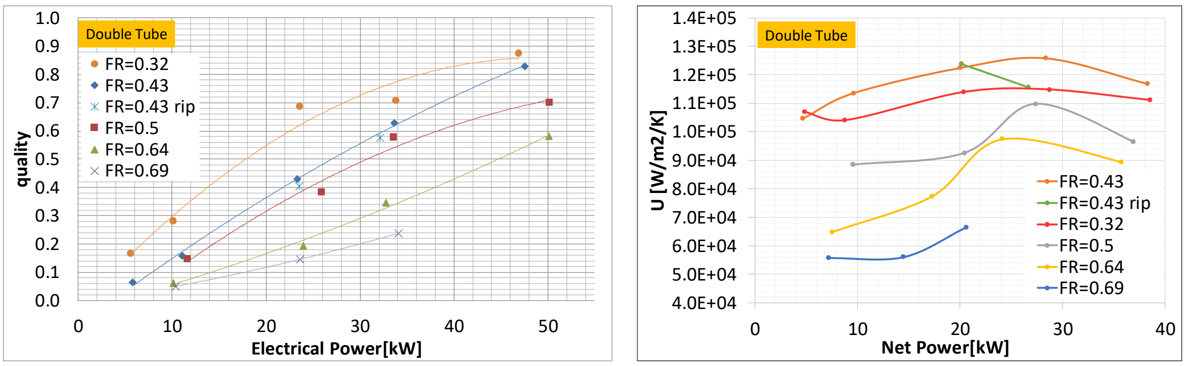

FIG. 4. Quality and global heat transfer coefficient in double tube tests.

At a given FR, the power removal from the system can be characterized by two parameters, i.e. the logarithmic mean temperature difference on the condenser and the global heat exchange coefficient. The first tends to increase with the loop pressure that sets the saturation temperature, the second may increase or decrease depending on how the two-phase thermal coefficients vary with flowrate and temperature that mainly depends on the flooding level of the condenser tube. In the latter case, the higher the saturation temperature, the higher the amount of power exchanged and in turn the vapor condensed.

Summarizing, for lower FR with higher compressible volume available, the increase in quality prevail over the pressurization up to a given power in which the circuit tends to pressurize more than the quality increases. The increase of quality means higher pressure drops, explaining the lower average flowrates, lower condenser flooding levels, higher global heat transfer coefficients and lower logarithmic mean temperature difference.

For higher FR with lower compressible volume available, the increase in pressure results in higher flowrates. The lower quality causes greater condenser flooding with reduction of the global heat exchange coefficient. Therefore, the logarithmic mean temperature difference through the pressurization drives the system behavior.

The main results of the test conducted with injection of nitrogen are shown in Fig. 4. Making the comparison against the curve FR 0.5 in Fig. 2, the higher pressurization at increased nitrogen concentration can be appreciated, meaning a deterioration of the heat removal, while the flowrate tends to increase as effect of the pressurization and the lower vapor quality that reduce the pressure losses.

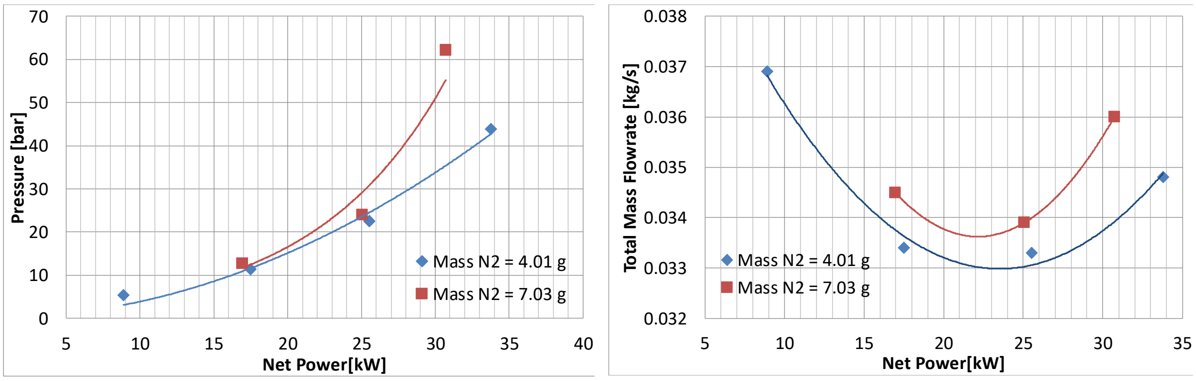

FIG. 5. Pressure and total mass flowrate vs net power and N2 mass injected (FR=0.5).

4. CONCLUSIONS

In the frame of the National Research Program funded by the Italian Minister of Economic Development, the collaboration among ENEA, POLIMI and SIET is carrying out some studies on innovative heat exchangers for SMR applications. The HERO-2 test section was previously tested to characterize the bayonet tube heat exchange capability and flow instability conditions. The present experimental campaign aim to characterize the operability of the bayonet tubes as DHR system working in natural circulation for PWR LWR conditions, thus creating a consistent database for the characterization of the heat exchange capability in passive accidental conditions, useful for the qualification of computer codes supporting the design and safety analysis of such innovative reactors.

The main achievements obtained from the series of steady-states conducted with single active tube and double-tube, with and without injection of calibrated quantities of non-condensable gas, have been briefly presented, confirming the validity of the database produced. The test campaign will be used for the assessment of RELAP5 system code.

ACKNOWLEDGEMENTS

The activity has been funded by Italian Minister of Economic Development within a National Research Program.

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PERSPECTIVES RELATED TO PROBABILISTIC SAFETY ANALYSIS

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Q. ZHANG
China

SAFETY DEMONSTRATION OF GEH ADVANCED NUCLEAR POWER PLANT DESIGNS
A summary of risk results and benefits of risk-informed regulatory initiatives

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Abstract

Probabilistic Safety Assessments (PSAs) have been performed during the design process for GE Hitachi Nuclear Energy (GEH) advanced nuclear power plant designs, including the ABWR and ESBWR advanced LWRs and the advanced sodium-cooled PRISM fast reactor. As part of the process for each plant, PSA results and risk insights have been developed and provisions were included to ensure that PSA modeling assumptions remain valid for future plants referencing the certified designs. The paper discusses the approach followed to analyze, summarize and leverage the PSA results. The PSA results encompass the entire quantified risk profile which includes internal and external severe accident progression analysis and the offsite consequence for at-power and shutdown operating conditions. Risk-informed performance-based programs, especially when applied to advanced reactors with passive design features, can help both licensees and regulators reduce regulatory burdens on safety classified systems and identify opportunities to reclassify/eliminate some safety-related Structures, Systems and Components (SSCs). These programs can relax/eliminate some technical specification requirements with potential to provide significant cost savings during reactor design, construction, and operation phases and can achieve more operational flexibility and optimize maintenance while maintaining the same safety and reliability levels. The US NRC and nuclear industry have conducted public workshops to quantify the benefits of risk-informed performance-based programs, such as the Risk-Informed Completion Time (RICT) and 10 CFR 50.69 program as applied to new reactors. GEH is developing and implementing additional risk-informed performance-based programs for advanced reactors which will make these reactors more viable options in the ever-increasing, competitive global power generation market.

1. INTRODUCTION

GE Hitachi Nuclear Energy (GEH) has developed advanced nuclear power plant designs, including the Advanced Boiling Water Reactor (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR) for the Light Water Reactor (LWR) product line and the advanced sodium-cooled Power Reactor Inherently Safe Module (PRISM) fast reactor for the non-LWR product line. For more than 60 years, GEH has advanced BWR technology that has become much simpler and safer. These designs have utilized Probabilistic Safety Assessments (PSAs) during the design process. Similarly, the PRISM fast reactor has implemented more passive features and is inherently safer, as a result. Fig. 1 shows the cut-a-way views of these advanced reactor designs.

The US Nuclear Regulatory Commission (NRC) certified the ABWR design in 1997 and the ESBWR design in 2014. The US NRC also completed the Pre-application Safety Evaluation Report (PSER) for the PRISM reactor in 1994. As part of the process for each plant, PSA results and risk insights were developed and provisions have been included to ensure that PSA modelling assumptions remain valid for future plants referencing the certified designs. The PSA results encompass the entire quantified risk profile which includes internal and external hazards, severe accident progression analysis and offsite consequence analysis for at-power and shutdown operating conditions.

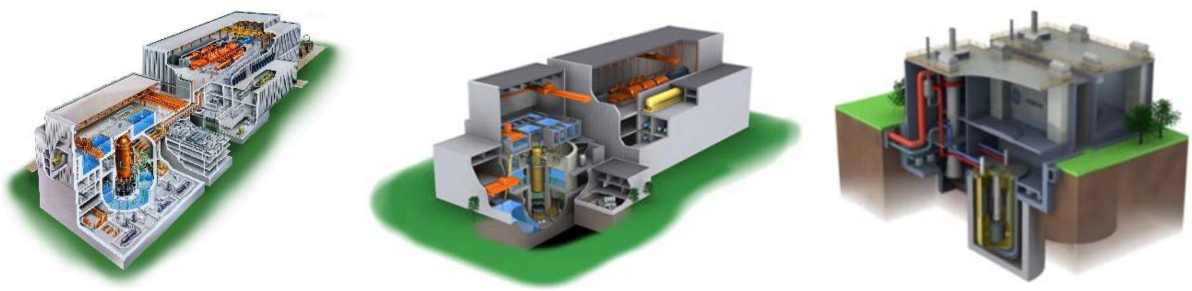

FIG. 1. GEH advanced reactor designs (left to right: ABWR, ESBWR and PRISM).

2. SUMMARY OF RISK RESULTS FOR GEH ADVANCED REACTORS

GEH ABWR technology has achieved extensive operational experience since 1996. There are multiple ABWRs in operation and new ABWR plants are being licensed in the UK. It has the lowest Core Damage Frequency (CDF) of any Gen III reactor with its simpler yet safer design with advanced technology. ESBWR is GEH Gen III+ reactor technology, which has received Design Certification (DC) from the US NRC. At the time of developing the paper, one Combined License Application (COLA) for Detroit Edison's Fermi site has been approved and the other for Dominion's North Anna 3 site is near approval. Table 1 shows the ESBWR DC PSA results. ESBWR has the lowest CDF of any Gen III+ reactor. It is a simpler design with 25% fewer pumps, valves and motors and proven passive features, compared to a typical active plant.

Table 1 includes the PSA results comparison between ABWR and ESBWR for both CDF and Large Release Frequency (LRF). Table 2 presents more detailed ESBWR PSA results.

TABLE 1. COMPARISON OF ABWR AND ESBWR PSA RESULTS (EVENTS PER PLANT OPERATING YEAR)

	ABWR [1]	ESBWR[1] [2]	IAEA Safety Goals [3]	US NRC Safety Goals [4]
Total CDF	2.37E-06	1.16E-07	1.0E-04 (1.0E-05[2])	1.0E-04
Total LRF	7.60E-07	7.95E-08	1.0E-05[3]	1.0E-05[4]

For PRISM, the risk metrics defined as CDF and LRF are not meaningful. Therefore, using the US NRC Safety Goal Policy Statement [4], the PRISM PSA defined risk in terms of three technology-neutral Quantitative Health Objectives: individual risk, societal risk, and dose risk. Because PRISM is considered inherently safe, ESBWR and PRISM risks associated with Internal Events At-Power (IEAP) PSAs are compared together in Table 3 to gain more insights, which clearly demonstrate that the ESBWR IEAP risks are very low with respect to the US NRC safety goals and can be considered comparable to the PRISM risks.

[1] For ESBWR DC PSA, a seismic margin analysis has been performed instead of a seismic PSA.

[2] Per Reference [3], application of all safety principles and the objectives to future plants could lead to the achievement of an improved goal of not more than 10^{-5} severe core damage events per plant operating year.

[3] This IAEA safety goal is set for Large Early Releases. Another objective for the future plants is the practical elimination of accident sequences that could lead to large early radioactive releases.

[4] This US NRC safety goal is also set for Large Early Releases.

TABLE 2. ESBWR PSA RESULTS (EVENTS PER PLANT OPERATING YEAR)

	Internal Events	Fire	Flood	High Winds
At-Power CDF	1.65E-08	1.25E-08	6.95E-09	8.51E-09
Shutdown CDF	1.70E-08	9.56E-09	5.21E-09	3.95E-08
At-Power LRF	1.38E-09	1.56E-09	4.09E-09	1.24E-09
Shutdown LRF	1.70E-08	9.56E-09	5.21E-09	3.95E-08

TABLE 3. ESBWR VS. PRISM INTERNAL EVENTS AT-POWER PSA RESULT COMPARISON

Risk Metrics	ESBWR [2]	PRISM [5]	US NRC Goals[5]	Meet Goals
Individual Risk (prompt deaths within 1 mile per 100,000 people per plant-yr)	<1E-04	<1E-04	<0.0391	Yes
Societal Risk (latent deaths within 10 miles per 100,000 people per plant-yr)	<1E-03	<1E-03	<0.169	Yes
Dose Exceedance Freq. (DEF, the frequency of a release with onsite whole body dose exceeding 25 rem)	<1E-08	<1E-08	<1E-06	Yes

As part of the process for each plant, PSA risk insights have been developed and provisions were included to ensure that PSA modelling assumptions remain valid for future plants referencing the certified designs. For example, ESBWR Design Control Document Tier 2 Chapter 19 [6] Table 19.2-3 includes a list of significant PSA insights and assumptions regarding how the design features affect the risk profile, and how uncertainties affect the PSA model in representing an estimate of the risks of the plant. A systematic method is used to identify PSA insights and assumptions, and to distinguish those that could have a significant effect on the PSA results if alternative assumptions were used. In order to ensure that this information is incorporated into the design process, the PSA insights and assumptions are categorized as follows:

— *Design Requirement*: an assumption that requires specific design details be preserved to maintain its validity. If a future design change affects a design requirement, the PSA model is analysed to determine the significance of the change.
— *Operational Program*: an assumption that requires specific operational programs, such as procedures or training be preserved to maintain its validity. Development of operating and maintenance procedures is the responsibility of the COL Applicant. Other operational programs that address PSA insights and assumptions are the Maintenance Rule, Technical Specifications, and development of the Site Baseline PSA model.
— *Insight*: an assumption that provides significant information about the PSA model or its results, but does not require design details or operational programs to maintain its applicability. Insights should be maintained in the Site Baseline PSA model development and should be considered when developing conclusions regarding risk-informed decisions.

[5] The US NRC goals were calculated based on the NRC Safety Goal Policy Statement [4] in Reference [2].

In order to maintain a PSA model that reasonably reflects the as-built and as-operated characteristics of the plant, controls are implemented to develop the Site Baseline PSA.

3. BENEFITS OF RISK-INFORMED REGULATORY INITIATIVES

The above risk results from GEH advanced reactors have demonstrated superior safety performance. This is an important factor for the public to accept an advanced plant in the near future. However, the design features associated with such superior safety performance bring in more diverse and redundant Structures, Systems and Components (SSCs), which not only adds cost to the capital investment of these advanced reactors, but also could potentially add more complexities to plant operations and maintenance. Therefore, it is critical to take advantage of these advanced reactors' superior safety performance and develop and apply risk-informed performance-based evaluation methodology during the design, construction and operation of new advanced reactors to identify opportunities to reduce costs while maintaining the same safety and reliability levels.

To explore potential benefits of applying risk-informed performance-based evaluation of the enhanced safety performance of advanced reactors, the US NRC decided to perform "tabletop exercises" (i.e., a simulated exercise of the process by discussion of process steps for hypothetical case studies) as a means to test the adequacy of existing regulatory guidance as applied to new reactor designs. The US NRC and nuclear industry also conducted public workshops on both Risk-Informed Completion Time (RICT) to reduce regulatory burden on safety classified systems and on the application of the 10 CFR 50.69 [7] regulation to new reactors. To support these public workshops with tabletop exercises, GEH performed RICT sensitivity studies and 10 CFR 50.69 analysis using the ESBWR DC PSA model. These exercises demonstrated that the advanced reactors, especially those with passive design features, can use risk-informed performance-based programs to achieve the benefits discussed above.

Furthermore, US NRC has taken some initiatives in developing a risk-informed regulatory structure applied to license and regulate advanced (future) reactors, regardless of their technology that could enhance the effectiveness, efficiency, and predictability (i.e., stability) of future plant licensing. NUREG-1860 [8] has been published to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future (advanced non-LWR) nuclear power plants (NPPs). A licensing process derived from a Technology-Neutral Framework (TNF) will establish a level playing field based on risk criteria and fundamental safety principles like defence-in-depth and safety margin and has acceptance criteria applicable to all reactor designs.

3.1. Results of the ESBWR RICT Studies

Tabletop exercises were performed to determine the adequacy of existing regulatory guidance on the Risk-Informed Technical Specifications (RITS) initiative 4b and Maintenance Rule 10 CFR 50.65(a)(4) as applied to new reactor designs. As part of the industry efforts, GEH performed RICT sensitivity studies using the Rev. 6 ESBWR DC PSA model.

The methods for the RITS initiative 4b, Risk-Managed Technical Specifications (RMTS), are documented in NEI 06-09 [9]. Table 4 provides the calculated RICT sensitivity study results that were developed to simulate some Limiting Condition for Operation (LCO) conditions specified in the Technical Specifications (TS) [10]. The RICT results showed that the ESBWR RICTs are generally calculated to be much, much longer than the TS Completion Time (CT) for these LCO conditions. If the RITS initiative 4b is adopted, the plant can achieve extra operation flexibility under conditions that another safety division is failed, which is typically associated with a CT of 12 hours.

This is a significant advancement in the industry's application of the methodology in large part due to the ESBWR passive safety features.

TABLE 4. ESBWR RICT CALCULATION EXAMPLES

LCO #	Affected System	Technical Specification Completion Times for LCO Conditions	Calculated RICT (Days) from Sensitivity Cases	
			Division I OOS + Channel 2 or Loop B or Division B Inoperable	Division II OOS + Channel 1 or Loop A or Division A Inoperable
3.3.1.1	RPS	12 hours for one or more functions with one required instrument channel inoperable	>1E+06	>1E+06
3.3.1.2	RPS	12 hours for one required RPS auto actuation div inoperable	>1E+06	>1E+06
3.3.1.4	NMS	12 hours for one or more functions with instrument channel inoperable in one required division	>1E+05	>1E+05
3.3.5.1	ECCS	12 hours for one or more functions with one required instrument channel inoperable	>1E+06	>1E+06
3.3.5.2	ECCS	12 hours for one or more functions with one required actuation division inoperable	>1E+06	>1E+06
3.6.1.7	PCCS	8 hours for one or more PCCS condensers inoperable	>1E+06	>1E+06
3.7.1	IC/PCCS Pools	30 days for one or both IC/PCCS expansion pools cross-connect valve DPS initiators inoperable 20 hours for one required pool level instrument channel inoperable 20 hours for one required cross-connect actuation logic division inoperable 8 hours for pool inoperable for other reasons	>1E+04	>1E+04
3.8.1	DC - Operating	72 hours for one DC source on one required DC division inoperable	>1E+05	>1E+05

DC Direct Current
DPS Diverse Protection System
ECCS Emergency Core Cooling System
IC Isolation Condenser
OOS Out Of Service
NMS Neutron Monitoring System
PCCS Passive Containment Cooling System
RPS Reactor Protection System

3.2. Results of the ESBWR 50.69 Studies

The US NRC and nuclear power industry worked for several years to develop a generic special treatment process, which was eventually codified in 2004 via 10 CFR 50.69 [7]. The NRC issued implementation guidance in 2006 through Regulatory Guide 1.201 [11], which references NEI 00-04 [12]. The roots of this risk-informed application go back to the development of the initial risk-informed Regulatory Guides that accompanied Regulatory Guide 1.174 [13] in the form of Graded Quality Assurance.

A preliminary study was conducted for ESBWR plants to implement the 10 CFR 50.69 program (now called Risk-Informed Engineering Program) to identify expected benefits (i.e., cost savings). Table 5 shows the categorization by relevance to safety of components assessed in the study. All PSA results from the modelled hazard groups and plant operation states have been used in this preliminary analysis, including the seismic margin analysis. Both the direct and indirect savings are investigated in this study.

- The total savings based on the original purchase price differences between the safety and nonsafety-related component prices are estimated to range from \$11 MM to \$20 MM.
- The implementation costs include extra efforts in PSA Update and Peer Reviews, the conduct of Integrated Decision-making Panels (IDP), and the generation and approval of License Amendment. The total estimated costs associated with the implementation of 10 CFR 50.69 are about \$3 MM to \$4 MM.
- The potential cost savings for a typical maintenance cycle for the ESBWR 60 year design life are estimated to be significant. The cost savings range from \$120 MM to \$220 MM.
- The total savings postulated for a plant's capability to avoid the use of enforcement discretion for exceeding the LCO time or potentially averting a forced outage are estimated to be about \$220 MM over a design life of 60 years.

TABLE 5. ESBWR PRELIMINARY 10 CFR 50.69 STUDY – ASSESSED COMPONENTS

Item	Number of Components
Total ESBWR PSA modeled Components	~2000
Safety-Related Components	~890
Potential LSS SR Components	522
Additional Potential LSS SR Components	105
Additional Potential LSS SR Components after changing risk significance criteria	102
Low-end percentage of potential LSS SR components	522 / 890 = 59%
High-end percentage of potential LSS SR components	(522 + 105 + 102) / 890 = 82%
Additional potential SR components not modeled in PSA	1110
Additional potential LSS SR components not modeled in PSA	1110 * ¾ = 830

SR Safety-Related LSS Low-Safety-Significant

There are two major costs associated with implementing 10 CFR 50.69. The first involves establishing a robust PSA to support categorization. The 10 CFR 50.69 process has been established so that the more complete the PSA (in terms of scope), the more likely licensees are able to categorize SSCs as low safety significance. The second cost involves implementing the modified special treatment requirements. Licensees must modify their implementation procedures and train personnel on the changes to IST, procurement, quality assurance, and other programs necessary to incorporate the results of the 10 CFR 50.69 categorisation.

However, for advanced reactors like ESBWR, these two cost factors are not significant concerns. New plants will be required to meet the PSA technical adequacy described in the PSA Standards that are endorsed by the US NRC one year prior to the initial fuel loading. Therefore, the PSA costs do not add any new burdens for any risk-informed applications. The second cost factor can be addressed by incorporating these special treatment requirements directly in the plant design to reduce plant trips and maximize the plant availability, as well as to optimize plant operation and maintenance.

3.3. Considerations of More Risk-Informed Applications and Cost Savings

As discussed above for 10 CFR 50.69, the PSA technical adequacy for advanced plants is no longer a cost issue. Therefore, it is strongly recommended to consider all risk-informed performance-based applications for the new reactors.

Also, as demonstrated in Section 2, ESBWR has an equivalent safe level to the PRISM. It may be possible to apply the Technology Neutral Framework (TNF) to an international ESBWR plant design to take full advantage of its robust safety features, which would be similar to the ongoing PRISM license basis study.

4. SUMMARY AND CONCLUSIONS

The paper summarized the PSA results for the GEH advanced nuclear power plant designs, including the ABWR and ESBWR advanced LWRs and the advanced sodium-cooled PRISM fast reactor. Considering the superior safety performance of these advanced reactor designs, it has been demonstrated that risk-informed performance-based programs (e.g., risk-informed technical specifications and 10 CFR 50.69), especially when applied to advanced reactors with passive design features, can help both licensees and regulators reduce regulatory burdens on safety classified systems, achieve significant cost savings, provide more operational flexibility and allow for optimized maintenance while maintaining the same safety and reliability levels.

GEH is developing and implementing additional risk-informed and performance-based programs for advanced reactors, which will make these reactors more viable options in the ever-increasingly competitive global power generation market.

ACKNOWLEDGEMENTS

The paper takes some PSA results from previous sensitivity studies with feedback and comments from the US NRC and nuclear industry peers during the workshops, which are greatly appreciated. Mr. Mark Colby and other GEH staff in the Chief Engineers' Office have performed thorough reviews of the draft of the paper, which are also greatly appreciated.

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USE OF RELIABILITY ASSESSMENT METHOD TO QUANTIFY PROBABILISTIC SAFETY OF REACTOR THERMAL HYDRAULIC PARAMETERS

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Abstract

Reliability analysis describes the methods to determine probability of failure, P_f, of structural components and Monte Carlo simulation has received much attention among these methods. However, due to extensive computations in determining, P_f, of structural components when P_f is particularly small, variance reduction techniques such as directional simulation approach has caught much attention. In this approach, it needs to generate number of random trials through generating random vectors in *n*-dimensional space in which the failure surface is mapped by an explicit algebraic expression. Then determining the *n*- random distances *r* from the failure surface to the origin, P_f, needs to be calculated. Although these methods are mainly applied in the field of structural engineering, however, they become popular in different branches of engineering even in financial system. The authors already applied the method successfully in probabilistic fracture mechanics to determine P_f, of structural components, but, at present the authors intend to apply the method in reactor thermal hydraulics. Although reactor thermal hydraulic and transient analysis is a common issue needs to be performed as a part of reactor safety analysis, but most of these analyses based on deterministic concept and probabilistic safety analysis is still new or beginning to appear in reactor analysis. The authors applied the method to quantify departure from nucleate boiling ratio (DNBR), the most critical thermal hydraulic phenomenon that needs to be kept within safety margin to protect integrity of reactor fuel. TRIGA research reactor of Bangladesh has been taken as reference and COOLOD-N2 code has been used considering 50 trials. It is found that P_f to increase with the increase of fuel hot rod factor, which is expected. It can be concluded the method has worked here successfully and recommended to be used further.

1. INTRODUCTION

The importance of reliability analysis in the nuclear industry has grown from a need to quantify the risk from nuclear power plant. Probabilistic risk assessment techniques for this purpose become important where the essential input is estimation of probability of failure of the key structural components. Reliability analysis describes the methods used for determining the probability of failure of these key components. In practice, Monte Carlo simulation has received much attention rather than First Order Reliability Method (FORM) and Second Order reliability Method (SORM) because of their ability to produce fast and accurate results. Nevertheless, the method needs extensive computations in estimating failure probability P_f of structural components particularly when P_f is small. Later, numerous variance reduction techniques have been developed and among these, directional simulation is found to be relatively more efficient than the other methods. It is found that although these methods had their origin in the field of structural engineering, they now have very wide applicability in the electrical, mechanical, electronic and even financial systems. From this point of view, the authors intend to apply the method first time to quantify probabilistic safety of reactor core thermal hydraulic parameters.

TRIGA research reactor of Bangladesh has been taken as reference to conduct the present reliability analysis. There are many published papers on TRIGA reactor which focuses mainly on calculation of core thermal hydraulic parameters such as fuel clad temperature, coolant temperature, minimum DNBR (departure from nucleate boiling ratio) etc, with the aim whether these parameters remain within their safety limit [1, 2]. However, all the analyses so far have been based on deterministic concept where inclusion of uncertainty or randomness in the inputs were not been considered during modeling TRIGA core. The major interest of this work is to integrate reactor thermal hydraulic analysis with the reliability prediction and hence the aim is to develop a methodology to determine the reliability of reactor fuel under normal and accidental condition through quantifying probabilistic safety of reactor thermal hydraulic parameter.

2. BRIEF OVERVIEW OF RELIABILITY ASSESSMENT METHODS

Reliability-based methods were introduced early in the 1950's and have been in practical use since the 1980's. Hence to achieve structural safety in design, construction and subsequent operation, three levels of procedure were defined by JCSS [3] generally used and these are Level 1 which is described as "Design methods in which appropriate degrees of structural reliability are provided on a structural element basis (occasionally on a structural basis) by the use of a number of partial safety, or partial coefficients, related to pre- defined characteristic or nominal values of the major structural and loading variables."Similarly, Level 2 as "Methods involving certain approximate iterative calculation procedures to obtain an approximation to the failure probability of a structural or structural system generally requires an idealization of the failure domain and often associated with a simplified representation of the joint probability distribution of the variables." Finally, Level 3 as "Methods in which calculations are made to determine the exact probability of failure for a structure or structural component, making use of a full probabilistic description of the joint occurrence of the various quantities which affect the response of the structure and taking account of the true nature of the failure domain." Methods such as First Order Reliability Method (FORM) and Second Order reliability Method (SORM) belong to the Level 2 classification. Simulation techniques and numerical integration belong to the Level 3 classification. For more information about these methods, standard text can be referred to, for example Thoft-Christensen and Baker [4] and Melchers [5] etc.

The probability of failure of a structural component obtained from the use of a particular reliability assessment method must be verified by experience and/or comparison with the results obtained from another method. It means that a FORM result should be verified by SORM. Again, for the degree of accuracy of a FORM/SORM analysis to be confirmed a completely separate method needs and Monte Carlo simulation technique is one such approach. Later, numerous variance reduction techniques have been developed and among these, directional simulation is found to be relatively more efficient than the other methods. The directional simulation technique was originally suggested by Deak [6] for evaluating multinormal distribution functions. Ditlevsen et. al [7] have used it in conjunction with load combination problems and Ditlevsen & Bjerager [8] applied the method to the reliability analysis of the plastic structures. Later, Baker et. al [9] improved the method extensively in system reliability assessment. Following the method successfully applied in probabilistic fracture mechanics (PFM), the authors intend to use the method to derive probability of fuel failure during thermal hydraulic safety study of TRIGA core. Section 2.1 has outlined the approach briefly. Section 2.2 has introduced the Box and Muller method for random number generation which has been followed in this work and section 2.3 expresses the failure criteria followed to evaluate the failure probability of TRIGA fuel.

2.1. Directional simulation method

In this approach, random vectors have to be generated in n-dimensional standard normal space where n represents the number of random variables involved in modeling the uncertainties in the physical problem. Then in calculating the corresponding distance r from the origin to the failure surface, as in Fig.1.for a number of trials, say, q, the probability of failure is given by

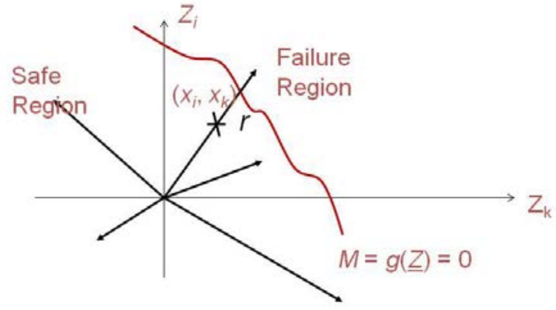

FIG. 1. Distance r calculation in the directional simulation approach.

$$P_f \approx \hat{P}_f = \frac{1}{q}\sum_{j=1}^{q}\left(1 - \chi_n^2\left(\chi r_{jf}^2\right)\right) \quad (1)$$

Where $\chi_n^2(.)$ is the chi-squared probability distribution on n degrees of freedom.

2.2. Random number generation

The essential feature of all Monte Carlo computations is that it is necessary to replace each random variable by a corresponding set of numbers having the statistical properties of that random variable. These numbers that are used are called random numbers and have to be produced by a suitable random process. The method for generating normal deviates due to Box and Muller [10] is useful because of its computational ease. It produces a pair of independent standard normal variates given by

$$\left.\begin{aligned} z_1 &= (-2lnU_1)^{\frac{1}{2}}\cos(2\pi U_2) \\ z_2 &= (-2lnU_1)^{\frac{1}{2}}\sin(2\pi U_2) \end{aligned}\right\} \tag{2}$$

Where U_1 and U_2 are independent random variables from the same rectangular distribution in the interval [0,1]. The advantage of this method is that it is accurate over the complete range and depends only on the randomness and independence of U_1 and U_2.

2.3. Definition of limit state function and fuel failure criterion in TRIGA reactor

In a reliability context, the random variables associated with each limit state need to be identified. Therefore, it is necessary to group these variables into demand and capacity variables so that a safety margin equation can be defined. In the fundamental case, the reliability of a structure is expressed in terms of two random independent variables such as a load variable (S), also known as the demand variable, and resistance effect variable (R), also known as a capacity variable. Then the system failure is given by single limit state function, say

$$g(R, S) = R - S \leq 0 \tag{3}$$

In practice, the resistance is a real performance of any physical parameter, say, temperature, pressure, flow rate which may be a function of different random variables such as geometry, loading etc and the demand is the expected value of this performance. Additionally, the demand variables themselves may depend on environment conditions and may change with time.

For safe operation of reactor, it is required to impose limits on some variables of the major process in order to protect the reactor barrier. As fuel clad temperature represents the most important barrier against uncontrolled releases of radioactivity, a safety margin against the appearance of any critical phenomena has to be fixed up. The most critical thermal hydraulic phenomenon that leads to a rapid increase in clad temperature and consequently damage in fuel elements is departure from nucleate boiling ratio DNBR of the hottest fuel rod which is defined as the ratio of the critical heat flux developed in that hot rod to the heat flux achieved in the core. To prevent the most adverse set of mechanical and coolant conditions in the core, the value of DNBR need to keep as minimum as its design value. So, the limiting criterion for fuel failure is written as

$$g(Z_{min}, Z_T) = \mathrm{DNBR}(Z_{min}) - \mathrm{DNBR}(Z_T) \leq 0 \tag{4}$$

Where, Z_T is the inlet temperature and DNBR(Z_{min}) is the minimum DNBR.

3. APPLICATION OF THE METHOID TO CALCULATE FUEL FAILURE PROBABILITY

COOLOD-N2 code [11] has been used for steady state thermal-hydraulic analysis of TRIGA reactor. Assuming one dimensional heat conduction equation together with constant heat generation in fuel meat (pellet) along the radial direction and, axial fuel rod (pin) temperature distribution has been calculated from local bulk temperatures of the coolant and axial peaking factors. Bernath correlation has been used for TRIGA type fuels for DNB heat flux calculation. Reactor was in full power operation and detailed specification could be obtained from other papers of the author. To allow uncertainties in the input quantities, two random variables have been chosen to investigate the effects on component failure probability and Table 1 can be noticed in this regard.

TABLE 1. PROBABILISTIC DESCRIPTIONS OF THE TRIGA CORE

Random Parameters	Distribution Types	Mean	CoV
Inlet temperature, Z_T	Normal	40.6	20%
DNBR, Z_{min}	Normal	2.4	20%

Generating random vectors in two dimensional space (as two random variables are considered in the present problem) following equation 2, these random vectors have to be normalized in order to get a circle of unit radius which in turn lead the random points to lie on a circle of unit radius.

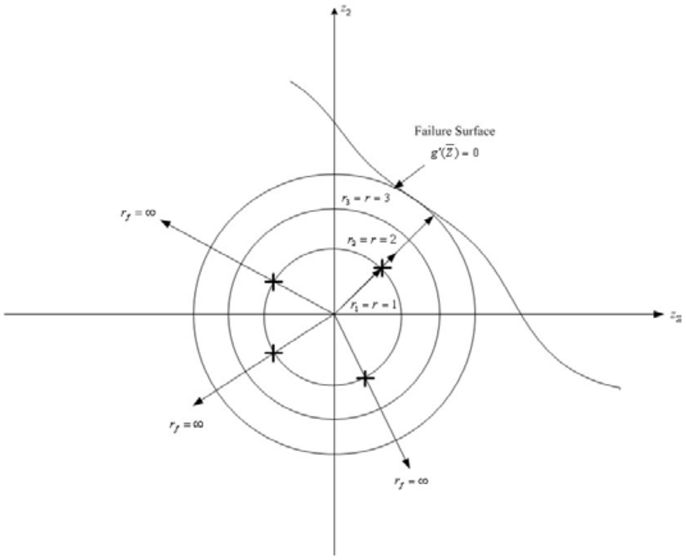

FIG. 2. Random vectors in directional simulation approach.

Transforming the random variables into physical variables, $g(Z_{min}, Z_T)$ has been calculated by subtracting DNBR(Z_T) from the corresponding DNBR(Z_{min}) as in equation (4). If $g(\bar{Z}) > 0$, the above procedure needs to be repeated by taking r = 1, 2, 3, etc until it is found that $g(\bar{Z}) < 0$ for a particular value of r, say, r_2, r_3, r_4 etc. as in Fig. 2.

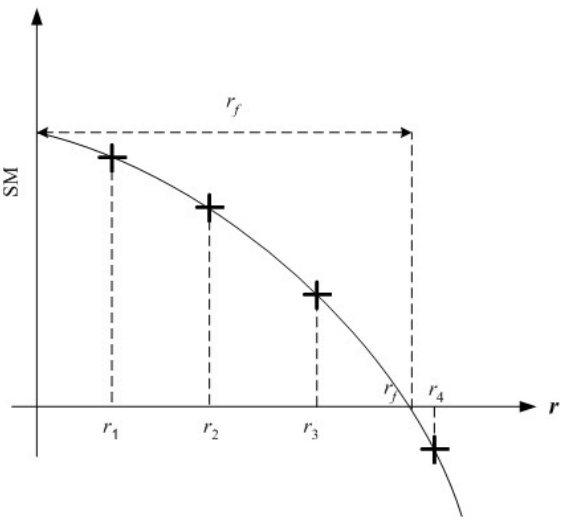

FIG. 3. Scheme for determination of r_f.

Fig. 3 shows the SM value plotted for different values of r and the value r_f corresponding to $g(\bar{Z}) = 0$ can be found by simple interpolation. The above procedure is repeated for different random vectors, as in Fig. 2, say, a total of q trials and finally, P_f has been calculated according to equation (1).

4. RESULT AND DISCUSSION

Considering the reactor power rod factor 1.5 and following the procedure discussed above, the distance r_f with for $g(\underline{X}) = g(\underline{Z}) = 0$ for 50 trials have been calculated. Hence, the r_f values obtained in a number of trials are reported and the corresponding failure probabilities have been calculated using equation (1). Using the same random number set with 50 trials, similar exercise has been repeated for rod factor 1 and rod factor 2 and finally failure probability of TRIGA fuel is plotted in Fig. 4 for three different rod factors. It is seen that the probability of failure increases with increase in rod factor values, which is expected. Hence, it seems the method works well in determining fuel failure probability. However, this is indeed a very preliminary attempt where reactor steady state has been considered. As a part of reactor safety analysis, the authors at present have given focus to calculate the fuel failure probability as a function induced reactivity considering reactivity initiated transient analysis. It can be concluded that the method opens a new area for further research in reactor thermal hydraulics.

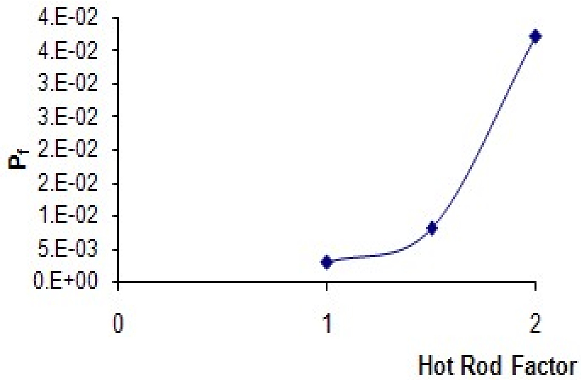

FIG. 4. Probability of fuel failure calculation using directional simulation approach.

ACKNOWLEDGEMENT

The authors gratefully acknowledge professor M.J. Baker of Safety Engineering Unit of University of Aberdeen, Aberdeen, Scotland from whom the authors have learnt the method during PhD study.

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PRACTICAL EXPERIENCE OF THE RUSSIAN VVER DESIGN ORGANIZATION IN THE USE OF PSA FOR VERIFICATION OF COMPLIANCE WITH SINGLE AND DOUBLE FAILURE CRITERIA

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Abstract

JSC "Atomenergoproekt", Moscow, is a Russian design organization, which is part of the integrated group of companies of ASE, and the main designer of AES-2006 with the VVER-1200 reactor. The AES-2006 project provides an enhanced level of safety through the introduction of innovative design solutions. JSC "Atomenergoproekt" extensively uses probabilistic safety analysis both for the selection of design solutions and for assessing compliance with the requirements of Russian and international standards, including those of the IAEA and the EU. The paper gives an overview of the results of the use of the PSA for assessing compliance of the design with the single failure and the "N+2 failure" criteria. In addition, the paper discusses the requirements for PSA that are needed to perform such analyses.

1. INTRODUCTION

JSC "Atomenergoproekt", Moscow, is a design organization that is part of the integrated group of companies of the "Atomstroyexport" (ASE). The first unit with NPP-2006M ("generation 3+" design) of JSC "Atomenergoproekt" is already operating at Novovoronezh NPP (NVNPP-2). A number of other units are at various stages of construction in Russia, Turkey and Bangladesh. The design provides an enhanced level of safety and efficiency through the use of innovative design solutions, namely, the combination of two-trains active safety systems (in some cases with additional redundancy of active elements within each train.) Figure 1 shows the diagram of the emergency core cooling system (ECCS) of NVNPP-2 as an example of a two-train system with internal redundancy.

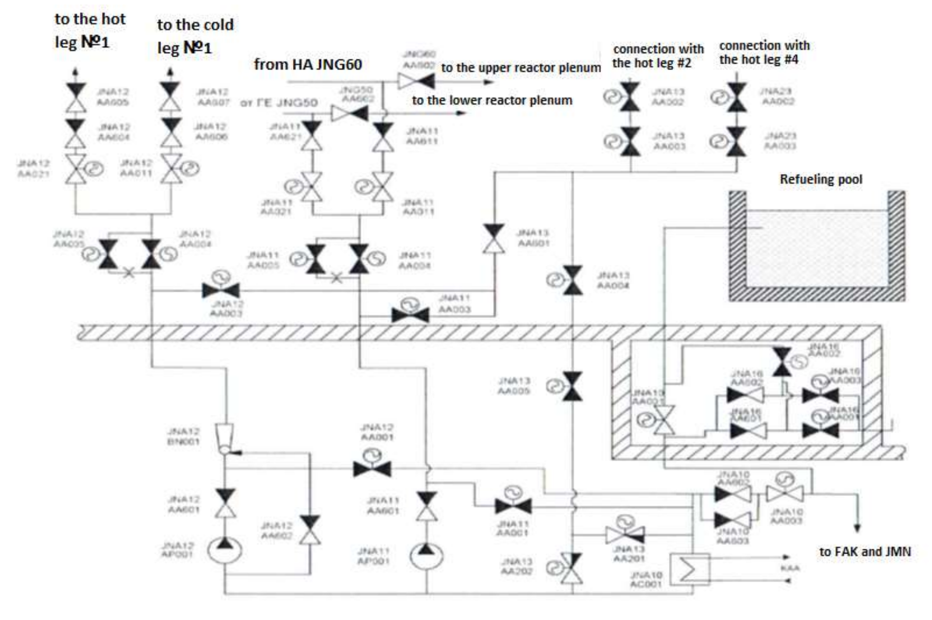

FIG. 1. System diagram of ECCS (JNA).

At the same time, the design provides a two-train structure with regard to connections of active elements with support systems (power supply, control, information representation). Passive systems, along with active safety systems, participate in performing all basic safety functions providing a time window sufficient to connect additional facilities used in the design basis accidents. Nevertheless, taking into account the two-train structure of active safety systems, including emergency power supply systems, the justification for fulfilling a number of safety criteria requires additional consideration. It is necessary to take into account not only the states with the operation of the unit at power, but also the shutdown states of the unit, in which part of the safety systems can be disabled or can be taken out of service for scheduled maintenance.

In particular, it is required to justify compliance with one of the main safety criteria - the single failure criterion (SFC), as well as the "N+2 failures" criterion, which in fact is an extension of SFC.

According to section 2.1 of [1], SFC is formulated as follows: “An Assembly of Equipment satisfies the Single Failure Criterion (SFC) if it can perform its Safety Function despite a single random failure assumed to occur in any part of the assembly during any plant design condition in which the assembly is required to operate. This includes unrevealed pre-existing failures. Consequential failures resulting from the assumed single failure shall be considered to be an integral part of the single failure”. However, according to 2.1.3.4F [1], failures of passive components may not be considered for SFC if they are designed, manufactured and assembled in accordance with the specified high quality requirements (e.g. pipelines, heat exchangers, tires, etc.).

According to paragraph 2.8.4.1.1.E [1], the criterion "N+2 failure" is formulated as follows: As Preventive Maintenance for some specific equipment performing Safety Functions is foreseen during plant Normal Operation conditions, the N+2 concept (unavailability due to maintenance plus SFC) shall be applied on a case by case basis depending on the level of function performed and on the design conditions for which the function may be required”. In this formulation N is the minimum number of redundant elements required for the function being performed. At the same time, since this criterion refers to complex sequences exceeding the SFC conditions, its fulfilment can be demonstrated on the basis of a realistic (non-conservative) approach to safety analysis. Note that for AES-2006M, N = 1 (one element is needed for performing safety function).

The assessment of the design compliance with these criteria requires an analysis of the detailed list of postulated initiating events, and, more importantly, of all dependencies between structures, systems and components. The analysis should consider a huge number of combinations of failures and repair states of equipment and should take into account all explicit and implicit dependencies potentially leading to inability to perform critical safety functions. The probabilistic safety analysis (PSA) is an effective tool that allows the most complete and justified confirmation of the fulfilment of the above criteria. The approach and results of the analysis to confirm the compliance with the SFC and the criterion of "N+2 failure" performed at JSC “Atomenergoproekt”, including the use of PSA models and results for the AES-2006M unit (exemplified by NVNPP-2) are briefly presented below.

2. OVERVIEW OF DETERMINISTIC ANALYSES OF THE DESIGN FOR JUSTIFICATION OF THE COMPLIANCE WITH SINGLE AND “N+2” FAILURE CRITERIA

In support for SFC, the NVNPP-2 design has the following features:

Emergency shutdown of the reactor is carried out by an emergency protection system, designed with appropriate backup in the initiation and implementation parts.

Residual heat removal is carried out by active safety systems (SS) consisting of two redundant independent trains, where all active components check valves inside each train are reserved (with the exception of diesel generators (DGs)).

The localization function is performed by the passive component (primary containment) and the active ventilation system of the inter-enclosure space (2 trains), failures of which cannot be induced by the design basis accidents.

A two-train sprinkler system, which operates automatically in the event of leakage under design conditions, removes heat and reduce pressure in the containment, and also helps to limit the release of radioactive products beyond the contaminant boundaries. Failures of this system cannot be induced by the design basis accidents.

In accordance with 2.1.3.4F [1] and generally accepted practice, SFC does not apply to passive components such as reactor vessel and steam generator, refuelling pool, primary and secondary containment. Also, failures of passive components (pipelines in the safety train) may not be taken into account when justifying the SFC, because they are assigned the 2-nd safety class, according to Russian regulatory requirements [2] and they are under the workload only for a short period necessary for cooling of the unit (less than 24 hours) that altogether guarantees their high reliability.

For DGs of the emergency power supply system (SAE), redundancy is not required, since, with deenergizing and failure of one DG, another DG remains in operation (there are no failures dependent on the design basic accidents).

In operating states when the reactor head is removed (before combining the volumes of the reactor shaft and the refuelling pool to carry out the refuelling), there are two trains of the emergency and planned cooling system (EPCS) are available, one of which performs the function of heat removal, and the second is in stand-by. In each train, the active parts are reserved. Scheduled maintenance (SM) of safety systems trains is not allowed in these states.

During refuelling, when the volumes of the reactor and the refuelling pool are combined and scheduled maintenance on a safety system train is allowed, operator has sufficient time to connect alternative systems to compensate water losses.

Thus, it can be concluded that the structure of the safety systems in NVNPP-2 ensures the compliance with SFC.

In terms of compliance with the "N+2 failure" criterion, the following is envisaged in the NVNPP-2 design:

When the unit is operating at power, scheduled maintenance of equipment of safety systems is not allowed. Therefore, in this state, the "N+2 failure" criterion is automatically satisfied.

For various possible configurations of safety systems in the conditions of failures of individual systems, their trains or components, the design justifies the permissible time for performing repairs at power using the criterion of the minimum frequency of severe core damage, the value of which is significantly lower than 1E-06 per year. In this case, when a single train is put into unscheduled repair of a system, the redundant components in another train (with the exception of the DGs) ensure the compliance with the "N+2 failure" criterion without the use of passive systems. The design also includes functional redundancy, which allows performing the most important safety functions by different systems.

In case of station black-out, heat removal from refuelling pool is done in a passive way, the water reserve (about 10 days) is sufficient to fulfil the autonomy criterion (6h), providing sufficient time for connecting an additional heat removal system from the refuelling pool based on an alternative independent from designed safety systems heat transfer circuit to the ultimate heat sink using mobile DG and dry cooling tower.

Connecting a mobile DG also makes it possible to use a sprinkler system to reduce the pressure in the containment, however, there is no pressure increase in station blackout conditions, since no significant evaporation of the coolant occurs. In all cases, the function of heat removal from the reactor and the refuelling pool is carried out, the limitation of radioactivity releases is ensured by the contaminant, the integrity of which is maintained.

In the state of shutdown for refuelling, planned maintenance on safety systems is allowed only when volumes of the reactor shaft and the pool are combined. In this case, failure of the second DG during loss of of-site power leads to a regime similar to that discussed above. In this case, the time reserve for fuel assemblies uncovery is at least 80 h (about 10 hours before the boiling point), an additional system of heat removal from the pool will remove heat from the total volume of the reactor-pool, preventing the boiling away of the coolant.

Thus, the "N+2 failure" criterion is satisfied for all operational states.

3. EVALUATION OF COMPLIANCE WITH SINGLE AND "N+2" FAILURE CRITERIA USING PSA

The above justification for verifying the compliance with the criteria obviously does not guarantee the completeness of taking into account all possible dependencies, in particular, there is no explicit consideration of failures caused by initiating event, as well as there is no comprehensive analysis of dependencies from the support systems (instrumentation and control, cooling water, power supply, ventilation systems) and between support that can violate the "N+2 failures" criterion.

The most effective way to verify the compliance of the design with the "N+2 failures" criterion is an analysis based on the PSA, which in this case should be developed for all operational states of the plant. It should be noted that the use of PSA in the context of verifying the "N+2 failures" criterion is fully acceptable, since complex sequences are subject to analysis.

One of the most important elements of probabilistic safety analysis is the construction of a logical model that takes into account all interrelations between systems and equipment, both intra-system and inter-system. The analysis of such a logical model using mathematical tools based on the application of Boolean logic allows identifying the minimum combinations of failures and operator errors sufficient to damage the core for the entire range of possible initiating events, including the events falling in the design extension conditions envelope. These combinations are usually called "minimal cutsets" (MCSs); they represent combinations of an initiating event, equipment failures and operator errors. In this case, if the calculation of the model is organized in such a way that there is no truncation on probability, it can be ensured that all MCSs containing at least two of the following: equipment failures and/or operator error and/or basic events representing maintenance unavailability are identified.

The analysis of the MCSs allows confirming:

— Compliance with the single failure criterion (in addition to the deterministic analysis) if it is shown that there are no MCSs containing initiating event and one equipment failure or one operator error.
— Compliance with the criterion "N + 2 failure", if it is shown that there are no MCSs containing initiating event, unavailability of the equipment due to maintenance and repair (planned or not planned) and one equipment failure or one operator error.

The developed full-scale Level 1 PSA for NVNPP-2 [3] allowed revealing all MCSs containing up to four basic events.

Note, that MCSs with four basic events have to be reviewed because for non-symmetric PSA model some MCSs may include “conditional” basic events and those MCSs should be also analysed. For symmetric model consideration of MCSs containing three basic events is sufficient. MCSs containing more than four basic events are not of interest, since for them both criteria are obviously satisfied.

Table 1 summarizes the analyses of all MCSs containing up to four basis events (including the initiating event) [3].

TABLE 1. RESULTS OF MINIMAL CUTSETS ANALYSES

Analysis of Minimal Cutsets	**Conclusion**
1. Review of MCSs containing only an IE (single basic event)	
The MCSs contain only the following initiating event caused by passive components failures directly leading to core damage: SGVR – steam generator rupture SGCR- steam generator header rupture RPRV – reactor vessel rupture These IEs are not part of the design envelope and do not require to be considered for compliance with SFC and “N+2 failure” criterion.	Compliance with the SFC and “N+2 failure” criteria is not required.
2. Review of MCS containing two basic events	
The MCSs, in addition to the IE identifier, contain basic events with “RPS” symbols. Such basic events are part of a simplified model of the reactor scram system, calculated outside the framework of the PSA model. The reactor scram system is designed in such a way that there are no single and double failures for any IE, leading to system failure.	SFC and “N+2 failure” criteria are satisfied.
The MCSs, in addition to the IE identifier, contain basic events with common cause failures that represent at least two equipment failures.	SFC and “N+2 failure” criteria are satisfied.
The MCSs, in addition to the IE identifier, contain basic events with USBT symbols. Such basic events are part of a simplified model of a control system, calculated outside the framework of the PSA model. The USBT system is designed in such a way that there are no single and double failures for any IE, leading to its failure.	SFC and “N+2 failure” criteria are satisfied.
3. Review of MCS containing three basic events	
The MCSs, in addition to the IE identifier, contain basic events with maintenance unavailability (identifier “___M”) and passive components (filters, heat exchangers, buses, electrical assemblies). Failures of passive components may not be considered when compliance with SFC and “N+2 failure” criterion is verified.	Compliance with SFC and “N+2 failure” criteria is not required.

Analysis of Minimal Cutsets	Conclusion
The MCSs include: design extension IE «Steamline rupture and leak from primary to secondary circuit", unavailability due to maintenance and single failure. Thus for these cutsets SFC is not fulfilled; however, for design extension conditions compliance with the criteria is not required.	Compliance with SFC and "N+2 failure" criteria is not required.
Other MCSs containing three basic events include: IE, unavailability due to maintenance and common cause failure.	SFC and "N+2 failure" criteria are satisfied.
4. Review of MCS containing four basic events	
Other MCSs containing four basic events include: IE, unavailability due to maintenance and two additional components failure or human errors. There are no MCSs with "conditional" basic events because the model is developed as a symmetric model.	SFC and "N+2 failure" criteria are satisfied.

Based on the results of the analysis presented in Table 1, it can be concluded that the analysis of MCSs confirms the compliance with the SFC and the "N + 2 failures" criterion in the NVNPP-2 design.

4. REQUIREMENT FOR IMPLEMENTATION OF PSA FOR RATIONALE OF CRITERIA

Summarizing the above, Level 1 PSA model for NVNPP-2 was used as a tool for the analysis of the compliance with SFC and "N+2 failure" criterion for all operating modes of the plant.

The analysis procedure included the following steps:

— Calculation of the PSA model without probability truncation limit in order to identify all MCSs containing up to four basic events (up to three basic events if the PSA model is symmetric).
— Analysis of the MCSs for the absence of cutsets violating SFC and "N+2 failure" criterion.

Obviously, the approach used to justify the implementation of SFC and "N + 2 failure" criterion for the design can only be applied if a Level 1 PSA for internal IEs including all operating states of the unit (at full power, reduced power, shutdown) is available. In addition, in the software used to build the PSA model, it should be possible to identify all the MCSs containing up to four basic events regardless their probability.

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COMPLEMENTARY UTILIZATION OF PROBABILISTIC AND DETERMINISTIC SAFETY ANALYSIS

Chairperson

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IAEA

INVOLVEMENT OF DETERMINISTIC AND PROBABILISTIC CONSIDERATIONS FOR CHOOSING ACCIDENT SCENARIOS TO BE COVERED IN NPP DESIGN (NON-SEVERE AND SEVERE CASES)

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Abstract

The paper describes the approach for selecting accident scenarios to be considered in NPP design. The approach is based on a set of deterministic and probabilistic criteria, which allow generating an accident scenarios list that would meet the requirements (1) to cover all the scenarios with significant probability and (2) to guarantee representativeness of the selected scenarios.

1. INTRODUCTION

Modern understanding of the defence-in-depth (DiD) concept (see [1], [2]) presumes the presence of a defence level related with the management of accidents out of the scope of NPP design bases. The pre-requisite for efficiency of this level of the DiD (in line with the Russian regulatory approaches of [3], this is DiD Level 4 - beyond design basis accidents management) is a plenitude of possible scenarios of Beyond Design Basis Accidents (BDBAs) covered by technical and organizational means belonged to the mentioned DiD level.

What kind of requirements exist regarding the completeness of scenarios accounting when their number is unlimited? The requirements are equivocal.

Recently enforced, IAEA standard SSR-2/1 [4] supplemented the scientific and engineering vocabulary with the term “practically eliminated” conditions which are understood as physically impossible or very low probable conditions. Standard [4] provides a requirement stating that scenarios with a large or early release of radioactive substances should be "practically eliminated", and that is the first requirement related with the completeness of BDBA accounting. That is a requirement to a plant design and in other words it means that technical and organizational measures envisaged by the design should be sufficient for compliance with the established regulatory probabilistic targeted safety goals (for instance, regarding the probability of a large emergency release).

However, according to the sixty-year experience of nuclear industry, accidents occur regardless of our perception of their probability, as appropriate, it is necessary to have accident management documentation available and to demonstrate there a management strategy for any possible emergency scenario irrespectively of its probability. This is the second requirement for accounting of BDBA scenarios (requirement of representativeness), i.e. scenarios of BDBA shall be analysed in such an amount that accident management documentation developed based on the results of such an analysis would be applicable for management of any possible accident development scenario.

Both requirements are reflected in recently approved Russian regulations [3].

2. BEYOND DESIGN BASIS ACCIDENT SCENARIO DEVELOPMENT APPROACH

The following is the approach for selecting BDBA scenarios to be considered in NPP design. The approach is based on a set of deterministic and probabilistic criteria, which allow generating an accident scenarios list that would meet the above requirements (1) to cover all the scenarios with significant probability and (2) to guarantee representativeness of the selected scenarios.

In accordance with the proposed algorithm when developing a list of BDBAs, first of all, all the NPP locations (systems) with nuclear fuel (radioactive waste) should be identified if their operational occurrence may lead to an accident. As a rule, that is a nuclear installation, spent fuel pool, nuclear fuel and radioactive waste storage facilities, transportation casks. Establishing a complete nomenclature of possible normal operation occurrences at an NPP is also important because operational conditions differ by the range of occurrences which may develop into an accident and by the condition of physical barriers and preparedness of NPP systems and staff to withstand an occurrence.

After it is defined, one can proceed with development of a BDBA list.

The nomenclature of severe accident scenarios to be listed is formed in the following six-step way.

Step 1. A list of physical barriers is developed as well as of safety functions (SFs) which impact the strategy of actions to be taken for accident management and mitigation of its consequences. A safety function here is understood as an aggregate of technical means and actions aimed at protection of physical barriers, decrease of radiation impact on personnel, public, and environment.

Step 2. A list of physical processes threatening the integrity of physical barriers under a severe accident is developed (fundamental overview of processes occurred in case of a SA is provided in [5], [6]).

Step 3. For each physical barrier and SF identified at Step 1, a gradation of conditions is developed starting from full efficiency of a physical barrier (or SF) to complete inefficiency in such a way that different states of physical barriers and SFs require different accident management strategies to be applied (which can be expressed by different technical means applied for accident management and by different actions to be taken by the staff)[1]. Development of such a scale of conditions for physical barriers and SFs is a labour intense process which requires qualified experts to be involved and phenomenology of severe accidents to be analysed, while certain cases require calculation studies. Each assembly of physical barriers is also called "the level of severity of NPP condition".

The following condition should be met in the grading of physical barrier conditions: negative effect due to materializing of a threat related to a physical process established at Step 2 should be stated in a change of NPP severity condition (i.e. as a result of occurrence of such a threat the condition of at least one physical barrier should deteriorate according to the conditions grading).

Step 4. Generalized event trees are prepared to reflect development of an accident initiating event into NPP conditions with different levels of severity depending on performance or non-performance of SFs. Generalized event trees differ from the PSA event trees by the following: (1) In the function of end states, Level 1PSA event trees consider successful and non-successful states generally without a differentiation of the latter by the severity levels, while generalized event trees include considerable amount of different end states corresponding to different levels of severity; (2) Fulfilment of a SF is given in generalized event trees without specification of systems or personnel actions involved into their implementation.

Accident sequences of generalized event trees are developed until an end state is achieved as it can be maintained for an unrestrictedly long period.

Fig.1 demonstrates an example of a generalized event tree for accidents, which were not caused by primary leakages. Severity level indices shown on Fig. 1 consists of four digits each: the first digit reflects status of primary circuit (where 0 stands for intact primary circuit, 1 stands for small and very small leakages within containment, 2 stands for medium and large leakages within containment, 3 reflects evaporation through pressurizer safety valves, 4 means primary to secondary breaks and 5 stands for primary leakages outside containment other than primary to secondary ones); the second digit reflects fuel elements status (where 1 stands for unaffected fuel and fuel damaged within DBA limits, 2 stands for severe fuel damage where core coolable geometry is preserved and 3 stands for severe fuel damage with loss of core coolable geometry); the third digit stands for RPV status (where 0 stands for intact RPV, 1 stands for high pressure RPV rupture and 2 stands for low pressure RPV rupture)4 the fourth digit reflects status of containment (where 0 stands for containment with leakage rate within design limits, 1 stands for containment with leakage rate exceeding design limits, 2 stands for penetration of containment concrete base by corium and 3 stands for containment bypass).

Step 5. The final list of BDBAs includes scenarios appropriate to each of the severity levels provided in the generalized event trees. Sever BDBA scenarios can be also given in the final list of BDBAs in a more general way, in particular, in the form of a scenario developing sequentially from one level of severity to another (with possible branching). Herewith, if a more general way of representing severe BDBAs is chosen in the final list of BDBAs, then a description of each scenario should be alongside with an indication that a strategy of BDBA management is subject to identification for each severity level of NPP condition passed through by an emergency scenario development. A tentative list of emergency scenarios for BDBAs, which are not caused by primary circuit leakages, is given in Table 1 for a VVER type reactor installation.

[1] PSA Level 2 (if available) would be useful for development of such a list.

Step 6. It is checked whether developed generalized event trees reflect NPP conditions related to severe accidents and recommended to be included into the list of BDBAs by the national regulatory documents and by widely recognized international documents. It is also checked (with the Level 2 PSA results available) that all the SA scenarios with a probability occurrence exceeding 10^{-7} 1/year (regulatory target established in [3]) are accounted in the final list of BDBAs.

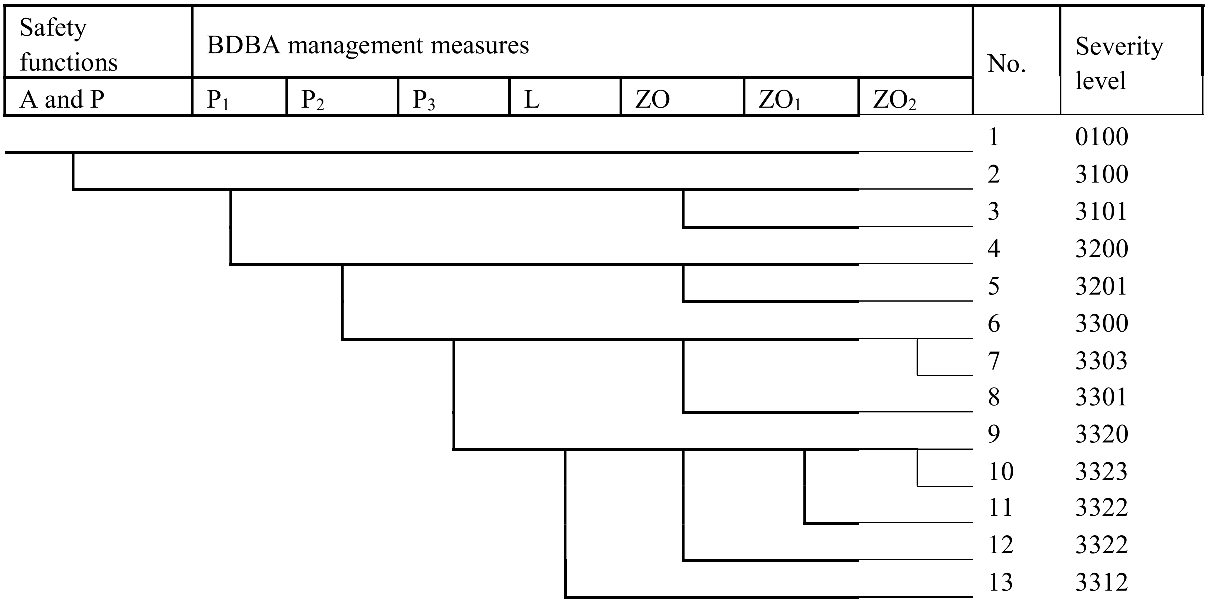

FIG. 1. Generalized event tree for BDBAs, which are not caused by primary circuit leakages (except for the scenarios with a thermal stress impact on the reactor pressure vessel (RPV) in case of secondary pipelines breaks. Station blackout is applied as an initiating event). Designation: ***A*** *stands for bringing the reactor into the subcritical state and maintenance of this condition;* ***P*** *stands for maintenance of the coolant inventory in the primary circuit and heat removal from the core;* ***ZO*** *means functioning of the containment (containment insulation, protection against overpressure, protection against a damage caused by direct heating, hydrogen explosion protection);* ***P₁*** *means limited damage of the core within the design limits;* ***P₂*** *means prevention of complete melt-through;* ***P₃*** *means prevention of RPV damage;* ***L*** *means primary circuit pressure decrease in order to prevent RPV from being damaged under a high pressure;* ***ZO₁*** *means prevention of the containment basement melt-through;* ***ZO₂*** *stands for prevention of high-temperature creep of steam generator tubes.*

For accidents, which are not severe, the above-mentioned six-step algorithm has to be modified taking into consideration the fact that NPP severity levels in non-severe case do not include states with fuel damage as well as states with RPV failures or containment basement failures. It worth to note also that according to the practical experience, the scenarios which have not developed into a severe stage may have multiple options for restoration (maintaining) of certain SF, and consequently it can be recognized as reasonable to use probabilistic criteria which provide for BDBA list consideration of the most preferred methods of SF restoration in such a way that all the ways of SF restoration which have not been considered in BDBA list would not have significant impact on the DiD (i.e. non-use of these opportunities had extremely small probability of deterioration of accident flow).

The final list of BDBAs also includes severe and non-severe scenarios received in the above-mentioned way.

TABLE 1. A SAMPLE LIST OF ACCIDENT SCENARIOS OF SEVERE ACCIDENTS AT A VVER TYPE REACTOR INSTALLATION WHICH ARE NOT CAUSED BY PRIMARY CIRCUIT LEAKAGES AND INCLUDED INTO THE BDBA LIST

NPP severity levels	Accident scenarios
3200	Long-term NPP blackout which has lead to water boil-off in the steam generator, primary coolant, to core uncovering and fuel damage, steam-zirconium reaction with hydrogen release into the environment of the containment.
3200	SG pipeline break with a failure of emergency SG insulation and long-term blackout at the NPP leading to boil-off of SG water, primary circuit coolant, core uncovering and fuel damage, steam-zirconium reaction with hydrogen release into the environment of the containment.
3201, 1201	Scenario No. 1 accompanied by non-closing of isolating devices of the containment (or any other equivalent looseness in the containment).[2]
3300, 3303	Continuation of development of Scenario No. 1 leading to fuel melt in the core (complete damage of cooled geometry of the core) and SG tubes destruction due to high-temperature creep
3301, 1301	Scenario No. 4 accompanied by non-closing of isolating devices of the containment (or any other equivalent looseness in the containment).
3320	Further development of Scenario No. 4, accompanied by measures taken to decrease primary circuit pressure which leads to RPV destruction under low pressure in the primary circuit.
3322, 1322, 2322, 4322, 5323	Further development of Scenario No. 6 leading to melt-through of the concrete basement and corium release into the NPP premises.
3312, 1312, 4312, 5313	Further development of Scenario No.4 leading to RPV destruction under high pressure, melt-through of the concrete basement and corium release into the NPP premises.

3. CONCLUSION

The paper presents the algorithm developed with both deterministic and probabilistic criteria that allows to form a list of beyond design basis accidents (severe and non-severe) subject to consideration in the NPP design, satisfying the requirements of inclusiveness (only scenarios which have been practically excluded are not considered) and representativeness (analysis of the scenarios included into the list allows to develop a management strategy for any accident faced by the NPP personnel).

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[2] It is assumed that after developing into a more severe stage, further development of Scenarios 1&2 is identical and, as appropriate, BDBA management strategies are identical starting from the accident management stage.

COMPLEMENTARITY OF DETERMINISTIC AND PROBABILISTIC APPROACHES FOR SAFETY ASSESSMENT OF FRENCH EPR DESIGN

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Abstract

Even though the safety demonstration of nuclear power plants relies essentially on deterministic bases, the probabilistic approach takes an increasing place in the safety assessment. For the safety assessment of the French EPR reactor, deterministic and probabilistic approaches have been used in a complementary way at each of the preoperational phases since the early design stage.

Despite some challenges related to the development and use of PSAs for a new built especially at the design stage, PSAs have provided useful insights to confirm the safety of the design and even in some cases to highlight the need to study further design improvements. In particular, PSAs have been beneficial to appreciate the suitability of the design regarding the risk significant scenarios, to check the sufficient independence between the defense-in-depth levels, to identify the situations to be included in the design extension category DEC-A and to contribute to the assessment of the accidental situations that should be "practically eliminated".

The paper presents lessons learned (approaches, challenges, examples of insights) from the complementary use of deterministic and probabilistic approaches for the preoperational stages of EPR in France.

1. INTRODUCTION

The first generation III reactor in France (EPR) is under the final phase of construction at Flamanville. The plant applicant (EDF) submitted to the Safety Authority on March 15th, 2015 the operating license application. The reactor is a 1600 MWe class PWR with an evolutionary design derived from mature and proven technologies of the French N4 plants and the German Konvoi plants.

In the Technical guidelines for the design and construction of the next generation of PWRs [1], the French Safety Authority (ASN) requested for the Flamanville EPR compared with the existing French plants a significant reduction of:

— The number of incidents by a better systems reliability and better consideration of the human factor;
— The global core melt frequency: "*Implementation of improvements in the "defense-in-depth" of such plant should lead to the achievement of a global core melt frequency of less than* 10^{-5} *per plant operating year, uncertainties and all types of failures and hazards being taken into account*" [1];
— The radioactive releases in accident situations (no necessity of protective measures for people living in the vicinity of the plant in case of accident without core damage, very limited protective measures (in time and in space) in case of core melt accident).

ASN also requested the "practical elimination" of situations leading to large early releases [1].

As for nuclear power plants in operation, the safety demonstration of Flamanville EPR relies essentially on deterministic bases, with probabilistic approaches used in a complementary way. ASN required *"A probabilistic safety assessment to be conducted, beginning at the design stage, including at least internal events; this probabilistic safety assessment would indicate the frequencies of core melt sequences with a view on the possible consequences of the different types of core melt situations on the containment function*." [1].

Therefore, the designer and the applicant have provided several PSAs, beginning with a preliminary PSA at the early design stage, followed by more extended and completed studies during the next stages. These PSAs have been used to assess the early design and the detailed design.

IRSN, as technical support of the Safety Authority, has been highly involved in the safety assessment process of the EPR design and, among other safety issues, in the review of the PSAs provided by EDF all over the preoperational stages.

The paper presents the main lessons learned from the use of probabilistic approaches at the preoperational stages of EPR, in terms of objectives and challenges (section 3) and in terms of benefits and safety improvements gained from the use of level 1 PSAs (section 4). Detailed PSA results are not within the scope of the paper and insights from Level 2 PSAs are not developed.

2. PSA DEVELOPPED FOR THE FRENCH EPR PLANT

During the different phases of Flamanville EPR design, several PSAs have been developed by the EPR designer (AREVA) and then by the licensee (EDF).

During the early design stage, the preliminary studies consisted in a limited scope level 1 PSA for internal events and a simplified level 1+ PSA investigating the containment and some severe accident foreseen features. Their results contributed to define early design improvements, as presented in chapter 4.

In the frame of the application for the Flamanville EPR plant construction, EDF provided:

— More complete and updated level 1 and level 1+ PSAs, covering initiators of internal origin (equipment failures and human errors) in all operational states (full power and shut down states);
— Particular studies: assessment of long duration sequences in case of LOOP (loss of offsite power) and LUHS (loss of ultimate heat sink); PSA for the spent fuel pool; studies supporting the “practical elimination” of sequences which could lead to large early releases;
— Methodological developments for level 2 PSA and for hazards PSA.

The development of these studies by EDF and the PSA review performed by IRSN in 2006 contributed to confirm the main design options of the EPR reactor but also to identify several aspects needing more investigation. In view of the application to EPR operating license, EDF provided:

— A more complete level 1 PSA for internal events (for the reactor and the spent fuel pool) that incorporated updated plant information on the plant design and the main support systems;
— Simplified probabilistic studies for some internal hazards: fire, flooding and explosion;
— For external hazards: simplified seismic margin study; probabilistic assessment for long duration LOOP and LUHS induced by extreme winds; qualitative or semi-quantitative studies for other external hazards (frazil ice, air low temperature, external flooding and clogging of the pumping station);
— A complete level 2 PSA for the internal events.

The conclusions of these different PSA studies, which have been provided by EDF in advance to the operating license application, were presented by IRSN to ASN between 2014 and 2016. It has to be noticed that, in order to gain appropriate knowledge and tools for the review of EDF studies, IRSN developed its own internal event level 1 PSA. The PSA review has confirmed the relevancy of the EPR design choices and the compliance with the probabilistic targets, and only a few additional targeted probabilistic verifications are still expected before the operating license. PSA update is nevertheless scheduled by the end of the EPR start-up phase especially for internal hazards.

3. OBJECTIVES AND CHALLENGES TO USE PSA IN THE EPR DESIGN PROCESS

3.1. Objectives for using PSAs in the EPR design process

The PSAs have been performed and used for EPR with the objectives of:

— Assessing the overall risk and identifying the main risk contributors in order to ensure a well-balanced safety concept;
— Supporting the choice of design options (systems architecture, level of redundancy and diversification, design of safety systems and their support systems);
— Adjusting the set of design extension conditions without core melt initially established on deterministic basis and checking the appropriateness of the corresponding features;

— Contributing to the verification of the “practical elimination” of some accidental situations. The Technical Guidelines specify that accident situations with core melt which would lead to large early releases have to be “practically eliminated”: if they cannot be considered as physically impossible, design provisions have to be taken to design them out.

3.2. Challenges when developing PSAs in the design process

The development of a full scope PSA for a new-built raises some challenges, especially when performed during the preoperational stages and from the very early design phase.

The major constraint came from the fact that the design information was still partial when the PSA was developed. This was mostly the case at the early design stage. However, the already defined design options, in particular concerning the global architecture and the main components of the safety systems and their support systems, were modelled in a simplified way, allowing a preliminary evaluation of the core damage frequency and the identification of the dominant sequences. As explained in section 4, this kind of preliminary study was very beneficial for Flamanville EPR to assess the global relevancy of these main design choices. Over the next preoperational phases, the PSA was progressively completed, when more precise information on the design became available. This implied an “iterative process” between the progress of the deterministic studies and the PSA development, bringing about several PSA updates and additional sensitivity studies along with the detailed design development.

Other limitations were related to:

— The detailed emergency operating procedures that were not available while developing the PSA. The human reliability analysis was thus performed based on conservative assumptions and simplifications regarding future emergency operating procedures;

— The lack of specific reliability data. However, it was possible to use first some generic data taken from existing plants operating experience (French N4 plants and German Konvoi plants) and from other available sources (international data bases). For new specific components, reliability studies were performed or expert judgment was used.

The lack of information was mainly a challenge for the internal hazards PSAs, explaining that no hazard PSA was available at the early design stage and that only simplified studies were developed in view of the operation license (e.g. all the information regarding the cables routings were not fully available for the fire PSA). Due to some simplified and conservative assumptions adopted in the internal hazards PSAs (e.g. loss of the whole safeguard building postulated in case of fire or flooding in a room), it was recognized that these studies were useful to globally confirm the design but not appropriate for all PSA uses, and that they led to less insights than the internal events PSA. It was also agreed that an update of internal hazards PSAs would be particularly necessary by the end of the EPR start-up phase to incorporate as-built plant information.

Concerning the external hazards, IRSN considered that the quantitative and qualitative studies that have been submitted by EDF up to now, with very limited scope and significant uncertainties, are not sufficiently mature at this stage to contribute to the decision process regarding the protection against external hazards and that the efforts to develop external hazards PSAs should be strengthened.

As explained above, the scope and the level of details of the PSAs developed for Flamanville EPR had to be adapted to the limitations related to the available data and design and operational information. In the decision making process, these limitations were taken into account at the successive preoperational stages. Despite these challenges, many insights came up from all these PSA developments as presented in the next section. More information can be found in [2].

4. MAIN INSIGHTS FROM PSA FOR THE FRENCH EPR PLANT

4.1. PSA insights to support the design options

PSAs performed at the successive preoperational stages have been beneficial to check at each stage that the overall risk complies with the safety probabilistic targets.

The most important insights of PSA are related to the need for more systems and components diversifications than initially defined, and most of them came up from the preliminary simplified level 1 and level

1+ PSAs (early design stage). Indeed, these studies underlined the excessively dominant contribution to the risk of some support system failures. Although these probabilistic studies were not the only basis for making decisions, they played an important role in several early improvements of the initial design of support systems, in particular:

- The implementation of two diversified diesels generators in addition to the four main diesel generators;
- The implementation of a diversified back-up cooling (air-cooling) of two low pressure injection pumps among the four pumps (all four pumps being cooled normally by the main cooling chain), in order to reduce the contribution of the sequences induced by the loss of the main cooling chain;
- The implementation of a dedicated diversified cooling chain to cool the containment heat removal system (insight from the level1+ PSA revealing a significant risk contribution of the failure of the main cooling chain supporting the safety injection system and the containment heat removal system);
- The need for a third train on the cooling system of the spent fuel pool, with diversified power supply and heat sink.

Later, as the design evolved, updated PSA results highlighted some additional needs for diversification.

Another main insight from PSA was the assessment of the sufficient level of independency between the defence-in-depth levels, to complement the deterministic assessment of systems design in particular when the independency is not fully reached. In particular, as the low pressure safety injection (safeguard function) and the residual heat removal (used in normal operation and to maintain a safe state after some accidents) are ensured by the same system at Flamanville EPR, an analysis was performed in order to check that in case of a break on the residual heat removal system, sufficient safety injection means are still available to cope with the accident. The deterministic analysis was completed by a probabilistic verification for the sequences under consideration.

The PSA presented in the application for Flamanville EPR construction and later in view of the operating license pointed out other needs of design improvements (e.g. the need for a make-up to the emergency steam generators feed water tanks in case of long-term accidents) and contributed to highlight important issues to be deeper investigated in the safety assessment (for example, regarding the mitigation of cascading failures of main diesels in case of LOOP).

The review of the PSA and in particular of the reliability assessments of systems (results and detailed modelling) was also very useful for IRSN when reviewing the detailed design of the safety systems and of their support systems, and vice versa. The team in charge at IRSN of the “deterministic” assessment of EPR systems design and the “IRSN PSA team” have been working in parallel on the review of the EPR design, with close connection allowing mutual benefits for the evaluation of the safety level of the EPR plant.

Concerning the internal hazards PSAs, less detailed insights for the EPR design came up from these PSAs. However, their results confirmed that the safety objectives can be fulfilled and that EPR design is more robust to internal hazards than the French operating GEN II reactors (four safety trains, geographically separated buildings…). The PSA review by IRSN identified some particular rooms (such as the annulus between internal and external containments) and some scenarios (e.g. risk of flooding by the containment refuelling water storage tank following a rupture of the safety injection system) for which additional justifications and assessments had to be provided by EDF in order to confirm the deterministic design. Besides, the internal explosion PSA turned out to be useful in the design process even if the study was a simplified one (assessment of the frequency of occurrence of explosive atmosphere in the rooms for which an explosion risk was identified in the deterministic studies). The PSA results globally confirmed that the internal explosion risk was acceptable for most of these rooms, but also that some design evolutions were needed in the battery rooms (ventilation improvements …).

4.2. Approach for design extension conditions without core melt

For the identification of the “Risk reduction category A” (so-called “RRC-A”) conditions (i.e. design extension conditions without core melt) and of associated RRC-A features, EDF developed a methodology combining deterministic and probabilistic approaches. The methodology, which was assessed by IRSN, uses the internal events level 1 PSA in order to adjust the preliminary list of RRC-A conditions established on deterministic basis and to check the appropriateness of the RRC-A features. It consists of several steps:

- Identification of potential RRC-A features (systems, equipment and operator actions) in the level 1 PSA (e.g. feed and bleed in case of total loss of the steam generator feedwater supplies);

- Identification of functional sequences associated with the potential RRC-A features (i.e. groups of sequences whose frequency could be reduced by a same potential RRC-A feature);
- Determination of the list of the plant RRC-A features important to safety: a potential RRC-A feature is important to safety if the core damage frequency of a functional sequence without crediting the potential RRC-A feature is higher than a given threshold (10^{-8}/r*y);
- For the functional sequences that require a RRC-A feature, deterministic safety analysis and associated thermal hydraulic calculations, with conservative rules but generally less stringent than for design basis accidents.

This methodology turned out to be a systematic way to highlight the features which are not considered in the design basis but necessary to cope with multiple failures situations, and then to determine the safety analysis to be performed in order to design these features. As the preliminary PSA was not completed enough at the design stage to capture the whole set of RRC-A conditions, an iterative process was necessary to confirm the set of conditions. For that reason, the designer is defining a different methodology in view of the new EPR model, which aims at fulfilling the same objectives but with less dependence on the level 1 PSA achievement.

4.3. PSA insights for "practical elimination"

The "practical elimination" of situations which would lead to large early releases can be justified by deterministic and probabilistic considerations, taking into account the uncertainties due to the limited knowledge on some physical phenomena [1]. The contribution of the probabilistic approach as a complement of deterministic methods and the balance between the two approaches depend on the situations under consideration. For Flamanville EPR, level 1 PSA has been used in the frame of the verification of the "practical elimination" of core melt accidents with containment bypass (related to system rupture outside containment) and of heterogeneous boron dilution situations (whereas level 2 PSA and Spent Fuel Pool PSA were used for other situations that should be "practically eliminated").

EDF PSA studies and the review by IRSN have highly contributed at each of the EPR stages to appreciate the "practical elimination" of the core melt accidents with containment bypass and the heterogeneous boron dilution situations that were not physically impossible by design. Level 1 PSA results have contributed to define design and operational features improvements (automatic signals to avoid by-pass by RHRS, automatic isolation of some water sources to avoid dilutions, specific alarms and surveillance requirements…).

Apart from the quantitative risk assessment of the situations, the PSA approach was very beneficial to deeply investigate in a systematic way all possible initiating events and accident sequences that could lead to such situations and to identify all the features (design, operational) involved.

It has to be underlined that the "practical elimination" cannot be demonstrated by the compliance with a "cut-off" probabilistic value and that deterministic demonstration is also requested. Additional justifications of the efficiency of the engineered features involved and their associated requirements are still expected by the EPR operation license.

5. CONCLUSION

Despite some challenges related to the development and use of PSAs for the EPR new built especially at the early design stage, PSAs have provided useful insights to confirm the design and even in some cases to point out the need for design improvements. In particular, PSAs have been beneficial to verify the suitability of the design regarding the risk significant scenarios, to check the sufficient independence between the defence-in-depth levels, to identify the situations to be included in the design extension category DEC-A and to contribute to the assessment of the situations that should be "practically eliminated".

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LATEST ACTIVITIES OF NUGENIA IN THE FIELD OF DETERMINISTIC AND PROBABILISTIC ASSESSMENT

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Abstract

The paper is focused at the activities of the NUGENIA association in the field of deterministic and probabilistic safety analysis (DSA, PSA). The NUGENIA is an association focused on research and development (R&D) activities for Generation II and III nuclear plants. The association is structured into 8 major Technical Areas (TA). In the Technical Area 1 "Plant Safety and Risk Assessment" approximately 25 projects were prepared up to now. Most of TA1 projects, being developed or running, are focused on advancements in deterministic and/or probabilistic assessment of NPP. This paper describes objectives, methods and interim results of these projects plus major trends in the field.

1. INTRODUCTION

The NUGENIA is a young and dynamic association of research organizations, utilities, technical support organizations (TSO) and universities devoted to R&D support of Generation II & III NPP's. It was founded in 2012 on the basis of several preceding projects - the SNETP Technology Working Group Gen II & III, the NULIFE and the SARNET projects. Scope of activities of the new association was extended to the whole spectrum of research for NPP installations plus support of EC and SNETP. After initial phase of the association life connected with forming of teams, discussions at the first annual Forum, performing priorities exercise and preparation of the NUGENIA Roadmap [1], the development of new project ideas was started. For more information on the association - see the NUGENIA web pages http://www.nugenia.org.

2. STRUCTURE OF NUGENIA

The work in NUGENIA is structured into 8 major Technical Areas [1, 2]:

TA1 Plant safety and risk assessment
TA2 Severe accidents
TA3 Core and reactor performance
TA4 Integrity assessment and ageing of systems, structures and components
TA5 Fuel, waste management and dismantling
TA6 Innovative LWR design& technology
TA7 Harmonisation
TA8 In service inspection and qualification

The Technical Area 1 (TA1) "Plant safety and risk assessment" is further subdivided into the following technical subareas (STA):

STA1.1 Data, methods and tools for risk assessment
STA1.2 Deterministic assessment of plant transients
STA1.3 Impact of external loads and hazards
STA1.4 Effect of electrical grid disturbances
STA1.5 Effects of human errors and reliability evaluation
STA1.6 Advanced safety assessment methodologies
STA1.7 Design of reactor safety systems
STA1.8 Pre-normative research

The content of this paper is based on DSA and PSA oriented activities and outputs of TA1 technical subareas STA1.1, STA1.2, STA1.5, STA1.6, and STA1.8.

3. SCOPE AND OBJECTIVES OF TECHNICAL AREA 1 “PLANT SAFETY AND RISK ASSESSMENT”

The primary objective of the effort in the Technical Area 1 of NUGENIA [1, 2] 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.

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. The 8 technical sub-areas of TA1 listed above address the following major objectives:

- 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 and progress towards multi-level and multi-physics computational capabilities;
- Advanced safety assessment methodologies (identification and reduction of all uncertainties plus increase of their predictability, optimization of safety margins etc.);
- Development of methods and tools to better insure complimentarily 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.

It is also worth to mention that composition and structure of TA1 gives to the R&D community an unique opportunity of continuous communication and cooperation of experts from deterministic and probabilistic fields and thus utilize the synergies of such a connection. It is another indirect but important objective of TA1.

4. PROJECT DEVELOPMENT PLATFORM - NOIP

The NUGENIA Open Innovation Platform NOIP (http://noip.nugenia.org) is an online platform for development of projects and sharing information among NUGENIA members on their project ideas. Among the major benefits of this platform belong its modern and open character, when all registered members of NUGENIA – about 1000 experts from more than 100 organizations - can at any time either enter the team around project idea or only follow the project idea, discuss the project, rate it etc.

Development of a project idea at NUGENIA NOIP flows in the following steps: submission of Template 1 – evaluation by TA leader (possible consultation and modification) finished by approval and publication of Template 1 at NOIP – development of team and project – submission of Template 2 – evaluation and approval by TA leader - publication of Template 2 at NOIP – evaluation of Template 2 by NUGENIA Execution Committee (ExCom) – appointing of ExCom contact person (CP) - awarding by NUGENIA LABEL.

Even after completion of this process at NOIP, the NUGENIA support to project continues – looking for funding opportunities, support of meetings, dissemination of results etc.

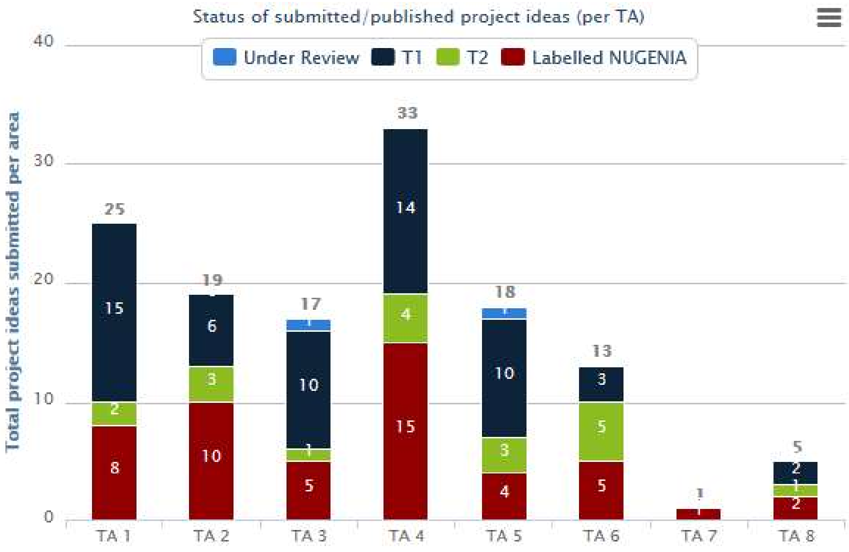

FIG. 1. Status of submitted/published project ideas at NOIP (3/2017).

5. DEVELOPMENT OF PROJECTS IN FIELD OF DETERMINISTIC SAFETY ASSESSMENT

An overview of all DSA oriented projects prepared in frame of NUGENIA is given in the list below. Several examples with more details follow. The projects are in various state of development – project idea, project proposal to a Call, running project, or lastly finished project disseminated through NUGENIA.

— NEWHAM (Nugenia Water Hammer);
— NONCOND (Noncondensable Gases Effects on Reactor Coolant System and their Improved Modelling and Analysis, [4]);
— CoupTH (Development of Coupled Computational Tools for Advanced Prediction of Local TH Phenomena and System Behaviour in Frame of PTS Evaluation);
— SPH-2PHASEFLOW (Simulation of two-phase flow patterns with a new approach based on Smoothed Particle Hydrodynamics);
— SOCRATES (Safety of Current Reactors – Applying Advanced Tools for Enhanced Safety);
— EXPRESS (Express Safety Analysis for Diversification/Modernization of Nuclear Reactor Core);
— McSAFE (High Performance Monte Carlo Methods for Safety Demonstration – From Proof of Concept to realistic Safety Analysis and to Industry-like Applications, TA3 crosscutting with TA1).

SPH-2PHASEFLOW

The aim of this project [5] was to develop a new numerical approach, based on the Smoothed Particle Hydrodynamics (SPH) method, to simulate changes in two-phase flow patterns. Two-phase flows play a key role in several processes in NPPs and therefore capturing changes in regimes of a two-phase flow is a major challenge with strong safety and efficiency implications. E.g. this issue is involved in studies of re-floods of damaged reactor cores, spray cooling, steam-generator operating conditions, as well in several practical concerns such as cavitation in turbo-machineries. The new numerical approach represents a radical departure from traditional methods and relies on a meshless, particle-based, method that has the potential to capture rapidly-deforming interfaces in two-phase flows.

Status: The SPH-2PHASEFLOW project proposal was prepared and submitted to the NUGENIA Plus Pilot Call in 2014, where it succeeded, was funded and run from 4/2015 to 10/2016.

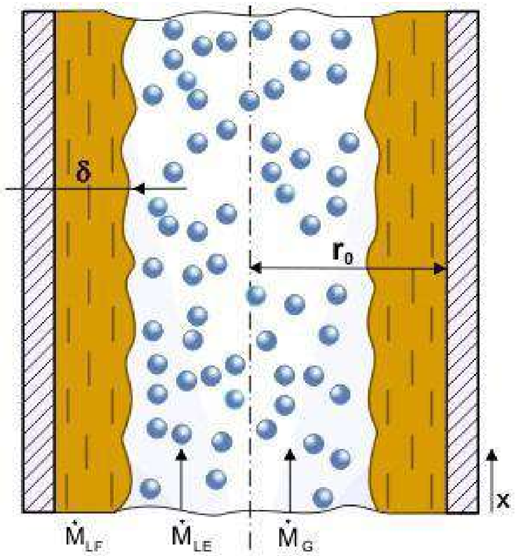

FIG. 2. Dispersed/separated two-phaseflow [5].

McSAFE

A running project [6] focused on further development of Monte Carlo (MC) methods for core analysis aiming to improve the performance and safety of current Gen-II and Gen-III reactors by developing and demonstrating novel and innovative methodologies of MC codes to include stable depletion, thermal hydraulic feedback, time dependent solutions and capability of simulate whole LWR cores at pin level making use of high performance computers (HPC) and finally delivering reference solutions for deterministic codes in case when no experimental data is available.

Status: Submitted to Horizon2020 Call 2016-2017, succeeded and got funding.

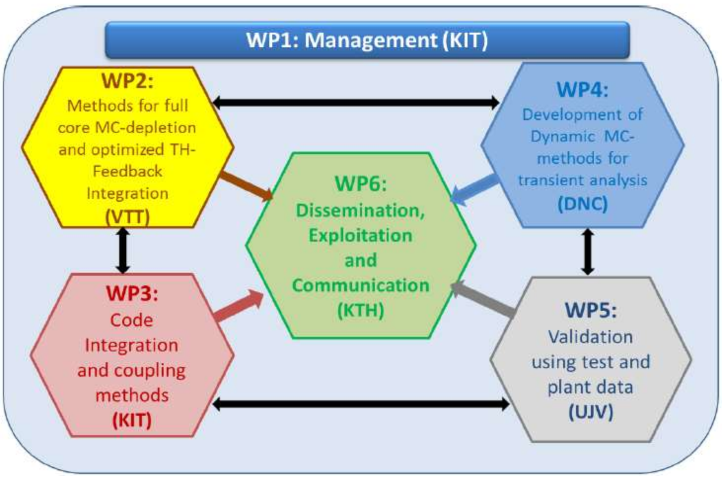

FIG. 3. Structure and Work Packages of McSafe project [6].

6. DEVELOPMENT OF PROJECTS IN FIELD OF PROBABILISTIC SAFETY ASSESSMENT

An overview of all PSA oriented projects prepared in frame of NUGENIA is given in the list below. An example with more details follows.

— HRA-in-RDM (Development, evaluation and application of methods for Human Reliability Analysis in the process of Risk informed Decision Making);
— SEMEPRA (Development of methods for risk assessment of seismic impact in low seismic risk areas);

— NARSIS (New Approach to Reactor Safety Improvements).

NARSIS

The project [7] aims to propose some elements of improvement to be integrated in the current PSA. Thanks to the diversity of partners constituting the consortium, from academic to operators and TSSO, the foreseen theoretical developments and the effectiveness of the proposed improvements will be tested and validated (numerical simulations, experiments) on simplified and real NPP case studies. The project aims at applying its outcomes at the demonstration level by providing improved supporting tools for operational and severe accident management.

Status: Submitted to Horizon2020 Call 2016-2017, succeeded and got funding.

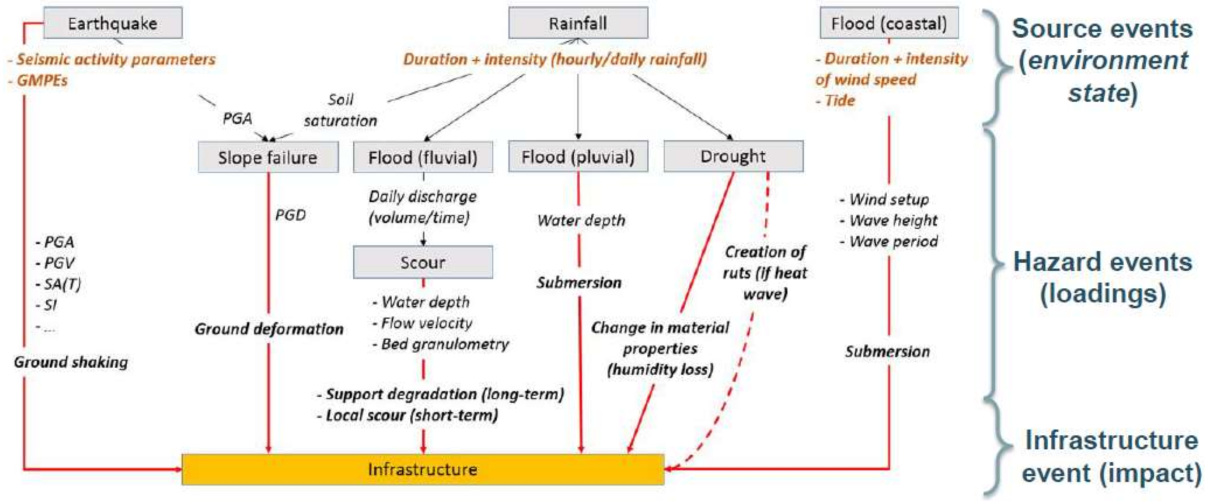

FIG. 4. Fragility in multi-hazard context – transport infrastructure (NARSIS [7]).

7. DEVELOPMENT OF PROJECTS COMBINING DETERMINISTIC AND PROBABILISTIC SAFETY ASSESSMENT

An overview of projects combining PSA and DSA and prepared in frame of NUGENIA is given in the list below. An example with more details follows:

- IDPSA (Integrated Deterministic – Probabilistic Safety Analysis);
- BEPILOT (Best Estimate Risk-Informed Pilot Safety Analyses);
- PROSAFE (Harmonized understanding of uncertainties and their propagation in Probabilistic evaluations on Safety margin assessments of nuclear reactor pressure vessels and piping);
- DEFI-PROSAFE (Definition of reference case studies for harmonized Probabilistic evaluation of Safety margins in integrity assessment for long-term operation of reactor pressure vessel).

DEFI-PROSAFE

The objective of the project [3] was to develop a probabilistic integrity approach to become a best estimate of the remaining margin as well as to create deck files for CATHARE/RELAP5/ATHLET.Reference case studies are defined for a benchmark dedicated to investigating the propagation of the uncertainties from the boundary conditions through the structural integrity, in order to get insights in the total safety margin. The work will be based on the lessons learned from earlier projects, such as PROSIR, IAEA-CRP9, FALSIRE, ICAS, and from a review of the work performed in NURESIM, NUREG 1806 and published UPTF results.

Status: The DEFI-PROSAFE project proposal was prepared and submitted to the NUGENIA Plus Pilot Call in 2014, where it succeeded, was funded and ran from 4/2015 to 10/2016.

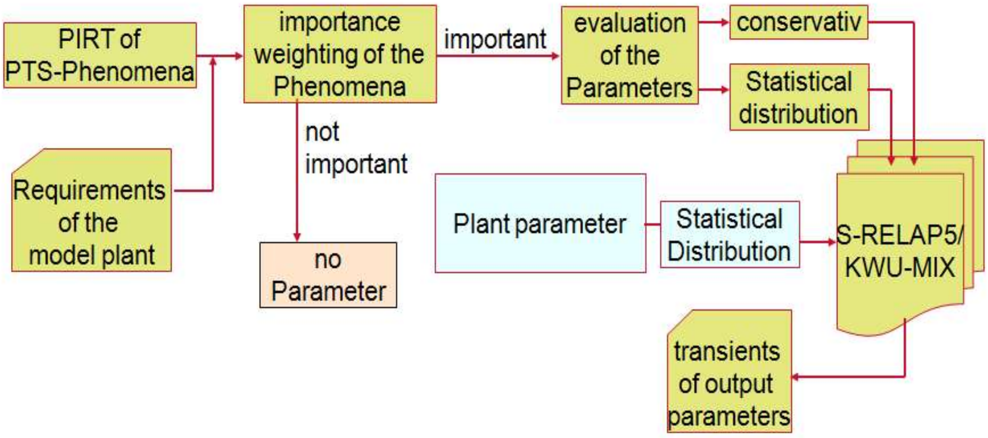

FIG. 5. Development of Statistical Fracture Event Methodology (DEFI-PROSAFE [3]).

8. OTHER RELEVANT PROJECTS IN NUGENIA PORTFOLIO

It is worth to note also the EU FP7 projects already running at time of foundation of NUGENIA, that were integrated into the NUGENIA project portfolio:

- NURESAFE (Simulation Platform for Nuclear Reactor Safety);
- MOTHER (Modelling T-Junction Heat Transfer);
- ASAMPSA_E (Advanced Safety Assessment Methodologies: Extended PSA).

For more information about these projects see the “Project Portfolio” or “Library” sections of the NUGENIA web pages (http://www.nugenia.org).

9. SUMMARY

The paper gives a brief overview of safety and risk assessment oriented activities and projects prepared in Technical Area 1 (TA1) of the NUGENIA. After basic information about the NUGENIA association and the TA1 “Plant Safety and Risk Assessment” all the DSA and PSA oriented projects – either being prepared or running or lastly finished – are briefly described. Some of the projects are discussed with more details and examples of result.

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SAFETY CLASSIFICATION: IMPLEMENTATION OF DEFENCE-IN-DEPTH CONSIDERATIONS AND PROBABILISTIC CRITERIA

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Abstract

New Russian regulations [2], published in 2016, introduce modernized rules for classification of NPP systems and elements. Updated classification rules take into account IAEA requirements, experience of existing classification rules application, the results of safety analyses (primarily results of PSA studies) and operational experience (including Fukushima accident experience). The paper shows that the new classification criteria reflect classified element impact on NPP safety. Also the paper demonstrates compliance of the new classification criteria with contemporary international practice reflected in IAEA standards and in various national regulatory documents from different countries.

All the NPP items important to safety shall be classified as stated in Requirement 22 of the IAEA Safety Standard [1]. Classification is required to present graded regulatory requirements for different systems and elements of a nuclear power plant depending on (a) their functions, and (b) their importance to safety. According to [1], the NPP items classification shall be established with a due account taken of factors such as: safety function to be performed by the item; the consequences of failure; the probability of a safety function to be performed by an item; the time following a postulated initiating event at which or the period for which, the item will be called upon to perform a safety function.

Strictly speaking, appropriate safety requirements can be developed without a classification, but the use of the classification allows development of safety requirements in a less cumbersome way and to make them better structured and traced.

Obviously, the classification of items (systems and elements) by their NPP safety importance should consider ranking when stricter regulatory requirements are applied to the items of the highest safety importance for the NPP. Meanwhile, grading of the regulatory requirements by functions (for instance, the systems are divided into protective, control, and supporting) performed by an item (system, element) may not be of the ranking nature.

In 2016, Russia introduced new General Provisions on Ensuring Nuclear Power Plants Safety, NP-001-15 [2], to replace the predecessor regulation [3] which had been in force for 19 years. The new document modifies the previous rules of NPP systems and elements classification.

Both NPP systems and elements are primarily ranged by [2] by their importance to safety, there are two grades: systems (elements) which are important to safety, and systems (elements) which have no safety impact. Table 1 provides classification rules and appropriate classification features established in [2].

TABLE 1. THE RULES OF ATTRIBUTION OF NPP SYSTEMS (ELEMENTS) TO THOSE IMPORTANT TO SAFETY [2]

Rule No.	Contents of a rule	Applied classification features
1	Affiliation with safety systems (elements)	Involvement of a system (element) into implementation of the third level of the Defence-in-Depth (DiD). Decrease in NPP preparedness for design basis accidents (degradation of the third level of the DiD) in case of a system (element) failure.
2	Affiliation with systems (elements) of normal operation whose failure jeopardizes normal operation of a plant, if the conditional possibility of this failure developing into a severe accident is 10^{-6} or higher	Involvement of a system (element) into implementation of the third level of the DiD. Degradation of the DiD first level in case of a system (element) failure, the need in actuation of systems related to the further DiD levels due to the failure of an NPP element.

Rule No.	Contents of a rule	Applied classification features
		The probability of inefficient operation of further DiD levels (conditional possibility of a system (element) failure developing into a severe accident).
3	Affiliation with systems (elements) of normal operation whose failure prevents from elimination of operational occurrences at a plant, if the conditional possibility of this failure developing into a severe accident is 10^{-6} or higher	Involvement of a system (element) into implementation of the second level of the DiD. Decrease in NPP preparedness for reacting to abnormal operation (degradation of the second level of the DiD) due to a failure of an NPP system (element). The probability of inefficient operation of other DiD levels (conditional possibility of a system (element) failure developing into a severe accident).
4	Affiliation with NPP systems (elements) whose failure leads to excess of established values for maximal permissible releases or permissible discharges of radioactive substances, or permissible levels of NPP premises radioactive contamination	Involvement of a system (element) into implementation of the third level of the DiD. Radiological consequences of a system (element) failure.
5	Affiliation with systems (elements) provided in the NPP design for accident management during first three days after an initiating event (or during another period determined in the NPP design which shall be at least three days);	Involvement of a system (element) into implementation of the fourth level of the DiD. Time period since the moment of an initiating event occurred after which a system (element) should be in operation.
6	Affiliation with the systems (elements) of radiation monitoring.	Involvement of an NPP element into implementation of the DiD (levels 2-5 of the DiD) as regards to the monitoring of radiation parameters

Thus, [2] applies the following classification features to define safety importance of a system (element):

— Involvement of a system (element) into implementation of the second level of the DiD;
— Probability of inefficiency of the DiD levels, other than the level directly degrading in case of a failure of a classified system (element), i.e. conditional probability of a system (element) failure development into a severe accident; and
— Time period since the moment of an initiating event occurred after which a system (element) should be in operation.

The first of the features is the main one (according to [2], only systems involved into the DiD are considered to be important to safety; classification rules Nos. 1-6 given in Table 1 provide classification features related to all the levels of the DiD).

The second and third features are specifying and they allow not to consider systems and elements as important to safety (although they are involved into the DiD but have minor impact on NPP safety) because in case of a failure they stay operable, and engineering and organizational measures are sufficient to ensure extremely high probability of non-development of such a failure into a severe accident (Classification Rules Nos. 2&3), or their failure does not impact the accident behavior in the course of the first three days after an initiating event and, if required, such systems (elements) can be simply replaced by other operable systems (elements) (Classification Rule No.5).

It is worth mentioning than ranking per Table 1 is not always sufficient. Safety important systems (elements) specified among others with the help of the rules provided in the a.m. table may have significantly different impact on NPP safety.

While manufacturing and operating equipment (elements), stricter requirements to reliability assurance shall be applied to equipment (elements) with higher impact on NPP safety. It is reasonable to perform further regulatory ranking of equipment (other elements) by its safety impact. We would like to emphasize the fact that for the grading purposes of reliability assurance during manufacturing and operation the ranking shall be applied to elements but not to systems because the systems contain elements that may have a different impact on NPP safety.

For this purpose, General Provisions [2] introduces NPP elements classification, i.e. elements are related to one of the four safety classes in line with the classification rules provided in Table 2.

TABLE 2. THE RULES OF ASSIGNMENT NPP ELEMENTS TO SAFETY CLASSES

Safety Class	Classification rule	Applied classification features
1st	Fuel elements and elements whose failure is an initiating event for a Beyond Design Basis Accident (BDBA) leading under a design operation of safety systems to damage of fuel elements beyond the limits established for design basis accidents.	Failure of the first, second, and third levels of the DiD due to a failure of a classified element.
2nd	Elements whose failures are initiating events which under design functioning of safety systems lead to damage of fuel elements within the limits established for design basis accidents	Failure of the first and second levels of the DiD due to a failure of a classified element.
2nd	Elements of safety systems with single failures resulting in violation of design limits established for the design basis accidents when such accidents occur.	Failure of an element leads to NPP unavailability for overcoming a design basis accident (failure of the third level of the DiD due to a failure of a classified element).
3rd	Safety important elements other than safety classes 1 and 2.	Assignment of an element to those important to safety (or non-assignment).
4th	Elements which do not have safety impact and are not classified as safety classes 1, 2, and 3.	Assignment of an element to those important to safety (or non-assignment).

It is clearly seen that different classes (from the first to the third) are applied to safety important elements depending on consequences of a failure of such a classified element for the DiD, i.e. if due to a failure three levels of the DiD degrade (from the first to the third), then the element is assigned to Class 1; if due to a failure of an element the first and second levels of the DiD or the third one become inefficient, then the element is assigned to Class 2. In all other cases safety important elements are referred to Safety Class 3. Elements which are not important to safety are referred to as Class 4.

Thus, primary ranking of systems (elements) by their safety impact is made in [2] based upon a deterministic criterion which considers involvement of a system (element) in the DiD with two specifying features applied, probabilistic and temporal (see Table 1). Further additional ranking of NPP elements applies additional (in relation to those provided in Table 1) classification features related to the scope of impact on the DiD as a whole caused by a failure of classified NPP elements.

A number of regulatory requirements provided in the Russian rules and regulations (both in [2] and others) cover safety important systems (elements) and do not cover NPP systems (elements) which have no safety impact (this, using the basic ranking of systems and elements given in Table 1). Alongside, a number of regulatory requirements apply additional ranking of NPP elements, i.e. safety classes (assigned in line with the rules provided in Table 2), requirements to analysis of reliability of elements [2], requirements for production, operation and control of equipment (in particular, [5]), seismic resistance requirements [6], etc.[1]

[1] Rules [5] and [6] apply additional classifications which take into account the safety classes established in line with [2].

In addition to the ranking of systems and elements given in Tables 1&2, the General Provisions [2] provide other classifications (which are not ranking): systems (elements) are divided into systems (elements) of normal operation, safety systems (elements) and special technical means for BDBA management, safety systems (elements) are also divided into protective, localizing, control, and supporting ones.

Thus, the classification rules for systems and elements provided in [2] comply with the requirements of the IAEA standards [1], [9]; the Russian classification rules base on deterministic approaches supplemented, if necessary, with probabilistic considerations and taking into account the temporal factor (thus, the classification takes into account all the four aspects which shall be accounted as per requirement 22 [1]). However, there are differences from the recommendations of SSG-30 [10].

Let us highlight the following case of use of the classification. When there is a need to grade regulatory requirements by implementation of the independence, redundancy and diversity principles, the object of classification shall not be NPP equipment or elements, nor the NPP systems because safety functions subject to the requirements related to the a.m. principles are implemented by functional groups which may or may not be equivalent to the systems. For instance, a functional group may consist of several systems, a system train, may be an aggregate of several channels of different systems, and other options are also possible. In that case functional groups are subject to ranking. Such an approach (with functional groups of categories A, B, and C) implemented in 2016 in the requirements for safety important control systems [7] which take into account the provisions of both the IEC standard [8] and the Russian regulatory documents. Whilst, the mandatory character of redundancy and diversity principles is established for the groups of category A; the possibility of refusal to follow these principles if appropriate justification is available is established for the functional groups of category B; and optionality of implementation of these principles is established for the functional groups of category C.

It is necessary to mention common features in the Russian classification rules provided in [2] and actual regulatory requirements of other countries, for instance, Finland. In line with the guidance [11], NPP systems are divided into Classes 2, 3, EYT/STUK and EYT; and the first three classes of the systems are considered as important to safety, and Class EYT involves systems which do not have safety impact. Only the systems of Class 2 may be considered as available in the analysis of design basis accidents. In case of this approach, assigning a system to Class 2 is similar to assigning a system to the safety systems in the Russian regulations. For the analysis of anticipated operational occurrences, NPP safety assurance may be considered with the account taken for the operation of other systems. EYT/STUK Class is actually a supplement to Class 3 (the regulatory requirements for the systems of these classes do not differ significantly). But if deterministic ideas are applied when assigning systems to Class 3 (as it is in case of Russia, such ideas are related to the DiD), then the rules of assignment to Class EYT/STUK also contain probabilistic criteria. As per [11], structures and components are related to the same safety class as the system. However, the class of a component (structure) may be decreased if it is demonstrated that there is no (or low) impact by the component on operation of the system, and it may be increased if the component is at the boundary with a system of a higher class. [11] also provides additional rules for classification of components and structures. In compliance with the rules, the reactor pressure vessel and other primary equipment whose failure has intolerable consequences are assigned to Class 1, while equipment whose failure has a threat of uncontrolled chain fission reaction is assigned to Class 2.

The analysis demonstrates that the totality of classification rules for systems, components, and structures provided in [11] considers involvement of a classification object into the DiD. It also considers probabilistic and a number of other ideas. The classification is made at the level of systems and other specific elements. All these things serve as evidence of common features of the Russian and Finnish classification rules.

The regulatory documents of Hungary apply different approaches to NPPs in operation (provided in [12]) and new NPPs ([13]). To a large extent, the approach suggested by [12] is similar to that of [3]. Safety functions (Classes F1A, F1B, F2), physical barriers (levels B1, B2, B3) are classified in compliance with [13]. The classification rules for systems and components are developed as unified ones (Classes ABOS1, ABOS2, ABOS3 are considered for systems and components with safety impact. Class ABOS4 is applied to systems and components with no safety impact. Classification criteria are related to the DiD. Probabilistic criteria or ideas related to time are not formulated in a clear way. [13] also contains instructions that state probabilistic criteria should be applied as auxiliary ones together with deterministic criteria. It is obvious that the approach suggested by [13] has similar features with the classification approach implemented by [2], but there are significantly less similarities than in case the Finnish regulation [11].

CONCLUSION

The approach to classification of systems, elements and functional groups implemented in newly introduced Russian Federal rules and regulations [2], [7] complies with the provisions of the IAEA standards SSR-2/1 [1] and GSR Part 4 [9]. However, it has some differences in comparison with the approach provided in SSG-30 [10]. The accumulated experience of classification rules provided in [3] was taken into account in the process of development of the classification rules of [2]. The classification approaches suggested by [2] have similar features as the approaches provided in the requirements for classification of NPP systems and elements of some European countries [11-13].

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RISK INFORMED DECISION MAKING

Chairperson

P. KRAL
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RISK-INFORMED DESIGN FOR UK ABWR PROJECT

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Abstract

Hitachi-GE is proposing to build multiple ABWR plants in the UK, based on an enhanced Japanese ABWR design, incorporating lessons learned from the accident at the Fukushima-Daiichi NPP. A key aspect of the UK expectations is to demonstrate that a risk from a proposed facility is As Low As Reasonably Practicable (ALARP). Hitachi-GE has developed a full-scope PSA for UK ABWR and used the PSA to inform the design process as well as demonstrate further evolution to ensure the design is ALARP. The paper introduces the scope of the UK ABWR PSA, use of peer review to support the PSA development, and the process and examples of risk-informed improvements during and after the PSA development.

1. INTRODUCTION

The Advanced Boiling Water Reactor (ABWR) [1], [2] was developed as an international project using multiple Architect Engineers including Hitachi-GE and GE-Hitachi, and was constructed as a Generation III plus reactor in both Japan and Taiwan [3].

Hitachi-GE is proposing to build multiple ABWR plants in the UK, based on an enhanced Japanese ABWR design, incorporating lessons learned from the Fukushima-Daiichi accident. A key aspect of the UK expectations [4] is to demonstrate that a risk from a proposed facility is As Low As Reasonably Practicable (ALARP). Hitachi-GE has developed guidance for the preparation of a safety case and methodology for the generic and consistent application of ALARP assessments across all areas in UK ABWR Generic Design Assessment (GDA). Fig. 1 provides a high-level overview of this approach. This process has been performed throughout the development of the UK ABWR using a variety of measures to assess risk [5].

The specific role of Probabilistic Safety Assessment (PSA) in supporting the ALARP assessment is defining the risk targets to be used, and scope of PSA in the delivery of numerical results for these targets that are necessary for the ALARP assessment. Hitachi-GE has developed a full-scope PSA for UK ABWR and used the PSA to inform the design process as well as demonstrate further evolution to ensure the design is ALARP. The paper introduces the scope of the UK ABWR PSA for GDA, use of peer review to support the PSA development, and the process and examples of risk-informed improvements.

2. SCOPE OF UK ABWR PSA IN GENERIC DESIGN ASSESSMENT

For the UK ABWR GDA, the PSA is used to demonstrate that the generic design meets numerical risk targets defined in the UK principle [4] and to inform the design organizations of potential vulnerabilities that guide any potential improvements as part of the ALARP demonstration.

The results of the PSA consequence analyses are combined with accident frequency information to derive the summed risk values, which are then compared with the numerical targets defined in the UK principle [4]. Basic Safety Limits (BSLs) and Basic Safety Objectives (BSOs) are established for each numerical target. The BSLs establish effective limits for acceptability for the installation.

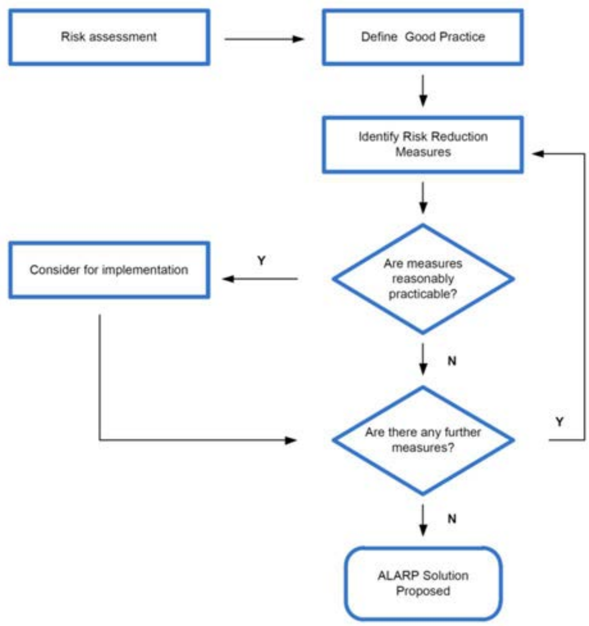

FIG. 1. High-level overview of ALARP process.

The BSOs were established to represent broadly acceptable levels beyond which regulatory resources will generally not be used to seek further improvements. However, it is understood that ALARP considerations must still be evaluated at such levels. The PSA also utilizes Core Damage Frequency, Fuel Damage Frequency, Large Release Frequency and Large Early Release Frequency as intermediate measures for risk-informed improvements.

The scope of UK ABWR PSA in GDA includes the following three inputs:

Sources of radioactivity at the facility: Sources of radioactivity are identified as fuel in the reactor, fuel in the Spent Fuel Pool (SFP), spent fuel in transfer (in the fuel route), sources in the Main Turbine and radioactive waste system, and irradiated structures and materials.

a) Types of initiating faults: Internal events are identified in a systematic manner, including logic tree analysis and review of existing analysis, for each source of radioactivity. Design Basis initiating faults are identified and treated through the process of internal events, Internal Hazard and External Hazard analysis. In the PSA, each of these initiating faults is reviewed as the starting point, and hazards whose risk should be quantified by PSA are identified through the hazard identification process by following the approach defined in IAEA SSG-3 [6].
b) Operational modes: In the analysis for fuel in the reactor, the at-power operation mode and all shutdown modes are taken into account. The low power mode is also addressed.

The scope is summarised as follows:

Internal Events at Power (Level 1 - Level 3);
Internal Events during Shutdown (Level 1 - Level 3);
Internal Events for the Spent Fuel Pool (Level 1 - Level 3);
Internal Fire at Power (Level 1 - Level 3) during Shutdown and for SFP (scoping analysis);
Internal Flood at Power (Level 1 - Level 3) during Shutdown and for SFP (scoping analysis);
Seismic Hazards at Power, during Shutdown and for SFP (Level 1 - Level 3);
Bounding assessments to determine the risks from Fuel Route, Tornado Missiles, Turbine Missiles and Accidental Aircraft Impact.

3. PEER REVIEW

GE-Hitachi organized and performed multiple peer reviews of the UK ABWR PSA using a similar process as performed in the US (i.e., NEI 05-04 process). Their purpose was to demonstrate that the UK ABWR PSA meets international PSA standards, including adequate methods and a complete PSA scope.

The UK ABWR PSA was reviewed against the UK regulatory expectation [7] as well as the US and IAEA standards to help determine the overall adequacy of the PSA. The UK regulatory expectation [7] describes that the UK PSA expectation is "broadly consistent" with those of IAEA and ASME/ANS standards.

The PSA peer review is a tiered process. The reviewer begins with a high level examination, and progresses successively to additional levels of detail as necessary to ensure the robustness of the model until all of the requirements are adequately reviewed. The UK regulatory expectations and the PSA Standard's Supporting Requirements (SRs) provide a structure which, in combination with the reviewers' experience, provides the basis for examining the various technical elements. The SRs help to ensure completeness in the review. If a reviewer discovers a question or discrepancy, a more thorough and detailed search is conducted.

The peer review process includes the documentation of any peer review comments, as well as resulting responses. A draft product for a specific peer review had an initial set of comments generated from the initial review, and these comments are then be addressed prior to the final review session. During the follow-on final review session, additional comments are generated based on the updated information. The disposition of the previous comments is noted and new comments are also generated for the finalized product as necessary.

4. RISK-INFORMED IMPROVEMENTS DURING DEVELOPMENT OF UK ABWR PSA

The UK ABWR PSA for GDA was developed along with the development of generic design improvements. Where sufficient PSA input was not available, assumptions were developed and recorded. Based on the preliminary PSA results, several design improvements were either recommended or specific design options were defined. Recommended and adopted design improvements were captured in the PSA as design assumptions. The following paragraphs provide the examples of improvements during the PSA development.

The result of the preliminary Internal Evens At Power (IEAP) PSA indicated that the dominant minimal cutsets included a Common Cause Failure (CCF) of digital Control and Instrumentation (C&I) system. This insight was provided to the designers, and a hardwired Human Machine Interface (HMI) was introduced to one of Class 1 Emergency Core Cooling System (ECCS) divisions [8] in addition to original digital HMI.

The result of the preliminary IEAP PSA indicated that the dominant minimal cutsets included a CCF of the chillers which support the local cooling units of Emergency Diesel Generator (EDG) buildings and Heat Exchanger Building (HxB). This PSA insight was provided to the designers, and a backup emergency supply fan (once-through type) was introduced to each EDG building and division of HxB building [9], although it was not required in the deterministic safety case.

During the development of the internal events SFP PSA, consideration was given to the significant benefit of SFP injection by fire protection pump. A design assumption was developed such that diesel-driven fire protection pump is capable of injecting water into the SFP although it is not required for the deterministic safety case of UK ABWR. The design process is ongoing to capture this assumption in the detailed design.

During the development of the Seismic PSA, consideration was given to the significant benefit of designing specific instrumentation pipework such that the resonant frequencies are sufficiently different among the redundant lines, which prevents simultaneous failures of multiple instrumentation divisions. This design enhancement was put in the assumption list and applied to the Seismic PSA. The design process is ongoing to capture this assumption in the detailed design.

High-level cable routing was determined at the beginning of the UK ABWR Internal Fire PSA for GDA based on deterministic assessment / design requirement. During the progress of Internal Fire PSA, critical cables were identified which greatly impacted the fire risk, although the cable routing satisfied the deterministic requirements such as divisional separation. Feasibility of re-routing these cables was discussed and some were assumed re-routed and captured in the design assumptions. This resulted in a significant reduction of the risk from internal fire. The design process is ongoing to capture these assumptions in the detailed design.

Bounding assumptions on the component critical heights were developed at the beginning of the UK ABWR Internal Flood PSA for GDA. During the progress of Internal Flood PSA, critical components were identified which greatly impacted the flood risk, although the layout design satisfied requirements for divisional separation. Feasibility of raising the heights of these critical components was discussed and some of them were implemented in the Internal Flood PSA as design assumptions. This resulted in a significant reduction of the risk from internal flood. The design process is ongoing to capture these assumptions in the detailed design.

5. RISK-INFORMED IMPROVEMENTS AFTER DEVELOPMENT OF UK ABWR PSA

After the development of the UK ABWR PSA, it has been used to demonstrate that the risk associated with the design and operation of the UK ABWR is ALARP and that it can be concluded that there are no further reasonable, practicable improvements identified for the generic plant design during GDA. The identification of risk characteristics and insights has been performed separately for each fault group since the results have different levels of detail and levels of uncertainty. The differing level of uncertainty is largely due to the availability of details in design information during the design phase.

The general process can be described in two steps: (1) A systematic review of each fault group is performed in support of the ALARP demonstration. This includes a systematic review identifying any plant vulnerabilities and major modelling uncertainties in the PSA or other improvements that could be made in the plant design or operation to reduce the plant risk. (2) PSA insights affecting the ALARP determination are established for risk significant characteristics. The results from each of the faults are reviewed systematically to determine if improvements could be made to the design or operation of the facility to make the risks as low as reasonably practicable. This process is shown in Fig. 2. The following paragraphs provide the examples of risk-informed improvements adopted in the GDA phase or recommended for the detailed design phase.

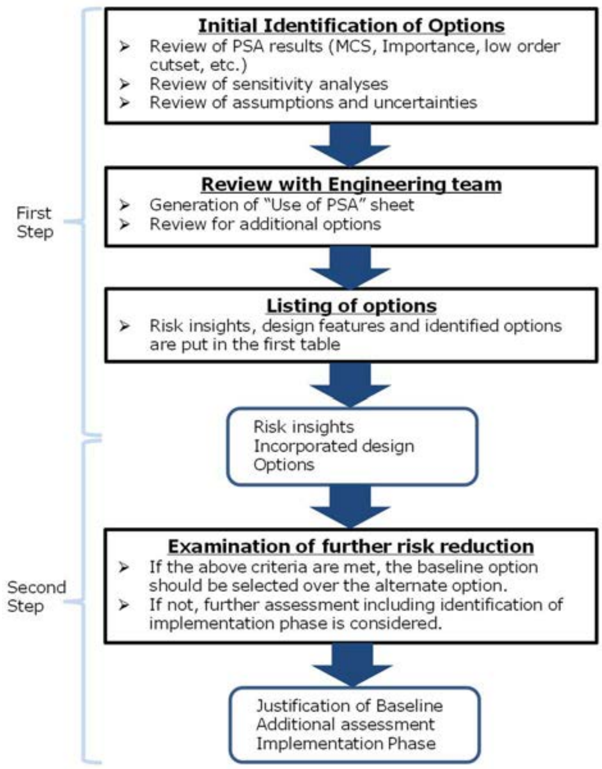

FIG. 2. Process of ALARP Assessment Support from UK ABWR PSA.

The IEAP PSA identified the importance of long term operability of Safety Relief Valves (SRVs). It was recommended to ensure that the nitrogen leak rate from a SRV unit is small such that the SRVs will not reclose during the PSA mission time without refilling. It was agreed to reflect this to a performance objective of a SRV.

The IEAP PSA identified a specific pipe segment which had relatively large contribution to the PSA risk from Interfacing System LOCA (ISLOCA). Although the risk from ISLOCA was already small, a recommendation was raised from the PSA to increase the thickness of that pipe segment for further risk reduction.

That recommendation was as a result of the consequence of core damage following an ISLOCA which could directly lead to Large Release, e.g., the containment is bypassed. Feasibility of this design enhancement will be assessed in the detail pipework design.

The internal events Shutdown PSA and SFP PSA identified risk significant Human Failure Events (HFEs), since the degree of automation for accident mitigation systems are generally low during the outage or for SFP operation compared to those for reactor at power. It was recommended to develop procedures and training to minimize human errors associated with these HFEs.

The internal events SFP PSA identified a human-induced initiating event that was potentially more significant than any other initiating event. This insight was provided to the designers and it was agreed to introduce measures to avoid this initiating event in the detailed design phase (e.g., using interlocks).

The Internal Fire PSA found that fire scenarios impacting multiple fire compartments within a safety division contributed significantly to the overall fire risk. The design of intra-divisional boundaries was not specified during the PSA development and thus the intra-divisional boundaries were assumed all non-fire rated. This insight was one of the drivers to apply fire rating to many intra-divisional boundaries.

The Seismic PSA in GDA was based on generic fragility curves derived from the assumed generic hazard curve. Some SSCs were found to have relatively large risk contributions. This insight was provided to the designers and will be used in plant specific aseismic design based on plant specific hazard curve.

6. USE OF PSA TO SUPPORT DESIGN OPTIONEETING AND DESIGN DIRECTION

Sections 4 and 5 introduced the process and examples regarding the improvements recommended from the PSA insights. Conversely, the PSA models and results have been used to judge the risk impacts including potential risk increase from design changes considered in design organizations. This activity includes scoring of design options from the PSA viewpoint and rejecting other specific proposals based on minimal impact on the risk results. For example, relocation of some electrical cabinets was rejected since the PSA suggested this could significantly increase risk from internal fire.

7. CONCLUSION

As part of the process for UK ABWR generic design, the full-scope PSA has been developed, peer-reviewed and used to inform the design process in GDA. Assumptions and recommendations raised from the PSA were captured and will be revisited in beyond GDA project, and used to inform detailed design process.

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USE OF THE BOWTIE METHODOLOGY IN THE GENERIC PRE-CONSTRUCTION SAFETY REPORT (GDA PCSR) FOR ADVANCED WATER COOLED NPPS

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Abstract

In the paper, the Author explores some potential advantages, that in his opinion, the Bowtie methodology can bring to both Responsible Party and Prospective Licensee when it is incorporated in the Generic Pre-Construction Safety Report development strategy. These advantages will contribute to the development of a good quality, usable and fit-for-purpose Safety Report and facilitate its development into a site-specific Pre-Construction Safety Report. This will allow the Prospective Licensee to fulfill its legal duties in relation to the management of safety during the subsequent site licensing phase and throughout the entire plant lifecycle. The conduct of highly interactive Bowtie workshops for the reactor bounding faults, facilitated by the Prospective Licensee's own technical staff and having the input from Suitably Qualified and Experienced Persons from the Requesting Party is also recommended. This process is a good vehicle for enhancing the transfer of knowledge from the Requesting Party and fostering discussions on plant design safety, both of which are essential for the creation of the Licensee's Design Authority capability. The advantages are exemplified with the help of a Bowtie diagram for High Pressure Core Flooder System pipe break, which is the bounding fault for Loss of Coolant Accidents within the containment for a generic Advanced Boiling Water Reactor. The recommendations contained in the paper are equally applicable to the development of the site-specific Safety Report by the Prospective Licensee during the site nuclear licensing phase.

1. INTRODUCTION

The Generic Pre-Construction Safety Report prepared by the Requesting Party (RP) during the Generic Design Assessment Process (GDA PCSR) plays a crucial role in the safety demonstration for Advanced Water Cooled Nuclear Power Plant design. However, since the Prospective Licensee (PL) has the ultimate responsibility in law for ensuring safety of the plant, it is also legally responsible for the Safety Report and its adequacy. The UK Office for Nuclear Regulation (ONR) clearly states that the GDA PCSR must be developed by the RP with a Licensee's legal duties in mind, not as a means to satisfy ONR [2].

The ONR encourages PL to make arrangements with the RP for it to be involved in GDA whenever possible, as this will be of significant benefit in being able to demonstrate (among others), during the nuclear site licensing process, an understanding of the safety case and an adequate knowledge of the plant's hazards and how to control them [2].

ONR expects that by the end of GDA, the RP writes its final submissions in such a way that they can be readily usable for a PL as part of the site-specific Safety Report. The ONR assesses the degree to which the PL understands and takes responsibility for the Safety Report in question [2].

2. ENHANCING LICENSEE CAPABILITY FOR PRODUCING A GOOD QUALITY SAFETY REPORT

The primary purpose of the Safety Report is to provide the PL with information required to enable safe management of the facility. Therefore, the ability to fulfil ONR's expectations regarding the production of a good quality Safety Report needs to be regarded as an important element to ONR in determining that a 'licensable' organization is in place and permitting activities on site. ONR has established clear expectations regarding the overall qualities of a good Safety Report [3].

The PL must always bear in mind that a technically sound Safety Report is not the same as a 'usable' and 'fit-for-purpose' Safety Report. It is then the PL's responsibility to determine the precise content of the site-specific Safety Report and develop the GDA PCSR into a robust site-specific version.

The early involvement of the PL in the GDA process must then be regarded as an invaluable opportunity to influence the PSCR development strategy and develop important capabilities which will be needed to produce a usable and fit-for-purpose site-specific Safety Report.

3. THE USE OF THE BOWTIE METHODOLOGY IN THE GDA PCSR

The Bowtie methodology is a qualitative risk management tool that has been suggested as a good method for increasing the value of the Nuclear Safety Reports for the management of nuclear safety risks [6,7,8]. Amongst the advantages that the Bowtie methodology can bring to the RP and PL in relation to the PCSR are:

— Improved understanding, visibility and accessibility of complex Safety Reports;
— A qualitative risk analysis tool that facilitates workforce involvement and ownership;
— A tool that links fault/risk analysis with the management system
— A barrier (safety measure) and knowledge management tool
— A tool that facilitates improved regulatory oversight

The essentials of the Bowtie methodology can be explained and its advantages for the GDA Safety Report realised with the help of a sample Bowtie diagram (see Fig. 1) which is the principal outcome of the Bowtie methodology. It depicts an accident scenario associated with a High-Pressure Core Flooder System (HPCF) pipe break which is regarded as a design basis infrequent fault and the bounding fault for all Loss of Coolant Accidents (LOCA) within the containment in a generic ABWR reactor.

The Bowtie model consists of different elements, all of which contribute to the build-up of the overall risk picture. The upper box of the Bowtie diagram represents the Hazard (i.e. Reactor Power Operation) which is one of the reactor operating modes with the potential to cause damage to the barriers confining the nuclear fuel (i.e. fuel cladding), if not properly controlled. The central knot of the Bowtie represents the Top Event (e.g. HPCF pipe break) which is the point in time when the control over the hazard is lost (i.e. the 'release' of the hazard) causing damage to the barrier.

The left-hand side of the Bowtie depicts the Threats (e.g. corrosion) that are the possible direct causes of HPCF pipe break. Safety measures are then identified (e.g. design using materials with low susceptibility to corrosion, etc.), which are in place to eliminate the threats entirely or prevent the threats from causing the Top Event. The right-hand side of the Bowtie depicts the fault sequence leading to consequences resulting from the HPCF line break and directly ending in some form of damage (i.e. core damage). Safety measures (protection systems) are then identified which reduce the likelihood of the consequence owing to the Top Event being "released" (HPCF pipe break) or mitigate the severity of the consequence (e.g. fuel core damage).

In addition, the bowtie model explores the escalation factors, i.e. 'failure modes' that can defeat or reduce the effectiveness of a particular safety measure (e.g. materials do not meet specifications), allowing the allocation of escalation factor controls (e.g. QA during manufacturing) which are in place to prevent the escalation factors having an impact on the preventative or mitigating safety measures.

The strengths of the Bowtie methodology as a tool suitable for enhancing the usability, accessibility and fit-for-purpose nature of the GDA PCSR are briefly explained below.

3.1. Enhancing visibility, understanding and accessibility of GDA PCSR

'A picture paints a thousand words'. The Bowtie diagram is a visual tool which provides a readily understandable visualization of the entire accident scenario (hazard, its causes and consequences, and the prevention and mitigation safety measures in place to control it) in a manner that is easily understood at levels of the organization, from senior management to front-line staff (engineers, operators, maintenance personnel, supervisors). It is therefore a highly effective aid for Safety Report's key end users in their understanding of the plant's hazards and how to control them. (see. Fig.1).

Additional information can be specified for each of the safety measures in the Bowtie. It is possible to reflect the rated effectiveness of the safety measures by assigning a specific colour to the safety measure (e.g. white – good effectiveness, grey – medium effectiveness and black – poor effectiveness). In the Bowtie in Fig. 1, for instance, the existence of a Regulatory Observation (RO) [5] on specific measures can be regarded as reducing the effectiveness of these safety measures at the GDA stage and therefore requiring a specific action, such as Resolution Plan (RP) [5] in order to improve it (see colour legend in the Bowtie page).

Other information can be incorporated that clarifies the design intent and the criticality of a specific safety measure, such as Safety Category, Class, Codes and Standards (see black flag next to text in the safety measure box) or Safety Functional Claims (see grey flag next to text in the safety measure box), or safety measure's main support systems (see white flag next to the text in the safety measure box).

3.2. Linking PCSR with Safety Management Prospectus (SMP)

The Safety Report is regarded as the most important way to demonstrate that the safety is being properly managed and that management controls are appropriate and sufficient, i.e. it must 'tell the story' of how the organization manages nuclear safety risks [7,8,9]. In practice, it is possible to establish strong links between the SMP, which is regarded by ONR as a 'top tier safety case' for nuclear safety management and the Safety Report via Bowtie diagrams.

The hand right side part of the Bowtie diagram in Fig. 1 shows the existing links between the engineering safety measures that prevent and mitigate accidents (e.g. SSLC, RCIC, ADS and RHR) with those safety critical examination, maintenance, inspection and testing (EMIT) activities that are carried out by competent people from the Responsible Party and/or the Prospective Licensee to ensure that the required safety and reliability will be achieved through the plant lifecycle (see. box with allocated activities during design, construction, commissioning and operation). By auditing the conduct of these activities, it is possible to ensure that effective safety measures will be in place at all times and throughout all plant phases [7, 8, 9].

3.3. Developing Licensee's Design Authority capability

Before a site-specific application for a new reactor can be made, the PL will need to begin establishing its management system, including organisational structure and resources, and there will need to be considerable knowledge transfer about the design. Joint working arrangements are usually established between the RP's Suitably Qualified and Experienced Persons (SQEP) responsible designers and PL's technical topic leads, and RP's Safety Case leads with PL's PCSR Chapter leads, bringing about significant benefits in the production of a GDA Safety Case that can be well understood by the PL following completion of the GDA PCSR. Hitachi-GE and Horizon Nuclear Power Joint Safety Case Office (JSCO) is an example of such arrangements [4].

The ONR will focus on the development of technical capability in the various engineering and design authority disciplines and its approach to the development of site-specific safety case and to ensure the necessary transfer of knowledge from the RP. The bow tie in Fig. 1 shows the allocated Licensee's DA responsibilities through PL's technical Topic Leads (see second line in the safety measures box) for each of the safety measures, that are expected to retain the capability to understand the totality of the design and nuclear safety case in the context of each stage of the full plant lifecycle and whose competence needs to be developed.

3.4. Safety Case production and management process

The Bowtie methodology is a risk management tool which forces an "active thinking" to answer the following questions: a) what are the hazards, b) what happens when the control is lost c) how can control be lost? d) what are the potential outcomes? e) how do we keep control? f) how do we minimise the effect? g) how might control fail? h) how do we make sure controls do not fail? and i) what else can we do?

By conducting the Bowtie workshop sessions with the active involvement of the SQEP responsible designer and applying an opposite mind-set, the Bowtie methodology can be a perfect complement to the Claims-Argument-Evidence structured approach (CAE approach) whose purpose is to build a case that "the reactor is safe". This will allow the PL to combat "confirmation bias and compliance-only exercises" [10].

As regards the management of the Safety Case during the subsequent phases of the plant's lifecycle, it is possible with the Bowtie methodology to assess the actual performance of safety measures based on actual data by incorporating the information provided by a wide range of different data sources such as incidents, audits and maintenance systems, thus making the Safety Report a true "living document".

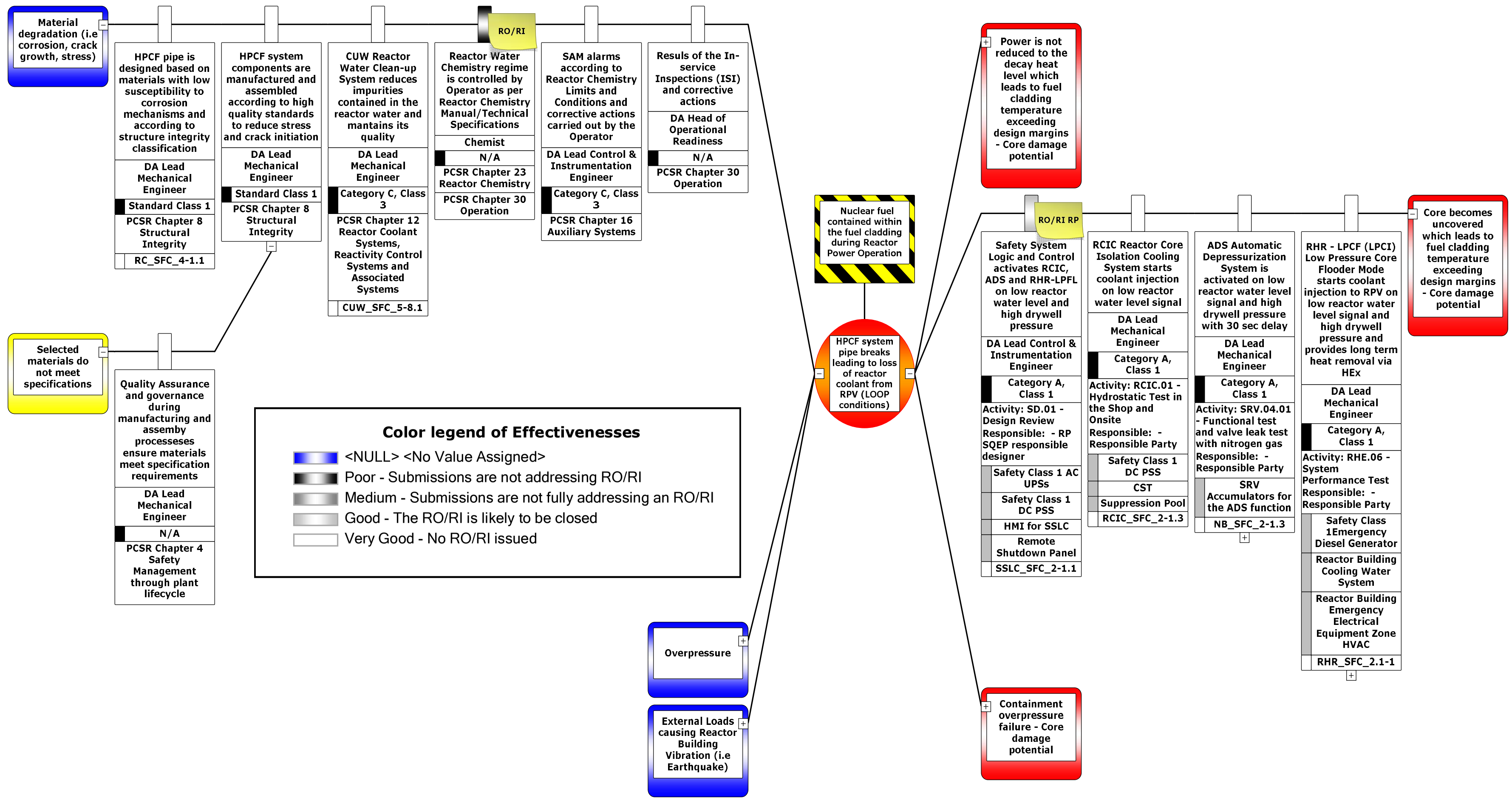

FIG. 1 Sample Bowtie diagram for HPCF pipe break LOCA for a generic ABWR.

(Note: This Bowtie does not represent any specific ABWR Reactor design; however, some information from public domains has been used randomly to make it understandable to the reader).

4. CONCLUSION

The early involvement and integration of the PL into the GDA PCSR production process can take many forms depending on a wide range of circumstances. Whatever the case, the early involvement of the PL in the GDA process can be considered as a capacity building process, during which the PL develops important capabilities necessary for it to become a licensable organization during the site licensing phase. It is therefore critical that the capability of producing a usable and fit-for-purpose Safety Case in a form that is practical and accessible to all those who need to use it, is dealt with upfront, in conjunction with those of the Design Authority/Intelligent Customer.

Both RP and PL can obtain significant benefits from the incorporation of the Bowtie methodology within the GDA PCSR development strategy due to the specific strengths this methodology engenders for the production of usable and fit-for-purpose Safety Reports. Particularly, the conduct of Bowtie workshops (i.e. for bounding faults), facilitated by the PL with the involvement of its own engineering staff and SQEP from the RP, can make a significant contribution to the creation of a Licensee's Design Authority capability which is essential in order to obtain a proper understanding of the design and the nuclear safety case. This will also "encourage people to think as actively as they can to reduce risks".

There is another important feature of the Bowtie methodology that due to the reduced space available for the paper was not addressed. This is the possibility of building Summary of Operational Boundaries (SOOB) Matrices. These are records of operations providing guidance for supervisors and managers in deciding whether to continue certain activities or stop operations (permitted or prohibited operations). Such decisions depend on the actual effectiveness of some safety measures and when to apply compensatory controls in case of reduced effectiveness of any SSC important to safety. This feature is of great help to ensure compliance with limiting conditions for safe operation contained in the Safety Report.

Industries such as aviation have made considerable progress in implementing the Bowtie methodology for the management of aviation risks. The nuclear industry can also benefit from incorporating this methodology within the Nuclear Safety Reports (for the analysis of Design Basis Faults and Beyond Design Basis Faults/Severe Accidents). This will confer an additional value to the Safety Report as an effective risk management tool, making the Safety Report a truly usable product that 'does not sit on the shelf'.

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LIMITED COMPARISON OF EVOLUTIONARY POWER REACTOR PROBABILISTIC SAFETY ASSESSMENTS

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Abstract

The paper presents the insights from a limited PSA comparison on four EPR designs: Olkiluoto 3 in Finland, Flamanville 3 in France, UK EPR, and U.S. EPR. The work was done within the Multinational Design Evaluation Programme (MDEP) design specific working group on the EPR. MDEP was established in 2006 as a multinational initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities. The aim of the PSA comparison was to examine the modeling approaches and outcome of EPR PSAs, and to find the rationale for possible differences. The comparison covered different types of initiators challenging a broad scope of safety functions. The EPR designs chosen for comparison represents various design and licensing stages, as well as levels of detail, which gives the main rationale for the identified differences. The paper also highlights the differences in modeling assumptions, applied reliability data, design solutions, and operational aspects. The insights and lessons learned from the comparison have been used to facilitate the regulatory reviews and assessment of EPR designs and to enhance the quality of EPR PSA models and documentation.

1. INTRODUCTION

The EPR is an Evolutionary Pressurized Water Reactor (a.k.a. European Pressurized Water Reactor), whose design takes benefit from operating experience especially in France and Germany. Reliable prevention and mitigation of severe accidents have been the main goals in the development of EPR. Probabilistic Safety Assessment (PSA) was initiated from the beginning of the conceptual design stage. PSA has been utilized for design optimization with respect to safety and availability [1].

The following organizations were responsible for the EPR PSA comparison: Radiation and Nuclear Safety Authority of Finland (STUK), Institute of Radiological Protection and Nuclear Safety (IRSN) of France, Office for Nuclear Regulation (ONR) of the United Kingdom, United States Nuclear Regulatory Commission (USNRC) and National Nuclear Safety Authority (NNSA) of China. The work was carried out within the Multinational Design Evaluation Program (MDEP) design specific working group on the Evolutionary Power Reactor (EPRWG). MDEP was established in 2006 as a multinational initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities. The Organization for Economic Co-Operation and Development (OECD) Nuclear Energy Agency (NEA) facilitates MDEP's activities by acting as technical secretariat for the program.

The PSA comparison was conducted on the following EPR designs: Olkiluoto 3 Nuclear Power Plant (NPP) in Finland (OL3), Flamanville 3 NPP in France (FA3), UK EPR design, U.S. EPR design, and partly also Taishan NPP (China) respectively.

The aim of the comparison was to examine the modeling approaches and outcome of EPR PSAs, and to find the rationale for possible differences, in order to provide support for safety evaluations and PSA reviews in MDEP member countries.

2. DEVELOPMENT OF EPR PSA

The EPR PSA development was performed in parallel with the early phases of EPR design work. The first Level 1 PSA for internal initiating events was completed at the end of the basic EPR design in 1999. EPR PSAs for OL3 and FA3 NPPs were developed on the basis of the first Level 1 internal events model and documentation. OL3 construction license PSA (2004) has been updated several times in the course of the detailed design process more or less independently from other EPR PSAs. OL3 PSA (2004) was used in the development of U.S. EPR PSA for Design Certification (DC) process in 2007. PSA for UK EPR Generic Design Assessment (GDA) process was at least partially based on the three aforementioned PSAs: OL3 (2004), FA3 (2006) and U.S. EPR (2007). Although EPR PSA developers have been exchanging PSA information and findings, each EPR PSA has been extended and updated in accordance with its own project specific requirements while the licensing and/or the detailed design processes have progressed.

Some PSAs are more or less so called full scope PSAs in terms of the coverage of initiating events i.e. internal IEs, and internal and external hazards are included in the analyses. The others include somewhat limited analysis of hazards, as summarized in Table 1.

TABLE 1. SCOPE OF PSA MODELS

PSA item	FA3 (FSAR)	UK EPR (GDA)	OL3 (Oper.License)	U.S. EPR (DC)
Level 1	x	x	x	x
Level 2	x	x	x	x
Level 3	simplified	simplified (full scope by HPC)	-	Full scope in support of Environmental Report
Internal Events	x	x	x	x
Internal Hazards	x	at power	x	x
External Hazards	x	limited	x	x
Seismic	simplified	PSA based SMA**	Seismic PSA	PSA based SMA
Fuel Pool Accident	x	x	x	-
LCHF*	x	x	-	-

* Scenarios with Low consequence and high frequency (no core damage)

** Seismic Margin Assessment

3. EPR PSA COMPARISON

3.1. Scope of comparison

Detailed comparison of large and comprehensive PSAs is a very labor intensive job. In order to enable effective use of limited resources and to keep the focus on most important aspects in PSAs, the scope of comparison was limited to the following four initiating events (IEs): medium loss-of-coolant accident (LOCA), loss of offsite power (LOOP), steam generator tube ruptures (SGTR), and loss of cooling chain (LOCC). The selection of IEs was based on two criteria: 1) at least one IE from all main initiator groups i.e. transients, LOCAs, primary-secondary leakages and common cause initiators (CCI), and 2) IEs challenging a broad scope of safety functions. The main focus in comparison was on the IE definition, modeling of accident sequences (i.e., timing, safety functions, success criteria, automatic and manual actions, etc.), minimal cut sets, importance measures, and quantitative results.

The analysis of internal initiating events forms the basis of plant specific PSA. EPR designs included in the comparison represent various stages of the design process, licensing process, as well as level of modeling detail. Therefore internal initiating events PSAs were selected for EPR PSA comparison effort. The source of the background information on the EPR PSAs is summarized in Table 2.

TABLE 2. EPR PSA MODELS AND DOCUMENTATION

EPR design	PSA information source
FA3	Final Safety Analysis Report (FSAR) (2010)*
UK EPR	GDA step 4 (2011) [2], GDA PCSR (2011) [3]
OL3	Pre-Operating License Application (pre-OLA, v104, 2010)
U.S. EPR	Design Certification (DC) rev. 5 + PSA (2013)

* Updated in 2015 (next update in 2017)

3.2. Main Results of EPR PSAs

Table 3 presents the results of four different EPR designs' internal initiating events PSAs for power operating modes. The total core damage frequencies (CDFs) are fairly similar but the risk profiles are not identical. Based on the experience from previous PSA comparisons performed e.g. in France and Finland, it was evident that the comparison should not focus on only those IEs, which CDF differs the most. Even with similar CDFs, significant difference may be identified related to IE frequencies, most important cut sets, modeling details, most important basic events, assumptions etc.

TABLE 3. EPR PSA INTERNAL INITIATING EVENTS CDF (1/YEAR)

IE	DESCRIPTION	FA3	UK EPR[A]	OL3	U.S. EPR
LOOP	Loss of Offsite Power	1,40E-07	2,97E-07	1,33E-07	1,23E-07
LOCA	Loss of primary coolant accident	5,70E-08	1,06E-07	7,08E-08	4,48E-08
MLOCA	*Medium LOCA*	*(3,6E-08)*	*(9,2E-09)*	*(3,1E-08)*	*(9,1E-10)*
V-LOCA	LOCA leading to containment bypasses	6,50E-10	3,70E-09	1,50E-08	-
Prim-Tr	Primary circuit transients	2,00E-08	5,25E-08	1,07E-08	-
Sec-Tr	Secondary circuit transients	4,60E-09	1,63E-08	8,37E-08	1,37E-08
Sec. Br.	Secondary circuit breaks	1,80E-08	1,3E-08	8,88E-09	-
SGTR	Steam Generator Tube rupture(s)	1,10E-08	1,02E-08	2,21E-08	2,63E-08
LOCC	Loss of cooling chain or heat sink	8,80E-08	9,46E-08	1,94E-08	3,61E-08
Other	Other IEs	1,38E-07	2,58E-08	1,21E-07	4,91E-08
	TOTAL	4,8E-07	6,2E-07	4,8E-07	2,9E-07

A PSA for a UK EPR™ at Hinkley Point C [4]

4. EPR DESIGN DIFFERENCES

The MDEP EPRWG PSA technical expert subgroup has held joint meetings with EPR vendors exchanging information related to regulatory review findings, modeling details, design differences and potential new design changes. The aim was to find rationale for differences in EPR PSAs, whether their origin is in design, PSA modeling or data.

Reasons for differences in design solutions and modeling of EPR PSAs are among others:

- Progress of plant design (GDA, DCD, OLA, FSAR…)
- Project specific customer requirements;
- Project specific regulations;
- Project specific rules and standards;
- Project/customer database;
- Project specific site characteristics;

— Project specific modeling assumptions/approaches of the PSA teams.

Examples of known differences, which are implemented due to regulations, site, operator, industry or project timing (not all of these are directly related to the PSA comparison exercise):

— All EPRs share the same objective to minimize the release to the environment in case of SGTR. Different SGTR management strategies exist. All EPR have faulty steam generator automatically isolated at the end of partial cooldown. If not, all EPR have manual isolation done around 60 minutes post fault. Specifically for OL3, the automatic signal can be initiated by "activity measurements".
— There are some differences in system design, e.g. HVAC systems, extra borating system, fuel pool cooling system, EDG size and cooling, fire zoning design, electrical supply for main steam relief train, and some of the I&C systems.
— Full rupture (2A LOCA) of reactor coolant systems is not studied as design basis event in all EPR designs.
— There are differences in reactor coolant system insulation material (mineral vs. glass wool), but this has no impact on the PSA.
— There are design differences related to severe accident management, for example:
  - Fulfilment of single failure criterion in severe accident management systems is required in OL3.
  - Diversity between severe accident and design basis accident equipment is required in some EPR designs.
  - Redundancy in severe accident depressurization is required in some EPR designs.
  - Severe accident containment filtered venting is required in some EPR designs.

The risk significance of identified design differences has not fully been evaluated. However, the overall insight based on the limited EPR PSA comparison suggests that none of these differences plays very important role in terms of risk.

5. SUMMARY AND CONCLUSIONS

The main comparison work was performed a few years ago and therefore the most recent developments in the EPR design and PSA models are not reflected in the paper.

The first overall insight of the PSA comparison is a global agreement on the most important results (total CDF and main contributions) leading to a reasonable confidence in the PSAs. However, the more detailed comparison identified several differences which could generally be explained.

One of the most important reasons for the identified differences is due to the fact that compared EPR PSAs represent various stages of the design process, licensing process, as well as level of modeling detail. Some PSAs are so called full scope PSAs in terms of the coverage of operating modes and initiating events, i.e. internal IEs, and internal and external hazards are included in the analyses. The others include somewhat limited analyses of hazards.

Comparison of the numerical results of different EPR design PSAs is not straightforward. Firstly, each PSA represents various phases of licensing and detailed design processes. Secondly, there are differences in EPR designs, which affect the risk. Thirdly, studying the numerical results alone does not reveal the definitions and assumptions related to the modeling of IE groups and the accident progression.

The following issues and insights were identified:

— Modeling of digital I&C: the differences in the details and assumptions related to the modeling of I&C systems explain some of the identified differences. The different I&C architecture play also an important role in the difference. In addition, the detailed design of the OL3 I&C system was under development and some changes were foreseen.
— Modeling of ventilations: modeling of HVAC systems is not at the same level in the different PSAs, although the contribution of HVAC can be significantly different due to site characteristics (tropicalization) and lead to different design choices (e.g. diversification of the safeguard buildings electrical divisions ventilation for OL3).
— RCP seal LOCA management: at the time of the comparison, the assumptions and the level of detail in the seal LOCA modeling appear as rather different and leads to differences in the results.

- Pipe ruptures frequencies: regarding the data used for the LOCA frequencies, there is a significant difference between the OL3/FA3 PSAs and the UK/US EPR PSAs. It is not clear which data is the most representative. However, the differences in design should not affect the initiating event data (the basis used to estimate the pipe rupture frequencies are not detailed enough to differentiate between minor design differences). The choice of applicable data may be driven by the licensee, the vendor or in some cases by the regulatory body.
- Success criteria and supporting (thermal-hydraulic) studies: for example the Steam Generator (SG) success criteria in case of MLOCA or the feed and bleed success criteria are different and can explain different results.
- Reliability data: certain component data are rather different, although the effect on the results remains limited.
- Human Reliability analysis (HRA): the human errors probabilities are in some cases quite different, due to different assumptions in modeling and different support calculations concerning the time available for the action.
- Common Cause Failures (CCF): the assumptions relating to CCF are different in some cases (notably for I&C and for batteries) and have a significant contribution to the results.

The outcomes and lessons learned from the EPR PSA comparison have been used to facilitate the regulatory reviews and assessment work of various EPR designs and to enhance the scope, level of detail, and quality of EPR PSA models and documentation.

The comparison made it possible to identify differences with a potential impact on the results, but generally the information provided is not sufficient for considering that an approach or another is the best practice. So a general recommendation is to review the issues identified by the comparison with a special attention.

In particular it is recommended to review (and improve if possible):

- Modeling of I&C (the treatment and assumptions concerning software failures and spurious actions of I&C systems as well as their impact on results and most important cut sets is to be reviewed. Comprehensive fault analyses are needed for a realistic modeling of I&C systems).
- Modeling of HVAC systems.
- Management of RCP seals LOCA.
- Data relating to LOCA frequencies, component failure rates, human errors.
- Supporting calculations (thermal-hydraulics).
- Identification of CCF groups. It has to be noted that, even if the assumption is the same for all PSAs (e.g. EDG diesel generators/ SBO diesel generators), the importance of this assumption indicates that its justification is very important.

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RISK-INFORMED DESIGN FOR CAP1400

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Abstract

The CAP1400 was developed as an advanced Generation III reactor by Shanghai Nuclear Engineering Research & Design Institute (SNERDI), incorporating with State Nuclear Electric Power Planning Design & Research Institute (SNPDRI), and the demonstration project will be constructed in Shandong, China. Based on the passive safety concept, CAP1400 was designed integrated with simplified safety systems and optimized plant layout, promoting its safety and economic features, in accordance with the lessons learned from Fukushima accident. The credit is taken for CAP1400 to withstand internal and external hazards, allowing a period of 72 hours without operator interactions after accidents. The design of CAP1400 benefits from risk-informed design changes addressed by Probabilistic Safety Assessment (PSA) results. A full scope, all-modes PSA was developed for CAP1400, including internal events PSA, internal fire PSA, internal flooding PSA, seismic PSA, spent fuel pool PSA, level 2 and level 3 PSA, etc. The paper will introduce the risk-informed design method and its application for CAP1400. Parts of design change derived from PSA insights are also presented in this paper, e.g., addition of very early warning fire detection system, improvement of capability of passive core cooling system, design change of in-vessel retention of molten core debris, etc.

1. INTRODUCTION

CAP1400 is a two-loop pressurized water reactor plant, based upon passive safety concept of AP1000, with re-design and optimization of reactor coolant system, safeguard systems, main nuclear auxiliary systems and nuclear building arrangement, etc. The CAP1400 is designed to meet the applicable requirements and safety goals defined for advanced light water pressurized water reactor with passive safety features. Compared with AP1000, the safety and economic features of CAP1400 are improved. The main technical specifications of CAP1400 are shown in Table 1.

TABLE 1. TECHNICAL SPECIFICATIONS OF CAP1400

Name	Parameters
Design lifetime	60 years
Reactor power	4040 MWt
Electrical power	~1500 MWe
Reactor operating pressure	15.5 MPa
Hot leg temperature	323.7 °C
Number of SGs	2
Number of reactor coolant pumps	4
Volume of containment	>75000 m^3

The concept of passive safety of CAP1400 includes passive core cooling system (Core Makeup Tanks, Accumulators, In Containment Refueling Water Storage Tank and Containment Recirculation, etc.), passive containment cooling system, automatic depressurization system, etc., which allows a period of 72 hours without operator interactions after accidents. The CAP1400 is designed in accordance with the lessons learned from Fukushima accident. The credit is taken for CAP1400 to withstand internal and external hazards, including internal flooding, internal fire, external flooding, high wind, etc., and the safe shutdown earthquake is 0.3g with the High Confidence Low Probability Failure (HCLPF) value of 0.5g. For CAP1400, the risk-informed design method is applied, and during the phases of concept design and detailed design, the Probabilistic Safety Assessment (PSA) method was implemented. Based on analysis model, and informed by the risk, specific analysis is performed from different aspects, e.g., initiating events, accident sequences, minimal cutsets and risk importance, etc., to figure out plant design vulnerabilities and to perform design improvement and optimization continuously. The ultimate purpose is to realize relatively lower risk level and higher safety level.

2. RISK-INFORMED DESIGN

2.1. Overview

Since the first introduction of PSA for Nuclear Power Plant (NPP) safety analysis, it has been proved its positive role for proposing effective approach to reduce severe accident risk. As the PSA technology develops and matures, the application in NPPs becomes more extensive, and introduction of risk-informed method and design improvement are being implemented progressively. The following characteristics are included:

- Not related to reactor types, and generally applied
- Structural technique method, integrating probabilistic and deterministic analysis
- In accordance with basic safety principles, including defense-in-depth and safety margin
- Consistency, stability, predictability and flexibility

The main process of risk-informed design of CAP1400 is presented in Fig. 1. During the risk-informed design, the preliminary PSA model is developed based on top level safety goals (including Core Damage Frequency (CDF) and Large Release Frequency (LRF)) of CAP1400, then the preliminary reliability allocation target of systematic level is performed. The detailed system design is implemented, and reliability prediction is developed based on system design. On the basis of the results of reliability prediction, the system reliability allocation target is adjusted or the system design is improved. The process of risk-informed design improvements of CAP1400 is shown in Fig. 2. In the case without detailed information and the overall plant model cannot be developed, the risk-informed design is performed mainly for systems.

2.2. Risk-informed design of CAP1400

Risk-informed plant design is an iterative process, and the risk-informed analysis of CAP1400 includes three parts: developing detailed PSA model, system reliability design and design improvement.

Developing detailed CAP1400 PSA model

To perform risk-informed design, the basis and key point is to develop detailed PSA model.

A full scope PSA is developed for CAP1400, including Level 1, 2 and 3 analysis for internal events at power condition, Level 1 PSA at low power and shutdown conditions, spent fuel pool PSA, and external events (internal flooding, internal fire and seismic, etc.) PSA. The PSA of CAP1400 is developed in compliance with NB/T 20037 Standard [1-2] and ASME PSA Standard [3], for the purpose of evaluation of CDF and LRF of CAP1400 to obtain specific PSA insights.

The major tasks of Level 1 PSA include initiating events analysis, event tree analysis, system fault tree analysis, human reliability analysis, dependency analysis, data analysis, sequence quantification, importance and sensitivity analysis, uncertainty analysis， as well as result analysis and insights. The Level 2 PSA evaluates severe accident phenomena and fission product source terms, modelling of containment event trees and associated success criteria, hydrogen combustion and mixing analysis. Offsite dose evaluation analysis is performed in Level 3 PSA.

The initiating event analysis takes into account of differences between CAP1400 and other pressurized water reactors, e.g., safety injection line break, core makeup tank line break and passive residual heat removal heat exchanger tube rupture. The accident of loss of reactor coolant pump seal is eliminated as canned pumps are applied for CAP1400. Other plant specific design features are considered in analysis, e.g., the initiating event frequency of Steam Generator Tube Rupture (SGTR) is 2.50E-03/(reactor*year) for SG tubes manufactured by Alloy 690.

The event trees of CAP1400 are modelled using the small event tree/large fault tree method. For CAP1400, the late containment failure, a new sequence consequence, is also considered.

The CAP1400 PSA is characterized that the passive safety related systems are modelled in fault tree analysis, developed following the same process and method as other active system fault tree analysis.

The Technique for Human Error Rate Prediction (THERP) is used to evaluate pre-accident human actions and post-accident human actions are quantified with Human Cognitive Reliability/Operator Reliability Experiment (HCR/ORE)+Cause-Based Decision Tree Method (CBDTM)+THERP method.

The database of component reliability and unavailability due to test &maintenance of CAP1400 contains 71 types of components, including pumps, valves, and switches, etc., providing complete information of failure modes, failure probability/rate, uncertainty distribution, and distribution parameters, etc. The NUREG/CR-6928 [4] is referenced in CAP1400 PSA to develop general component reliability database, and detailed analysis are performed for CAP1400 specific passive components.

Four types of dependencies are evaluated in CAP1400 PSA, including functional dependency, physical dependency, human failure event dependency and component failure dependency. The Multiple Greek Letter (MGL) method is applied in CAP1400 PSA to assess Common Cause Failures (CCFs), and the parameters of CCF are defined referencing associated database of American plants.

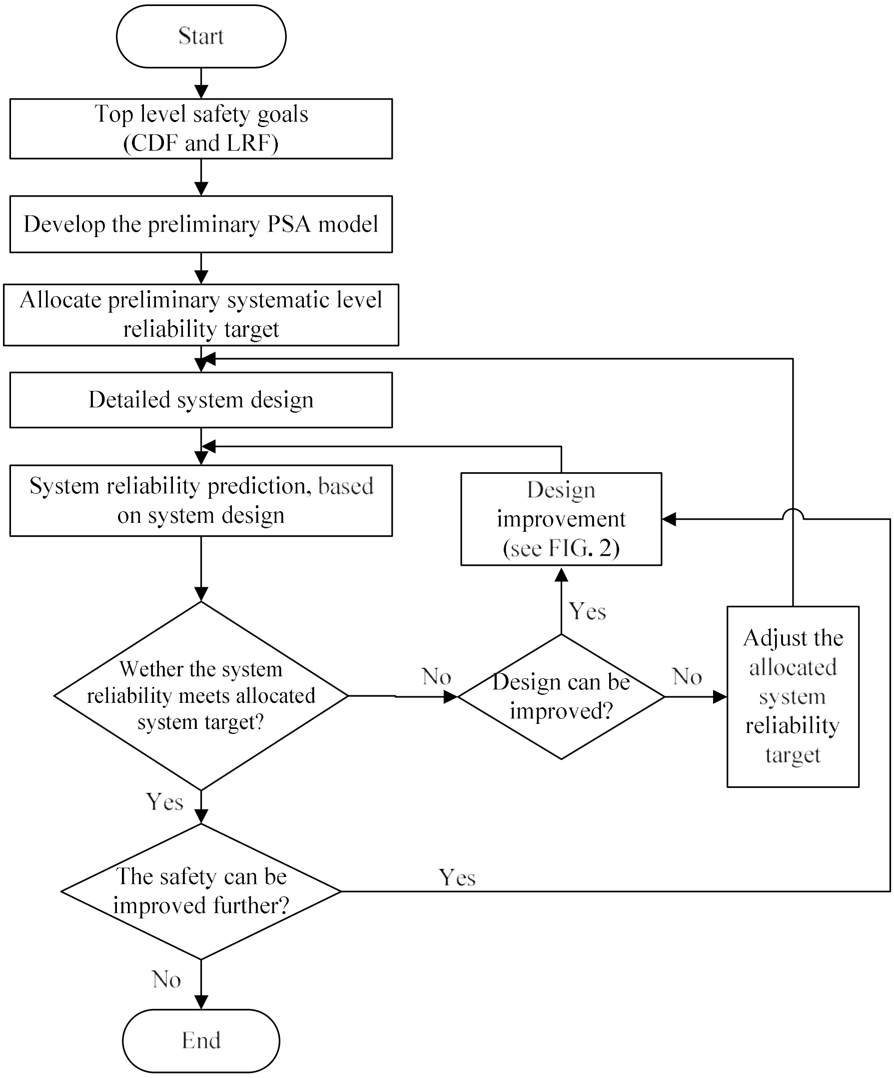

FIG. 1. Risk-informed design process of CAP1400.

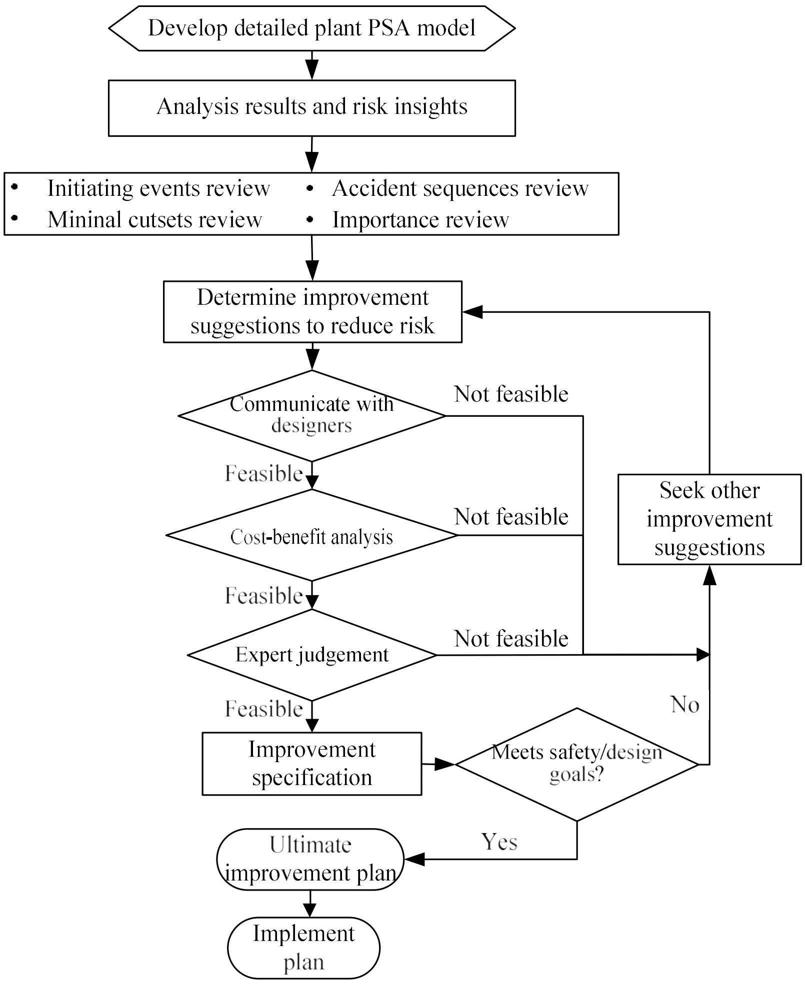

FIG. 2. Risk-informed design improvement process of CAP1400.

System reliability design of CAP1400

System reliability design of CAP1400 is comprised by reliability target allocation, system reliability prediction and adjustment of allocated reliability target.

Reliability target allocation

During the risk-informed design of CAP1400, the preliminary PSA model is developed based on the top level safety goals (including CDF and LRF), referencing system reliability of similar plants and specific design of CAP1400, then the preliminary systematic level reliability target is allocated.

System reliability prediction

As system design processing, the PSA model is developed based on detailed system design to predict system reliability.

Adjustment of allocated reliability target

Where the predicted system reliability could not meet the allocated system target, the system design improvement should be performed firstly. For the case that the design couldn’t be improved or the target still couldn’t be met after improved, the allocated system reliability target should be adjusted.

Design improvements of CAP1400

The design improvements of CAP1400 are performed from review of four aspects including initiating events, accident sequences, minimal cutsets and risk importance, and improvement plan and suggestions to reduce risk are proposed. After communication with designers, the improvement feasibility is analysed and cost-benefit analysis is performed. Finally the feasibility of improvement suggestions is determined through expert judgement, and the improvement plan is specified and implemented.

c) Determination of improvement suggestions to reduce risk

The followings are performed to determine improvement suggestions to reduce risk:

Dominant initiating events with significant contribution to CDF/LRF

Identify the vulnerabilities of plant safety function design, which could be used to review system design reasonability and to resolve vulnerabilities to improve safety.

Dominant accident sequences with significant contribution to CDF/LRF

Identify vulnerabilities of plant system configuration and human response, which could be used to review system configuration reasonability, to improve corresponding emergency operation procedures and to guide operator training as well.

Dominant minimal cutsets with significant contribution to CDF/LRF

Identify risk-important minimal cutsets, and propose improvement measures to avoid combination of these failure events or reduce its occurrence frequency.

Risk importance

Risk importance of components

Identify influence importance of component to CDF/LRF, and more resources can be distributed and more attention should be paid to component with high importance.

Risk importance of human events

Identify influence importance of human intervention to CDF/LRF, and this can be used to resolve vulnerabilities of accident mitigation procedures or maintenance procedures, improving effectiveness of operator and maintenance personnel training and optimizing human-machine interface design.

Based on four aspects above, the major factors influencing risk level can be found, and the analysts perform further detailed analysis to propose improvement suggestions.

Final improvement plan

For proposed improvement suggestions, detailed and multiple-level feasibility argument needs to be performed to obtain final improvement plan.

Communicate with designers

Communicate with system designers, who will give feasibility of improvement suggestions from different dimensions including component, system, arrangement, etc.

Cost-benefit analysis

Although the risk reduction has no limit, the cost of improvement certainly has some limitations. For improvements to reduce risk, cost-benefit analysis needs to be performed.

Expert judgment

Experts from different areas review the overall analysis process, the final improvement plan and make decisions based on their experience and calculation results.

3. EXAMPLES OF CAP1400 RISK-INFORMED DESIGN

3.1. Examples of system reliability allocation of CAP1400

Based on the system characteristics of CAP1400 and experience of AP1000, some allocated system reliability targets are shown in Fig. 3, and the results of system reliability prediction meet top level safety goals.

Where:

CMT: Core makeup tank

IRWST: In-containment refueling water storage tank

ADS: Automatic depressurization system

ACC: Accumulator tank

CIS: Containment isolation system

PRHR: Passive residual heat removal system

RCP: Reactor coolant pump

CCS: Component cooling water system

VWS: Central chilled water system

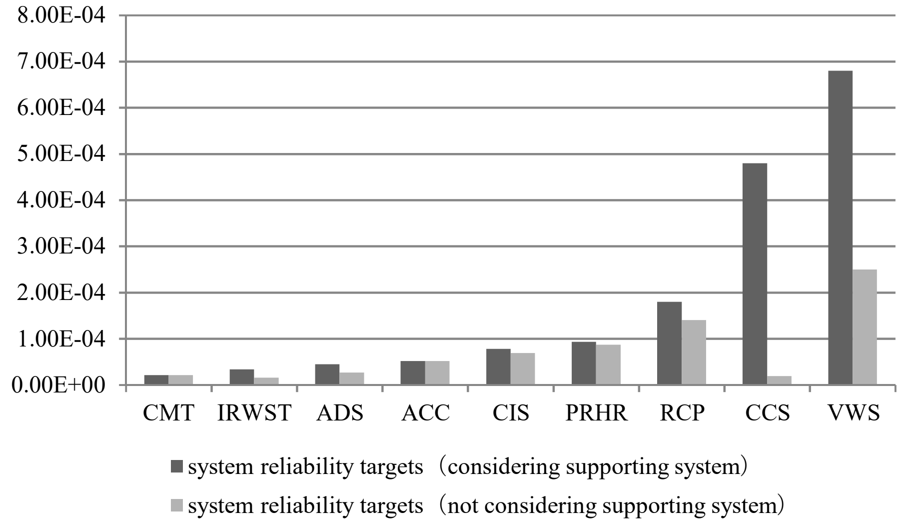

FIG. 3. Some system reliability allocation targets of CAP1400.

3.2. Examples of CAP1400 design improvements

To improve the safety of CAP1400 plant further, the detailed design of CAP1400 is reviewed, thereby the design improvements are proposed as following.

3.2.1. Addition of Very Early Warning Fire Detection System

d) Overview

The contribution of fire-induced risk to overall risk of CAP1400 is significant, and potential fire risk needs to be reduced by considering design improvement.

Very Early Warning Fire Detection System (VEWFDS) is a system used to detect low-energy fire before it threats to facilities, which could detect fire sign from 1hour to even several days earlier and provide personnel plentiful time to intervene. The system is extensively applied for fire protection of telecommunication facilities and its effectiveness of detecting initial fire arising from electrical cabinets, electronic cabinets and low-voltage circuits (e.g., cable tray paths, junction boxes, terminal cabinets, etc.) is totally proved.

As most of the low-voltage electrical component fire would have a long term initiating process, during which no open flame or smoke occurs, the VEWFDS could detect this initiating process, therefore the fire risk could be reduced.

Determination of improvement suggestions to reduce risk

The addition of VEWFDS is suggested in the area 1100 AF 11300B and 1100 AF 11500, based on the analysis results of Internal Fire PSA. Certain detectors are suggested to be installed in vicinity of safety related cable (A/B/C/D train cables) trays located in 11300Band 11500, according to the detecting range of VEWFDS, to ensure the detecting effectiveness for these safety-related cables and to avoid occurrence of fires affecting plant safety functions as far as possible.

For some specific important fire sources, it is suggested to install certain detectors inside or in the vicinity of these sources.

Final improvement plan

The high sensitive VEWFDS, which were already applied in some nuclear power plants, could detect the fire location early.

After communication with electrical designers, it is feasible to apply this modification, and corresponding electrical, instrumentation and control arrangements and plant fire protection procedures need to be modified. The fire-induced CDF during power condition could be reduced by more than 20%. Moreover, the consideration of areas where VEWFDS installed and its reliability is conservative, if the failure rate of VEWFDS would be reduced from 0.5 to 0.1, the fire-induced CDF would be reduced by more than 40%.

Meanwhile, the cost to install VEWFDS is relatively low with a high benefit, and the fire-induced risk could be reduced significantly.

After reviewed by experts, this modification plan is performed for CAP1400.

3.2.2. *Other improvements*

Other improvements include:

e) Increase valve blowdown rate of ADS-4, IRWST water level and decrease resistance of safety injection lines, to promote the performance of passive core cooling system, optimizing ADS success criterion from 3 in 4 (3 out of 4 valves of 4th stage ADS should be opened) to 2 in 4 (1 out of 2 valves in each loop of 4th stage ADS should be opened).

Increase the diameter of cavity flooding lines, to improve cavity flooding rate and provide longer time for operators to execute cavity flooding operation.

The junction boxes of valves of passive core cooling system that may fail due to spray or flooding are qualified with the capability against spray and flooding to maintain their availability under accident conditions.

The fire evacuation door located between nuclear island buildings and turbine building is modified with quality against flood to prevent that flood propagates from turbine building to nuclear island buildings.

4. CONCLUSION

The risk-informed design is introduced during design phase of CAP1400. From concept design phase to detailed design phase, the risk-informed method is used to analyze factors with significant risk contribution and to support proposing corresponding improvement suggestions. After detailed feasibility analysis, the applicable and feasible suggestions are implemented to optimize plant design. The application of risk-informed concept in CAP1400 design is an attempt and the practice of risk-informed design in the new plant provides good experience for further application.

ACKNOWLEDGEMENTS

Thanks National Science and Technology Major Project of the People’s Republic of China for fund support. Thanks all the designers of Shanghai Nuclear Engineering Research & Design Institute (SNERDI) for their contribution to CAP1400 design. Thanks the PSA team for their analysis and assessment for the risk-informed design of CAP1400. Thanks all the experts for their guidance.

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DESIGN SAFETY PRINCIPLES

CONSIDERATIONS ON NEW DESIGN SAFETY PRINCIPLES

Chairperson

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DIVERSITY ANALYSIS FOR ADVANCED REACTOR DESIGN

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Abstract

Along with redundancy, physical separation and functional independence concepts, diversity shall be considered wherever a safety function requires a high level of reliability in an advanced reactor design. The paper presents the main stages of an innovative approach for a diversity analysis identifying the needs for additional diverse line to back-up main prevention line of safety functions involved in several Defence-in-Depth levels as well as its minimal reliability. The first methodology step consists in considering design extension condition by analysing both the combination of design basis events combined with the most credible common cause failure affecting the main prevention line, and multiple failures affecting safety systems used during normal plant operation. Then, this functional analysis is supplemented with probabilistic insights based on system reliability analysis in order to identify the components candidate for diversity. Finally, a detailed Failure Mode and Effect Analysis of the selected component allows the identification of critical redundant internal parts considering the probability of common cause failure affecting component safety functions. Diversity provisions to eliminate the occurrence or reduce the consequences of such common cause failures are then proposed.

1. DEFINITIONS

Diversity is defined as the presence of two or more engineered Structures, Systems and Components (SSCs) to perform an identified function, where these SSC have different attributes so as to reduce the possibility of Common Cause Failure (CCF): Functional diversity, Equipment diversity, Software or signal diversity, Design diversity, Human diversity. It is important to consider diversity at all stages of the system design and implementation, care must be taken not to unduly increase complexity (in design, architecture or maintenance), which may defeat the overall objective to reduce the overall nuclear risk.

Implementing diversity in the design allows preventing CCF occurrence by eliminating their potential sources or shared failure mechanisms. According to [1], those sources of CCF, or “Coupling Factors”, could be categorized as follows: Environment based (component location…), Operation based (staff, maintenance procedure…), Hardware based (design, parts…), Hardware quality (manufacturing and installation).

2. DIVERSITY ANALYSIS

2.1. Safety Requirements

As stated in [2], the Defence-in-Depth (DiD) concept shall be applied in nuclear power plant (NPP) design to provide several levels of defence that aimed at preventing harmful effects of radiation on people and the environment, and ensuring that adequate measures are taken for protection of people and the environment and for mitigation of consequences in the event that prevention fails. DiD levels shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular safety features for design extension conditions shall be independent to the extent practicable of those used for more frequent accidents. The adequate level of independence between DiD levels is determined on a case-by-case basis.

Application of appropriate diversity allows increasing independence effectiveness of engineered SSCs belonging to different DiD levels by lowering or eliminating CCF occurrence probability in order to achieve the necessary safety function reliability.

2.2. Overall approach

The proposed approach for diversity analysis in the design of an advanced reactor is firstly based on a systematic functional analysis of CCF affecting the main redundant features that perform safety functions, and allows determining where system diversity is required for the most frequent Postulated Initiating Event (PIE) (i.e., PIE with occurrence frequency larger than a specified value) and is driven by probabilistic criteria. The analysis is then completed with the identification of the set of components that need to be actually diverse within the redundant systems and which practical provisions for component diversity should be implemented.

Moreover, a comprehensive probabilistic safety assessment (PSA) should be carried out throughout the design process for a nuclear power plant with iterations with design activities and increased scope and level of details as the design programme progresses. When available, PSA results could be used to establish that a balanced design has been achieved and that to the extent practicable, DiD levels are independent. The goal of the diversity analysis, performed at an early stage of the reactor design phases, is to identify the needs for improving system reliability by diversity with sufficient safety margins for anticipating future detailed PSA development.

2.3. Identification of diversity requirements

2.3.1. Probabilistic criteria

In an early design approach when no details on severe accident mitigation features are known, only the reactor core damage risk could be considered so that diversity requirements apply to features belonging to the third level of DiD as defined in [2] (i.e., DiD level for reactor protection and prevention of core damage). An example of a representative accident sequence leading to core damage is presented on Fig. 1 where the “front line mission” and the “diverse line mission” represents the independent safety features required for core damage prevention.

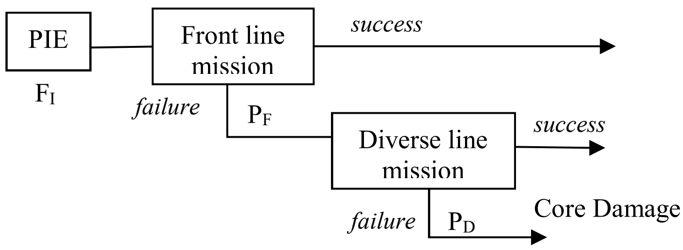

FIG. 1. Representative design extension accident sequence.

The Core Damage Frequency (CDF_s) for this single and simplified accident sequence could be expressed as:

$$CDF_s = F_I \ x \ P_F \ x \ P_D \qquad (1)$$

Where, F_I represents the PIE frequency, P_F the front line mission failure probability and P_D the diverse line mission failure probability (as stated, no dependencies are assumed between the front line and the diverse line).

Considering as an example that the CDF for this accident sequence should be lower than about10^{-8} per reactor.year (/ry), Table 1 gives the claimed value for the diverse line reliability P_D according to the frequent PIE frequency, F_I, (e.g., larger than 10^{-4}/ry), and the front line reliability, P_F.

TABLE 1. RELIABILITY TARGETS FOR DIVERSE LINE WITH CDF < 10^{-8}/RY

PIE frequency (FI)	Front line reliability (PF)		
	10-2	10-3	10-4
10-1/ry	PD < 10-5	PD < 10-4	PD < 10-3
10-2/ry	PD < 10-4	PD < 10-3	PD < 10-2
10-3/ry	PD < 10-3	PD < 10-2	PD < 10-1
5 10-4/ry	PD <2 10-3	PD <2 10-2	PD < 2 10-1
≤ 10-4/ry	No Diverse line required		

On this example, by assuming a front line reliability better than 10^{-3}, one can see that a diverse line is required with a reliability better than 10^{-2} for PIE frequency larger than 10^{-3}/ry.

Depending on the required diverse line failure probability P_D, there should be an adapted solution to meet the safety requirement.

2.3.2. Functional analysis of PIE

For the functional analysis of PIE, a functional breakdown structure starts from the three Fundamental Safety Functions (FSFs) that shall be fulfilled in order to achieve the overall safety objective of protecting people and the environment from harmful effects of ionizing radiation [2]; the three FSFs are: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material.

A systematic structured approach is taken to identify those safety features that are necessary to fulfil the FSFs: They are split into more detailed safety functions and the front lines, including supporting systems, are identified that contribute to these safety functions.

In the next step, and only for frequent PIEs (e.g., PIE frequency > 10^{-4}/ry), each main front line feature is separately assumed to be unavailable due to CCF, if such failure is plausible, then it is analysed and a diverse line is required for providing the same function in a DEC context.

A simplified example of the functional analysis of a PIE for a PWR could be as follows:

- PIE: Small LOCA, reactor at full power;
- Fundamental Safety Function: Decay Heat Removal;
- Safety function: Ensure RCS water inventory by make-up;
- Front line: Medium Head Safety Injection (MHSI) trains;
- Diverse line: Diverse Safety Injection trains (with or without RCS depressurization).

3. PRELIMINARY SYSTEM RELIABILITY ANALYSIS

The next stage consists in carrying out a quantitative diversity analysis in order to select the most important components to analyze in more details within the diverse line.

As an example, a simplified 3-train architecture for safety injection is provided on Fig. 2. Each redundant train is composed of a pump, P, operated by a motor, M, and injecting through a motor-operated valve (MOV), V.

As a result of the previous example of the functional analysis, it could be chosen to diversify the third redundant safety injection train. The issue that should be addressed is to determine which component of the three ones, i.e., pump, motor, or MOV, in this diverse line needs to include actual diverse provisions.

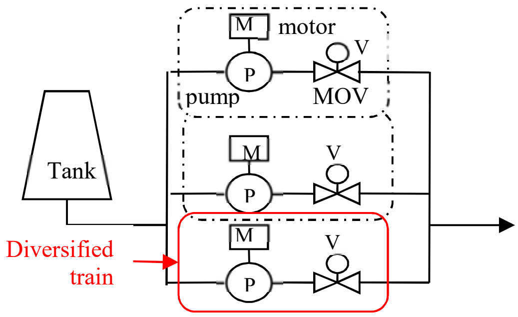

FIG. 2. Simplified 3-train architecture for safety injection.

The quantification of the total system reliability based on the respective reliability component, yields to the conclusion that using MOV or motor diversity would not lead to a significant reliability increase (e.g., between 1% and 3%) compared to the configuration without any diversity. On the other hand, diversifying the pump within the third redundant safety injection train leads to a significant increase in the function reliability (e.g., 86%).

With this quantitative approach a first list of components, candidate for diversity, could be proposed according to their expected reliability level, mainly:

- Active equipment: pumps, emergency supply generators, fans, compressors;
- Instrumentation and Control (I&C) feature: acquisition, processing, actuation, communication.

On a case by case basis with appropriate justification, there could be no diversity requirements for sufficiently reliable or simple equipment such as:

- Passives devices: filters, tanks, exchangers, check valves, manual valves, orifice plates, ducts, pipes…
- Batteries (diversity is possible by aging);
- Electrical motors, motor-operated valves;
- Electrical components: cables, breakers, contactors, transformers…

4. DETAILED COMPONENT RELIABILITY ANALYSIS

4.1. Approach

The approach is based on an industrial practice for identifying the cause of a component failure: the Failure Mode and Effect Analysis (FMEA) [3] applied on the components that have been identified as candidate for further diversity analysis in the previous stage.

The first FMEA step consists in a functional analysis for defining the main component functions, its boundaries and operation profile. The component is then broken-down into internal parts (or sub-systems) consistent with the level of detail considered in applicable operating experience databases, e.g. NUREG/CR-6819 [4] or OCDE/NEA ICDE project [5].

Then independent failure modes of each functional internal part, based on engineer judgment and applicable operating experiences on similar equipment, leads to the identification of the critical internal parts. The criticality level is based on the failure gravity estimation, based on severity level, probability of occurrence and possibility of failure detection

The critical internal parts associated to their failure modes are then further analyzed by identifying the potential CCF that will cause more than one internal parts of redundant components to fail simultaneously. CCFs are characterized according to their coupling factors. As for independent failures, engineer judgment and applicable operating experiences are used for this analysis

Diversity provisions for eliminating those coupling factors are finally proposed, related to the different attributes of diversity: design, quality, manufacturing, maintenance.

4.2. Example of application

As an example, safety pumps from EDF power plants have been considered in order to investigate the associated diversity provision for a new reactor design. The critical internal parts identified were: coupling, bearings, lube oil, mechanical seals and bolting. Diversity provisions have been characterised to eliminate CCF affecting those parts used in redundant pumps and specific diversity requirements would be specified by EDF in the equipment contracts.

As insights of this example of application, it has been shown that the proposed approach is:

- Feasible: No blocking points have been identified so far. The approach is based on industrial practices and on existing operating experience;
- Fast: A couple months have been needed to provide first recommendations overview;
- With simple recommendations: The methodology leads to limited recommendations for diversity provisions on specific internal macro-parts. There are no needs to redesign new equipment;
- Efficient: Actual implementation of the diversity provisions allows considering critical CCF between redundant safety features as eliminated.

This approach has been not applied to I&C systems as appropriate guidance can be found in numerous reports dealing with “Diversity and DiD” assessment [6].

5. CONCLUSIONS

This innovative methodology development has requested a strong synergy between safety teams and machinery teams in order to provide a consistent approach, based on current industrial standard, from safety requirements to proposal of diversity provisions for component internal parts.

The main stages to address diversity requirements for important safety electrical and mechanical components could be summarized as follows:

- Identification of diversity requirements so that SSCs should be designed to deliver their safety functions for core damage prevention with adequate reliability;
- Preliminary system reliability analysis for determining the diversity requirements at component level;
- Detailed FMEA of the selected component taking into account operating experiences and focusing on the identification of coupling factors amongst redundant components leading to CCF;
- Proposition of diversity provisions for eliminating those coupling factors in the plant design or operation.

This approach carried out at the basic design stage of a new reactor allows characterizing diversity provisions to limit or eliminate CCF to consider in future component design. Thus it insures that the reactor design will fulfil both qualitative and quantitative high safety objectives for new NPP, as independence between DiD levels.

ACKNOWLEDGMENTS

Acknowledgments to Caroline LEROY, Etienne COURTIN and Sébastien VERMUSE for your support and technical review.

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CONSIDERATION OF DEFENCE IN DEPTH IN THE RECENTLY REVISED IAEA STANDARD ON DESIGN SAFETY

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Abstract

Defence in depth, “practical elimination”, and categories of plat states concepts are examples of requirements that were revised and emphasized in the International Atomic Energy Agency recently published revision to its standard on design safety requirements [SSR-2/1 “Safety of Nuclear Power Plants: Design”, 2012]. These requirements, developed in an international consensus, are designed to enhance the safety of nuclear power plants considering also lessons learned from the Fukushima Daichi accident. The requirements, if followed strictly, may have significant impact on the design of new power reactors as well as on safety analyses.

The application of passive safety systems and the strategy of early intervention and protection of the reactor from severe accidents and potential early and large releases may results in deployment of these systems at three levels of DiD simultaneously, therefore deviating from the defence in depth requirement as outlined in the IAEA standard. In this paper, we are providing an overall interpretation and discussing the potential impact of the revised IAEA standard on NPP design and specifically we are examining the application of the standard to passive designs using as example the AP1000 and the NuScale reactor designs.

1. INTRODUCION

In 2012 the IAEA issued a safety standard [1] that provides requirements for safe design of nuclear power plants (NPPs). This document replaced and older version of safety design requirements, NS-R-1 [2] which was published in 2000. The new standard assured consistency with Fundamental Safety Principles [3] that IAEA issued in 2006 and reflected the most current safety design knowledge.

The actions initiated by the IAEA after the Fukushima Daichi NPP accident in 2011 included review of the Safety Standards on the level of requirements with the goal of strengthening the standards and their implementation considering the lessons learned from the Fukushima accident [4]. One of the standards addressed was the SSR-2/1 standard that was at that point in publication (approved shortly before the accident). The review concluded that the safety requirement standards are adequate and revealed no significant weaknesses and only small amendments were proposed to strengthen the requirements. The proposed amendments were reviewed by the appropriate safety standards committees and endorsed in November 2014. The proposed revisions to the SSR-2/1 addressed the following areas:

- Prevention of severe accidents by strengthening the design basis of a plant,
- Prevention of unacceptable radiological consequences of severe accidents, and
- Mitigation of the consequences of severe accidents to avoid significant releases of radioactive materials.

The proposed revisions in the identified areas were developed by a group international experts and the IAEA staff, approved by relevant safety standards committees and the IAEA Board of Governors and the revised standard was issued in 2016 [5].

In the next sections we will discuss three safety concepts that are covered in the SSR-2/1 (both 2012 and 2016 versions) which have a significant impact on design and enhanced safety of new reactors: design extension conditions, conditions to be practically eliminated by design and the defence in depth concept.

This paper does address only some key issues and consequences of the design requirements based on SSR-2/1 Revision 1 [5].

2. DESIGN BASIS FOR NUCLEAR POWER REACTORS & DESIGN EXTENSION CONDITIONS

There are some significant differences between the older version (NS-R-1) and the newer version (SSR-2/1) of the design requirements affecting the design basis[1] for the plant design. One of the changes in the requirements are the plant states (conditions) to be considered in the design. Both versions consider in the same way the normal operation (NO), anticipated operational occurrences (AOOs) and design basis accidents (DBA). These three plant conditions are identified in the older requirements for the NPP design basis. All other plant states including severe accidents are considered as beyond design basis accidents for which no dedicated safety features were required and the safety of the plant was addressed through accident management. The new standard extends the design basis to provide dedicated safety features for accident conditions that previously were considered as beyond design basis conditions. These conditions are referred to as design extension conditions (DEC) and include complex accidents without (DEC 1) and with severe core damage including core melt (DEC 2). SSR-2/1 Revision 1 [5] provides the following definition of the design extension conditions: "*Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include conditions in events without significant fuel degradation and conditions with core melting*".

The plant states which the design of an NPP addresses are identified in [5] and are shown in the Fig. 1 below.

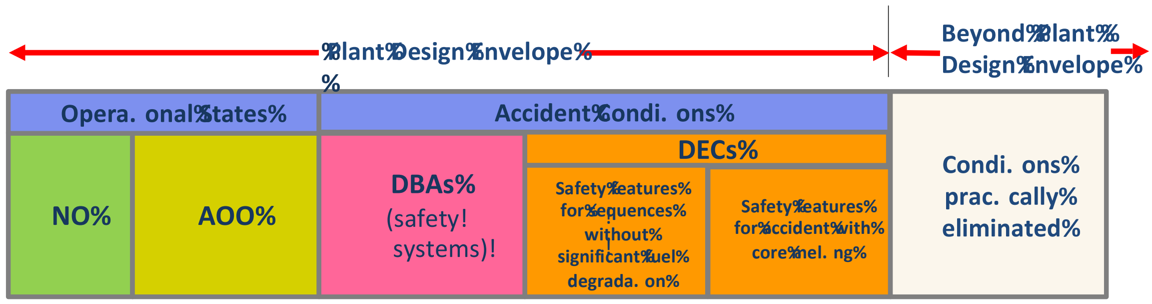


FIG. 1. Plant states to be considered in the design of a NPP.

The extended design basis is referred to as plant design envelope. Consideration of design extension conditions and their inclusion into plant design envelope should lead, as consequence, to practical elimination of significant radioactive releases[2].

The design extension conditions can result from a deviation of normal operation that has escalated due to multiple failures of safety systems caused by equipment failures or human errors [6]. These conditions and loss

[1] The **Design Basis** of a structure, system or component (SSC) is a set of information that identifies conditions and requirements necessary for the design of the SSC. Such a set usually includes:
- the functions to be performed by a structure, system or component of a facility;
- the conditions generated by operational states and accident conditions that the structure, system or component has to withstand;
- the conditions generated by internal and external hazards that the structure, system or component has to withstand;
- the acceptance criteria for the necessary capability, reliability, availability and functionality;
- specific assumptions and design rules.

[2] SSR-2/1 defines practical elimination as: "The possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high level of confidence to be extremely unlikely to arise."

of safety functions may be result from common cause failures (resulting in loss of redundancy) due to extreme internal or external hazards or human actions.

To fulfill the requirement imposed on design extension conditions to limit the radioactive releases within acceptable limits dedicated features need to be implemented in the design that will provide control over the accident. Features and equipment should be available that prevents DEC 1 to escalate to DEC 2 and for mitigation of the effects of core melt to maintain containment integrity and to prevent early or large releases. The features for the mitigation of DEC 2 prevent those severe accident phenomena such as hydrogen detonation, basemat melt through due to core-concrete interaction, steam explosions and containment bypass to practically eliminate the loss of containment integrity. The features used to control DECs are design dependent. Reference 6 discusses some typical features necessary to prevent early or large radioactive releases.

3. PRACTICAL ELIMINATION OF SIGNIFICANT RADIOACTIVE RELEASES

The IAEA safety standard defining the requirements for safe design [5] states "The safety objective in the case of a severe accident is that only protective actions that are limited in terms of times and areas of application would be necessary and that off-site contamination would be avoided or minimized. Event sequences that lead to early or large radioactive releases[3] are required to be practically eliminated".

4. DEFENCE IN DEPTH

Defence in depth (DiD) is a fundamental safety concept identified in the IAEA Fundamental Safety Principles [3] and used to the design of NPPs. However, the Fukishima accident and examination of NPPs that followed the accident, identified some weaknesses in implementation of defence in depth which led to some revisions and amendments of defence in depth related requirements in SSR-2/1.

The current approach to defence in depth, as identified in SSR-2/1 Revision 1 and discussed in more depth in [6], is summarized in the table below. The levels 1 and 2 of the defence in depth as defined in SSR-2/1 Revision 1 are very similar to previous definitions however as the table shows the main changes relate to Level 3 and 4. In the table the Level 4 of defence in depth was split into two sublevels to correlate it to the design basis accidents and design extension conditions without fuel melting. The Level 3, corresponds to the traditional approach to controlling DBAs by means of safety systems and emergency operating procedures. The safety systems are designed with a set of conservative, prescriptive rules and criteria (e.g. application of the single failure criterion) which provide high confidence in their success to meet the relevant acceptance criteria and safety limits. The reliability of equipment of level 3 of defence in depth is expected to be such that the probability of failure per demand is in the range of $10^{-3} – 10^{-4}$.

Level 4a was identified to stress prevention of core melt by protecting the core from the effects generated by multiple failures occurring in safety systems either in normal operation or following an AOO or a DBA (DEC1). The Level 4a expands the design for prevention of core melt. However, the design rules for SSCs for level 4a may be less conservative than those for level 3 but the acceptance criteria would be similar to Level 3.

If the protection of defence at Levels 3 and 4a fails core melt might occur (DEC 2). The essential means of achieving the objective of DiD at Level 4 include safety features for DECs and accident management procedures and guidelines. The application of single failure criterion is not required for the safety features for DEC. The accident management includes the use of non-permanent equipment as a measure to provide protection for containment integrity. Also, the SSR-2/1 Rev. 1 [5] requires the implementation of a Technical Support Centre (TSC) to provide technical support to the operation staff during accident conditions. Given its function, the TSC is an important and inherent feature for the Level 4 of the DiD.

[3] Ref. [SSR-2/1, Rev.1] ‘Large radioactive release’: a release for which off-site protective measures limited in terms of times and areas of application are insufficient to protect people and the environment. ‘Early radioactive release’: release for which off-site protective measures are necessary but are unlikely to be fully effective in due time.

TABLE 1. LEVEL OF DEFENCE IN DEPTH

Level of defence	Objective	Essential design means	Essential operational means	Corresponding plant state (conditions)
Level 1	Prevention of abnormal operation and failures.	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems.	Operational rules and normal operating procedures.	Normal Operation (NO)
Level 2	Control of abnormal operation and detection of failures.	Limitation and protection systems and other surveillance features.	Abnormal operating procedures/emergency operating procedures.	Anticipated Operational Occurrences (AOO)
Level 3	Control of design basis accidents (postulated single initiating events).	Engineered safety features (safety systems).	Emergency operating procedures.	Design Basis Accidents (DBA)
Level 4a	Control of design extension conditions to prevent core melt.	Safety features for design extension conditions without core melt.	Emergency operating procedures.	Design extension conditions without core melt (DEC 1)
Level 4b	Control of design extension conditions to mitigate the consequences of severe accidents.	Safety features for design extension conditions with core melt. Technical Support Centre.	Complementary emergency operating procedures/ severe accident management guidelines.	Design Extension Conditions with severe core damage (DEC 2)
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans	

Multiple consecutive levels of protection achieve the objective of DiD when failure of one of the levels of defence, does not result in failure of the subsequent level for the same cause (full dependency). The revision of the SSR-2/1 [5] stressed, based on lessons learned from the Fukushima Daichi accident, the requirement for independence of the levels of defence in depth (originally identified in the Fundamental Safety Principles [3]) by adding Paragraph 4.13A: "*The levels of defence in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of the other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.*"

It was recognized that a full independence of the levels of defence in depth cannot be achieved (due to issues such as exposure to external hazards which will be common to many SSCs, as well as unavoidable sharing of some SSCs such as the containment or the control room and ultimately the operating crew) [6]. This requirement can therefore be interpreted as a goal to reduce by design the dependence between the levels of defence. In the next section we attempt to examine briefly the application of the defence in depth requirement to a NPP design with passive safety systems.

5. DISCUSSION OF AP1000 DEFENCE IN DEPTH IN THE LIGHT OF SSR-2/1 REV 1 REQUIREMENTS

In the discussion of DiD in a passive NPP design we will focus only on the Level 3 and 4 of DiD. The approach to Level 1 and 2 is in plant with passive safety system not different to other plants. Also the issues of DiD Level 5 are not relevant to this discussion. To examine the application of the SSR-2/1 Rev 1. requirements to an NPP with passive safety systems we use AP1000 as currently the only design under construction that dominantly relies on the passive safety systems.

The new generation of SMRs is primarily equipped with passive system and the overall design significantly simplified, such as NuScale which we will examine as well. However, there is not much design detail in the public domain therefore, we can not perform any comprehensive review of the design response to accident conditions, however with the little information available we can point out the relation of the NuScale design to IAEA requirements for defence in depth features.

The Westinghouse AP1000 Pre-Construction Safety Report [6] provides description of the approach to DiD based on NS-R-1 [2] requirements:

— Level 1: Prevention of abnormal operation and failures by design.
— Level 2: Prevention and control of abnormal operation and detection of failures.
— Level 3: Control of faults within the Design Basis.
— Level 4: Control of severe plant conditions in which the Design Basis may be exceeded, including the prevention of fault progression and mitigation of the consequences of severe accidents.
— Level 5: Mitigation of radiological consequences of significant releases of radioactive substances."

The reference [6] identifies also the safety systems and features that are available to protect the plant from escalation of the accidents within levels 3 and 4 of DiD. For Level 3 these features include:

— Automatic reactor trip;
— Protection and safety monitoring system;
— Pressurizer safety valves;
— Passive residual heat removal (PRHR) heat exchanger;
— Automatic depressurization system (ADS);
— Core make-up tanks (CMT)
— Accumulators;
— In containment refuelling water storage tank (IRWST), its injection lines, screens and isolation valves;
— Containment recirculation lines, screens and isolation valves;
— Containment isolation system;
— Containment and the passive containment cooling system[4];
— Diverse actuation system.

The Level 4 safety measures are the SSC that mitigate the radiological consequences of severe accidents following the loss of a key safety function. On the AP1000, the following SSCs are in this category:

— Hydrogen igniters;
— Catalytic hydrogen recombiners.
— In-vessel retention capability;
— Post accident radioactive isotope containment by controlling the pH of the water in the containment sump;
— IRWST gutter isolation valves;
— The containment and provision of water to the passive containment cooling water storage tank beyond 72 hours into the accident sequence;
— Ancillary diesel generators;
— Spent fuel pool water sprays;
— Main control room emergency habitability system.

[4] The containment in AP1000 acts as the heat exchanger and removes the decay heat from the containment into the ultimate heat sink which is the environment air.

The AP1000 containment, the primary coolant system and the safety systems are a closed system with plenty of water and from which the energy is removed via the containment walls to the ultimate heat sink. The figurebelow provides an overview of the decay heat flow and emergency injection during plant accident conditions. The red arrows show the decay heat flow and the blue arrows show cooling water flow paths. These energy transport paths are valid for design basis as well as design extension conditions. This means that the same systems, as we discuss below, play key role in assuring decay heat removal in DiD levels 3 and 4 and therefore preventing early or large radioactive releases.

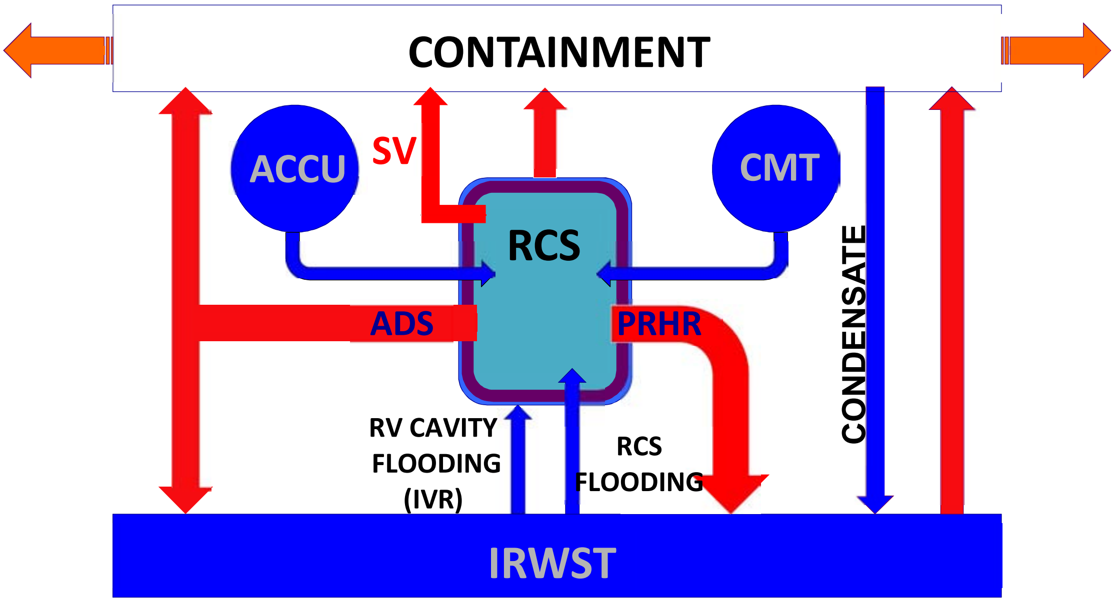


FIG. 2. AP1000 decay heat removal and emergency safety injection.

For discussion of a DEC 1 conditions and DiD Level 4a we consider a station blackout (SBO) event. Station blackout means the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency AC power system). A reactor should be designed with an ability to withstand and recover from an SBO of specified duration. The specified duration should be based on the probable time needed to restore offsite power. The IAEA design requirements [5] specifically address the issue of SBO with Requirement 68 which states that there shall be available an alternate power source that is capable of supplying necessary power to preserve the integrity of the reactor coolant system and to prevent significant core damage. The alternate power source shall be independent and physically separated from the emergency power supply. The connection time of the alternate power source shall be consistent with the depletion time of the battery.

The AP1000 safety design strategy is different from the approach described above. In case of a SBO event the water level in the steam generators will decrease in few minutes as no feedwater is available. The lower steam generator level will trigger the actuation of the passive residual heat removal (PRHR) inlet line by means of DC battery power. The resulting natural circulation through the PRHR will transfer the decay heat into the IRWST causing coolant temperature decrease and therefore pressurizer level decrease. The pressurizer lower level will trigger CMT valves opening and gravity injection of the borated CMT water into the reactor coolant system. After several hours enough decay heat is transferred to the IRWST resulting in boiling and steam flowing into the containment. The steam will be condensed on the colder containment vessel walls and the condensate will flow back to the IRWST. Containment vessel is being cooled by the passive containment system removing heat to the environment. This process is designed to continue as long as is needed to cool the reactor without any active means. It is evident that AP1000 uses in SBO event the same safety systems that are provided for the design basis accidents to protect the core. Therefore, from this perspective SBO in AP1000 is rather a design basis accident and is being controlled by safety systems used in Level 3 of DiD (Table 1).

In the case all the protection measures of the defence in depth at Levels 3 and 4a failed a core melting may occur and defence in depth Level 4b would be needed to prevent early or large radioactive releases. In such situation one of the worst challenges to the containment integrity is so called "direct containment heating" as a result of high pressure ejection of molten materials from the reactor vessel into the containment. The AP1000 design incorporated design features that prevent high-pressure core melt including the PRHR system and the automatic depressurization system (ADS). Depressurization of the AP1000 RCS in the event of an accident is provided by automatic or manual actuation of the ADS. The ADS consist of four different valve stages that open sequentially to reduce reactor coolant system pressure in a controlled fashion arranged into two redundant groups. Different valve types/sizes are utilized within the ADS stages to provide diversity. Based on these ADS design features, a highly reliable depressurization system is provided which precludes the potential for high-pressure core melt ejection in the AP1000 design.

The AP1000 design includes design features to retain and quench the core debris within the reactor cavity in the unlikely event of core debris relocation outside the reactor vessel. These features include:

— The reactor cavity geometry is arranged to provide a torturous pathway from the reactor cavity to the loop compartment and no direct pathway for the impingement of debris on the containment shell;
— A containment layout wherein the water accumulates in the reactor cavity region;
— The capability to manually initiate flooding of the reactor cavity by gravity draining the IRWST into the reactor cavity.

Clearly the same safety systems are used for dealing with these DEC 2 conditions, defence in depth Level 4b, as in design basis accidents. The containment internal compartment and structure design can be only credited as independent safety feature for DiD Level 4b for direct containment heating challenges. In summary, it can be claimed that for accidents with core materials relocated outside the reactor vessel, the AP1000 defence strategy with reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper compartment will protect containment integrity despite no clear distinction between the levels of defence in depth 3, 4a and 4b (based on strict interpretation of IAEA requirements).

In the table below we listed the AP1000 safety systems and features and identified the accident conditions (DBA, DEC 1, DEC 2) and associated levels of defence in depth in which these systems and features are used.

TABLE 2. AP1000 SAFETY SYSTEMS AND FEATURES DESIGNED FOR ACCIDENTS CONDITIONS: DBA, DEC 1 AND DEC 2

Safety systems / features	DiD Level 3 (DBA)	DiD Level 4a (DEC 1)	DiD Level 4b (DEC 2)
Protection and safety monitoring system;	active	active	Active – includes capability to be used for SA management
Pressurizer safety valves	Available	Available	Available
Passive residual heat removal (PRHR) heat exchanger	Active	Active	Active
Automatic depressurization system	Active	Active	Active
Core make-up tanks	Active	Active	
Accumulators;	Active	(Active)	(Active)
IRWST, injection lines, screens and isolation valves	Active	Active	Active
Containment recirculation lines, screens and isolation valves	Active	Active	Active
Passive containment cooling system	Active	Active	Active
Containment isolation system	Active	Active	Active
Diverse actuation system	Active	Active	Active

As the focus is on Level 3 and 4 of DiD we considered only passive safety and other relevant features. Despite accident management is an inherent part of Level 4 we are not including it in the table to illustrate the requirement of independence of DiD levels.

The table shows that there is no clear separation, based on the IAEA approach, between the levels of defence in depth and basically the same systems will be used to cope with all accidents conditions (DBA, DEC 1 and DEC2). This approach does not make these reactors less safe as compared to reactors with multiple active systems that would be activated consecutively on different levels of defence. The AP1000 strategy is not wait until a certain level of defence fails but to create conditions in which the escalation of an accident is mitigated early in the evolution of the acctident. Naturally, these claims should be supported by detailed deterministic analyses.

6. DISCUSSION OF A SMR - NUSCALE, DEFENCE IN DEPTH IN THE LIGHT OF SSR-2/1 REV 1 REQUIREMENTS

The NuScale reactor is a new design and requires some introduction. According to IAEA Advanced Reactor Information System [9] and a supplement [10] the NuScale is a small pressurized water reactor. A power plant may consist of a single reactor or of several power modules.[5] Thermal power of a single module is 600 MW(th) and electrical output is 50 MW(e) [8]. The integrated reactor pressure vessel (RPV) contains the core, two helical coil steam generators and a pressurizer. Natural circulations assures for heat transport from the reactor core to the steam generators in which the boiling occurs inside of the tubes. The RPV is located in a steal containment vessel in which vacuum is maintained during normal operation. The containment vessel is submerged in a water pool that is located in the reactor building below the grade. Figure 3 below provides an overview of NuScale power module design and more public information about the design can be found in [9].

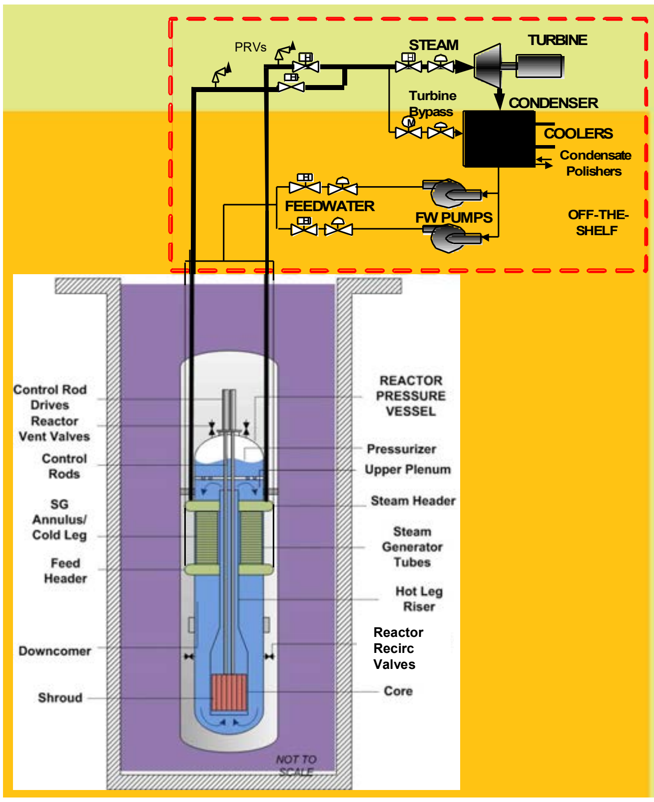


FIG. 3. NuScale reactor power module design schematic [9].

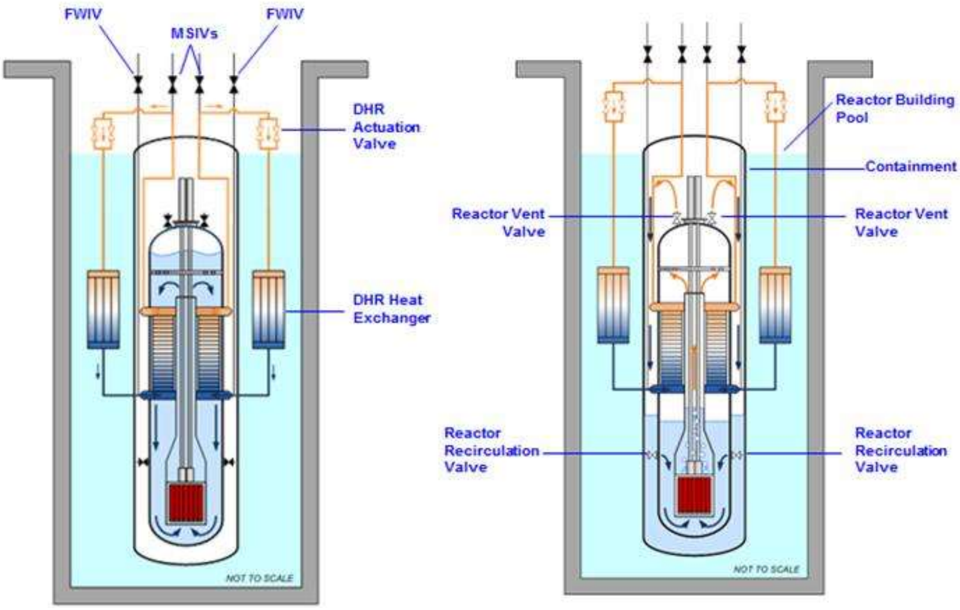
There are two safety systems which do not require external power for actuation. One is the two-train Decay Heat Removal System (DHRS), that provides reactor cooling for non-LOCA events when normal feedwater is not available. The system is a closed-loop, two-phase natural circulation cooling system. After the main steam line

[5] Currently licensing of a twelve power module plant was initiated. In this paper we will focus only on a single power module aspects of defence in depth.

isolation valves and the main feedwater isolation valves are closed the steam from the steam generators is guided to the heat exchangers where it is condensed and the condensate flows back to the feedwater lines (Fig. 4a). Each train is designed to remove 100 percent of the decay heat.

The second safety system is the Emergency Core Cooling System (ECCS) as shown in Figure 4b, consists of two independent Reactor Vent Valves (RVVs) and two independent Reactor Recirculation Valves (RRVs). The decay heat is removed from the reactor vessel by opening the RVVs. The steam condenses on the containment walls, is collected in the lower parts of the containment and provides cooling to the reactor vessel walls. When the level of the condensate reaches, the recirculation valves the coolant returns through these valves to the reactor vessel and natural circulation through the reactor core is established. Heat is transferred by conduction through the containment walls to the reactor pool.

For both modes of decay heat removal the pool assures energy removal, without boiling, for about three days. If no actions are taken, the pool water evaporates within 30 days, and after that the decay heat s removed via natural convection to the air for infinite time.



a) DHRS *b) ECCS*

FIG. 4. Emergency decay heat removal [9], a) natural circulation on the secondary side, b) through reactor vessel venting into the containment.

As in the AP1000 design, the containment and the primary system of the NuScale is a closed system during accident conditions. The energy and coolant flow paths are shown in Figure 5. This figure shows how simple is the design and the strategy of accident mitigation.

The decay heat transport paths are valid for design basis and design extension conditions. During design basis accidents when the primary system is not breached the DHRS can remove the decay heat from the reactor system to the reactor pool, from which the energy dissipates to the environment. In case the DHRS is not available or there the primary system is breached the ECCS takes care of core cooling. If there is a severe accident with core melt (DEC 2) ECCS is also responsible for cooling as the reactor vessel will be surrounded by water and in-vessel retention will be assured. Success of the accident mitigation of design basis accidents and design extension conditions depends therefore significantly on reliability of containment isolation – all steam lines and feedwater lines must be closed. Naturally also high reliability is required for the reactor vent valves and recirculation valves.

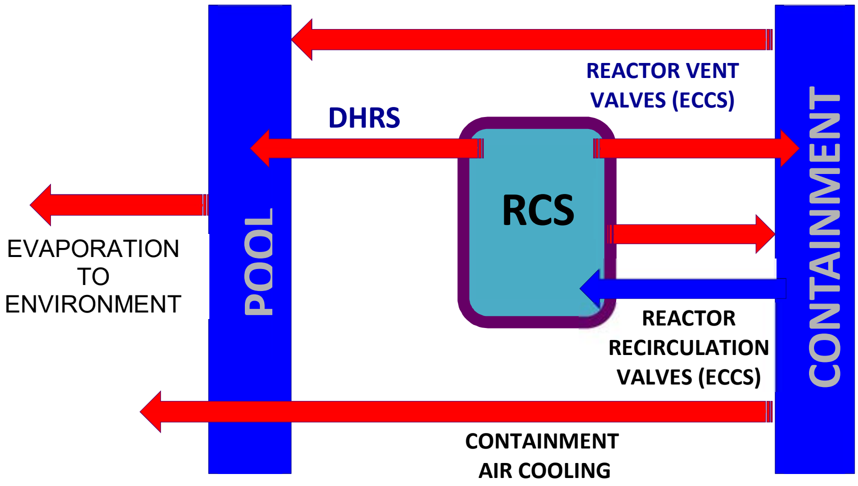


FIG. 5. NuScale decay heat removal and emergency safety injection flow paths.

Unfortunately, we did not find in the open literature any clear description of NuScale response to DEC events. However, the typical challenges to the containment integrity seem to be eliminated by design. For example high pressure melt ejection and direct containment heating is unlikely because of early opening of the paths between the reactor vessel and the containment, either through the safety valves or the vent and recirculation valves. Hydrogen deflagration in the containment is not expected because there is no oxygen in the containment. Steam explosion within the reactor vessel are also unlikely because of early initiation of the core cooling via ECCS. For the same reason steam explosions in the containment are unlikely. There is no potential for MCCI which is a long-term challenge to the containment as there is no concrete in NuScale, and if core debris shall land in the lower parts of the containment it will be well cooled by the water present in the containment.

It can be expected that SBO would trigger scram and isolation of the steam and feedwater lines. This would result in rise of pressure in the reactor vessel which would cause opening of safety valves relieving steam from the pressurizer into the containment. After few cycles of these valves reactor vent valves and recirculation valves would open and a long-term cooling of the reactor core would be initiated as described above.

There is not enough information in the public domain to make a comprehensive assessment of NuScale compliance to IAEA design standards. However, the information available indicates that the requirements for defence in depth might not be entirely applicable to designs like NuScale (or AP1000), and that evaluation of defence in depth effectiveness cannot rely only on deterministic methods and criteria. The defence in depth needs to be considered differently for such designs and some requirements for safety features and their independence DiD Level 3 and 4 replaced/achieved with reliability requirements. The safety of systems like NuScale will rely particularly on reliability of the valves assuring containment isolation and of those, which actuate the passive safety systems such as DHRS and ECCS.

7. CONCLUSIONS

The IAEA design requirements provide for enhanced safety of NPPs particularly with respect to potential early or large release of radioactive materials into the environment (practical elimination). This requirement is particularly important as it leads to extension of the design basis and if its implementation is successfully demonstrated it leads also to increased public acceptance.

Application of these requirements to some designs is not straight forward as we discussed above. NPP designs that use prevention and mitigation strategy such as the AP1000 or NuScale designs might not fully comply with defence in depth principles as presented in SSR-2/1 Rev 1, as the application of the passive systems may results in their activation and deployment at three levels of DiD simultaneously. These designs also might not fully meet the requirement for separation between safety systems (dedicated to the design basis accidents) and safety features (dedicated systems for design extension conditions). The combination of the IAEA design safety requirements related to the plant sates/conditions and defence in depth requirement leads to increased complexity of safety systems/features. However, on the other hand, the same level of safety could be assured through simpler designs such as AP1000 or NuScale as the result of simplification and reliance on natural phenomena and of the emphasis on early intervention and protection of the reactor from severe accidents and potential early and large releases. This approach does not make these reactors less safe as compared to reactors with multiple active systems that would be activated consecutively on different levels of defence. Comparison between those two approaches, based on in comprehensive integrated analyses using deterministic and probabilistic methods should be performed to validate whether there any differences with respect of protection of the reactor and ultimately prevention of early and large releases. In general, the deterministic evaluation should be based on multi-physics best estimate plus uncertainty methods with the capability to simulate the control systems included. Also, in the light that several new simplified designs with passive systems are on the nuclear power market, an IAEA interpretations of the SSR-2/1 requirements, similar to those provided in Reference 6, would be of benefit for comprehensive assessment of the new designs safety.

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HPR1000: ADVANCED PWR WITH ACTIVE AND PASSIVE SAFETY FEATURES

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Abstract

HPR1000 is an advanced pressurized reactor with the significant feature of active and passive safety design philosophy. On one hand, it is an evolutionary design based on proven technology of existing pressurized water reactor nuclear power plant, on the other hand, it incorporates advanced design features including 177-fuel-assembly core loaded with CF3 fuel assemblies, active and passive safety systems, comprehensive severe accident prevention and mitigation measures, enhanced protection against external events and improved emergency response capability. Extensive verification experiments and tests have been performed for the critical innovative improvements such as passive systems, reactor core and main equipment. Active and passive safety design is the most remarkable innovation for HPR1000 and also a typical instance to fulfill the diversity criteria. The design inherits the mature and reliable active technology, and also introduces passive system as the backup for active system in case of loss of AC power. Both active and passive features are employed to guarantee the safety functions of emergency core cooling, residual heat removal, In-vessel Retention (IVR) of molten core, and containment heat removal. The innovative design safety features, e.g. passive systems, in-vessel melt retention, will be illustrated in the paper.

1. INTRODUCTION

The use of nuclear energy for electricity generation began in the late 1950s and went through several phases. The designs of nuclear power plants (NPPs) are also categorized by “generation” accordingly. After prototype reactors of Generation I and commercial reactors of Generation II, Generation III Light Water Reactor (LWR) NPPs incorporated state-of-the-art improvements in the areas of fuel technology, thermal efficiency and safety systems [1, 2].

The safety systems design of pressurized water reactor (PWR) NPPs has also experienced three phases of development: First phase which uses rather original and simple safety system, developing to the second phase which takes more complicated active safety system focusing on dealing with the design base accidents (DBA). Before Fukushima Daiichi NPP accident, for the third generation, there are mainly two different design concepts representing two trends of safety system design, one is passive safety design concept with simplified safety systems, and the other is active safety design concept with more redundancy, safety system is more and more complicated.

The Fukushima Daiichi NPP accident draw the worldwide attentions towards the safety of NPPs. International Atomic Energy Agency (IAEA), governments or nuclear safety authorities respectively issued special reports on the lessons learned from Fukushima accident, which focused on the area such as protection against external events, robustness of emergency power and ultimate heat sink, safety of spent fuel pool, emergency response for multiple-unit accident, and inhabitability and availability of emergency facilities [3]. The safety inspections or stress tests, and necessary improvements were made for the current NPPs based on the Fukushima feedbacks. Meanwhile the safety requirements for new NPPs were also considered and discussed, which are reflected in documents like Safety of New NPP Designs drafted by Western European Nuclear Regulator’s Association (WENRA), Safety of Nuclear Plants: Design (No. SSR-2/1, Rev.1) issued by IAEA, and Safety Requirements for New NPPs during 12th Five-Year Plan drafted by China National Nuclear Safety Administration (NNSA). The improved safety requirements for new NPPs in the above documents generally covered the following aspects: revised and strengthened defence-in-depth approach, response capability for Beyond-Design-Basis Accident (BDBA) including multiple failures, practical elimination of large radioactivity release to mitigate off-site emergency, and protection against internal and external hazards. In addition, the concepts like residual risk and plant autonomy period were also brought into the horizon of international nuclear power industry.

In the background that the technology of advanced NPPs has been mainstreamed and nuclear safety standard for new NPPs in post-Fukushima era will be stricter, China National Nuclear Corporation (CNNC) developed the evolutionary advanced Pressurized Water Reactor (PWR) HPR1000. The design makes full use of the proven technology based on the design, construction and operation experience of large PWR fleet in China, and introduces a number of advanced design features to meet the latest safety requirements and address the feedback from Fukushima accident.

2. HPR1000 SAFETY DESIGN PHILOSOPHY

The fundamental safety functions to be ensured for the nuclear power reactors are: 1) control of the reactivity, 2) removal of heat from the core and the spent fuel, and 3) confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases. For HPR1000, in order to achieve the safety functions, the concept of defense-in-depth is performed throughout all the safety-related actives to ensure that they are subject to over-lapping provisions. Structure, systems and components important to safety shall be capable of withstanding identified initiating events with sufficient robustness, which are ensured by the design criteria such as redundancy, diversity and independence.

Generally, from the probabilistic point of view, the lower reliability of a single train, the greater contribution will be made by increasing in redundancy for system reliability improvement. Conversely, if the reliability of a single train is high enough, the increase in redundancy only has a limited contribution to system reliability improvement. For example, see Table 1, if the reliability of a single train is 0.9, adding a train can increase the system reliability by 10%, and adding the second train can only increase the system reliability by 1%. Furthermore, the contribution of redundancy to system reliability improvement will be smaller if taking common cause failure into account.

TABLE 1. RELIABILITY (WITHOUT COMMON CAUSE FAILURE)

Redundancy (Number of Trains)	1	2	3
Reliability	0.9	0.99（+10%）	0.999（+1%）
	0.1	0.19（+90%）	0.271（+43%）

As addressed in IAEA SSR-2/1 “for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures.” Whether it is from the deterministic and probabilistic point of view or referring to the practical operating experience, the widely adopted design of safety system with two active trains is sufficient to meet the latest safety requirements, including the single failure criterion. Therefore, to add redundancy on DBA level for NPP design, which not only has limited improvement for NPP safety, but also wastes precious resources, as the well-known Buckets effect.

During the preliminary design phase of HPR1000, the safety optimization or improvement decisions are made based on the identification of risk-informed methodology. For example, for the engineered safety system designated for DBA with 2 active trains, the improvement mainly focused on the vulnerabilities, not simply increase the redundancy. To effectively increase the capability of withstanding design extension conditions and improve integral safety, the passive safety systems are introduced. Both active and passive features can guarantee the safety functions of emergency core cooling, residual heat removal, In-vessel Retention (IVR) of molten core, and containment heat removal, as shown in Fig. 1.

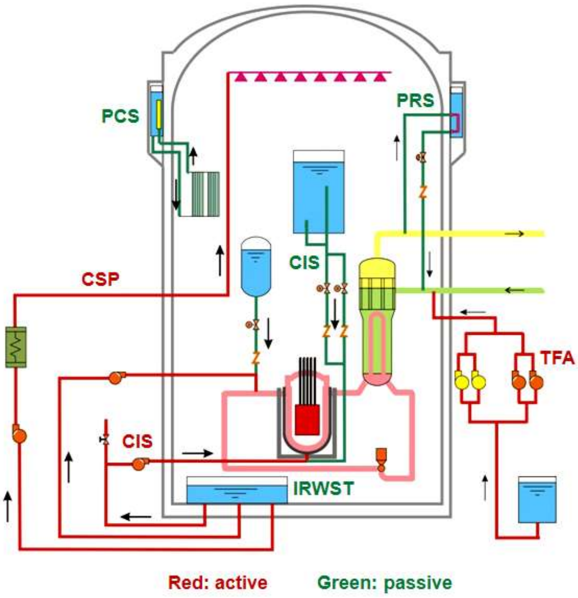

FIG. 1. Active and passive safety system configuration of HPR1000.

The Probabilistic Risk Assessment (PRA) result shows that this configuration has significant effect for the safety improvement, as shown in Table 2.

TABLE 2. PRA RESULT OF HPR1000 SAFETY SYSTEM CONFIGURATION

Configuration of safety system	CDF/ reactor·year	ΔCDF compared to HPR1000	The rate of change in CDF compared to HPR1000
2 Active Trains	6.11E-07	+4.81E-07	+370.00%
3 Active Trains	3.32E-07	+2.02E-07	+155.38%
2 Active Trains and 1 Passive Train (HPR1000, baseline configuration)	1.30E-07	-	-
3 Active Trains and 1 Passive Train	1.11E-07	-1.90E-08	-14.62%

3. HPR1000 SAFETY FEATURES

The general parameters of HPR1000 are presented in Table 3. The major safety features of HPR1000 are briefly introduced from the aspects of engineered safety features, severe accident prevention and mitigation measures.

3.1. Engineered safety features

The engineered safety features are adopted to mitigate DBA, which mainly include Safety Injection System, Auxiliary Feedwater System, and Containment Spray System, see Fig. 2. Engineering safety features are comprised of redundant trains to fulfil single failure criterion. The independence is ensured by means that each train is arranged in physically separated building and supplied power by each of the emergency diesel generators.

TABLE 3. GENERAL PARAMETERS OF HPR1000

Parameter	Values
Reactor thermal output	3050 MWt
Power plant output, gross	~1170 MWe
Power plant output, net	~1090 MWe
Power plant efficiency, net	~36%
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target	⩾90%
Refueling cycle	18 Months
Safety Shutdown Earthquake (SSE)	0.3 g
Core damage frequency (CDF)	< 10E-6 /Reactor-Year
Large release frequency (LRF)	< 10E-7 /Reactor-Year
Occupational radiation exposure	＜0.6 Person-Sv/ Reactor-Year
Operator Non-intervention Period	0.5 Hour
Plant Autonomy Period	ours

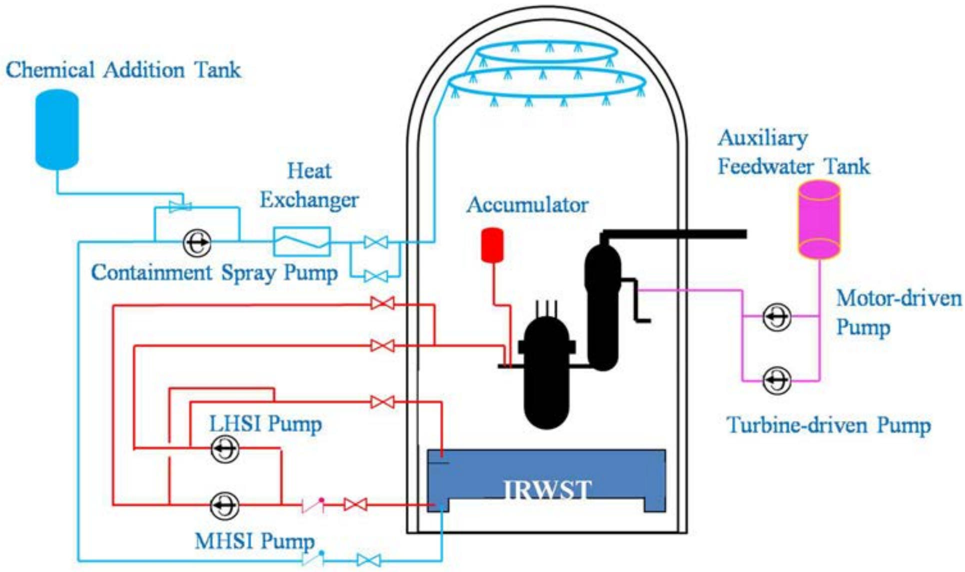

FIG. 2. Engineered safety features of HPR1000.

Safety Injection System consists of two active subsystems, i.e. Middle Head Safety Injection (MHSI) subsystem and Low Head Safety Injection (LHSI) subsystem, and one passive system, i.e. accumulator injection system. The In-Containment Refueling Water Storage Tank (IRWST) is adopted, with the benefits of providing protection against external events and saving the need for switching the water sources during long-term injection in comparison with the refueling water tank outside the containment. The MHSI and LHSI pumps take suction from the IRWST after Loss-of-Coolant Accident (LOCA), and inject the boron water to RCS to provide emergency core cooling to prevent core damage. With regard to the improved configurations compared to the existing NPPs, the safety injection pumps are not shared with other system to improve the equipment reliability and independence, the injection head is lowered to reduce the possibility of SGTR, and the boron injection tank and boron recirculation loop are cancelled to achieve system simplification.

Auxiliary Feedwater System supplies emergency feedwater to the secondary side of SG to remove the core decay heat in case of loss of normal feedwater. The feedwater is provided from two auxiliary feedwater pools with 2×50% motor-driven pumps which are backed up by emergency diesel generator and 2×50% turbine-driven pumps which are driven by steam produced by SGs. The diversity of the pumps improves the robustness of the system.

Containment Spray System is used to maintain the integrity of containment by limiting containment pressure and temperature within the design limits, by spraying and cooling the steam released in containment during LOCA or Main Steam Line Broken (MSLB) accidents. The spray water is taken from IRWST by spray pump, with chemical additive to reduce the airborne fission products (especially iodine) and limit the corrosion of structure material. LHSI pump can be used as back up for containment spray pump to ensure the reliability of long-term spray.

3.2. Severe accident prevention and mitigation measures

Comprehensive prevention and mitigation measures have been incorporated in HPR1000 against all the possible severe accident scenarios, including high pressure molten corium ejection, hydrogen detonation, basement melt-through, and long-term containment overpressure. For the specific BDBA considered as the weak point for the existing NPP such as Station Black-out (SBO), appropriate measures are also taken into account in the design. See Fig. 3.

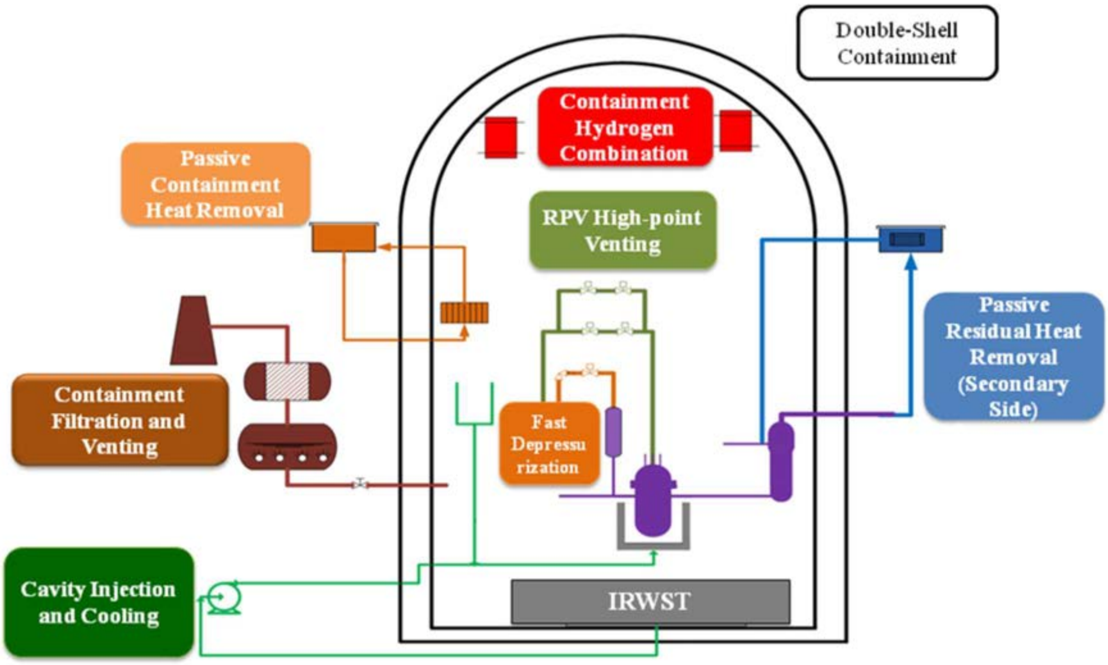

FIG. 3. BDBA/Severe accident prevention and mitigation measures of HPR1000.

Fast Depressurization System of Primary Loop is used to depressurize RCS rapidly during severe accident in order to prevent the high pressure molten corium ejection, which would cause direct containment heating. The system consists of two redundant parallel discharge lines connected to a nozzle on the pressurizer dome. Each line is installed with a gate valve and a globe valve in series.

RPV High-point Venting System is designed to remove non-condensable gases from RPV head during accident condition, so as to avoid adverse impact to the heat transfer of reactor core caused by the non-condensable gases.

Cavity Injection and Cooling System (CIS) is used to cool the external surface of lower dome of RPV by injecting water into the space between RPV surface and insulation layer, so as to maintain its integrity and realize IVR of melt core debris. CIS consists of an active subsystem and a passive subsystem. The active subsystem includes two parallel injection lines, each with a pump taking water from IRWST or firefighting water as backup. The passive subsystem is located in the containment with a high-point tank. In case of severe accident and the failure of active subsystem, the isolation valves can be opened and the water in the tank flowed by gravity to cool the lower dome of RPV.

Passive Residual Heat Removal System of Secondary Side (PRS) is put into action in the condition of SBO and the failure of the turbine-driven auxiliary feedwater pumps, to provide feedwater to the secondary side of SG in a passive way. PRS consists of three trains connected to three SGs, respectively. The nature circulation will be established in the closed loop between secondary side of SG and the heat exchanger submerged in the Heat Exchange Tank on the upper part of outer containment. The tank inventory can sustain the operation of PRS for 72 hours.

Containment Hydrogen Combination System is intended to decrease hydrogen concentration within the containment atmosphere to safe limits, to prevent hydrogen inflammation during DBA or hydrogen detonation during severe accident. The system is comprised of more than 30 passive autocatalytic recombiners installed inside the containment, which will be triggered automatically when the hydrogen concentration reaching the threshold.

Passive Containment Heat Removal System (PCS) is designed to remove the heat in the containment, to ensure that the containment pressure and temperature will not exceed the design limits during BDBA. The heat of high-temperature mixture of steam and gas inside the containment will be removed by water flowed in tubes of heat exchangers installed at high position on the internal surface of containment, to Heat Exchange Tank outside the containment. The temperature difference between the mixed gas within the containment and the water in Heat Exchange Tank, and the elevation difference between the tank and the heat exchanger are the driving force for natural circulation to remove the containment heat. The water in Heat Exchange Tank is heated and evaporated when the saturated temperature is reached, and heat finally dissipates to atmosphere. The tank inventory fulfils the requirement of 72-hour passive heat removal from containment after severe accident.

Containment Filtration and Venting System is an option to prevent the pressure of the containment from exceeding its bearing capability by initiative and planned venting. The filtration equipment in the venting line is used to reduce the discharge of radioactivity to atmosphere as much as possible.

3.3. Protection against the external events

The protection against the external events is dramatically enhanced. The containment is a double-wall structure, and the outer containment is made of reinforced concrete and withstands external events such as aircraft crashes, explosions and missiles. The nuclear island buildings are designed with a seismic input of peak ground acceleration of 0.3g for both horizontal and vertical directions. A seismic margin assessment is also performed to evaluate the plant's resistance to a beyond-design-basis earthquake. Protection against a large commercial aircraft crash is achieved by concrete shielding shells for the Reactor Building, Fuel Building and Electrical Building, and by complete physical separation for the Safeguard Buildings as shown in Fig. 4.

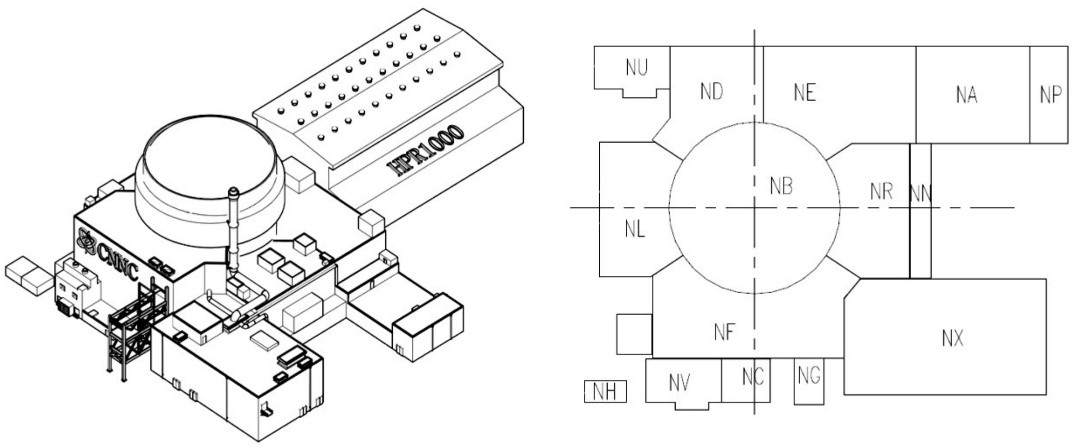

FIG. 4. General layout of nuclear island of HPR1000. (NB - Reactor Building; NF - Fuel Building; ND/NE - Electrical Building; NL/NR - Safeguard Building; NX - Nuclear Auxiliary Building).

4. CONCLUSION AND OPTIMIZATION FOR FURTHER DEVELOPMENT

In general, the active and passive safety design is the most remarkable innovation for HPR1000 and also a typical instance to fulfil the diversity criteria for Fukushima Daiichi NPP accident scenario. The design inherits the mature and reliable active technology, and also introduces passive system as the backup for active system in case of total loss of AC power and heat sink accident, to make sure integrity of last safety barrier to realize a large radioactive release can be “practically eliminated” in design.

Based on the R&D of HPR1000, further development of PWR technology is also considered by the designer following the latest trend of safety requirements and the marketing prospect. As a kind of product, the nuclear power development should not only focus on the safety, but also take into account the cost and constructability. The application of all the important plant design rules, including the single failure criterion, redundancy, diversity, etc. is actually a compromise between safety, reliability，economy and operability of guidelines on the basis of engineering judgement and combination of deterministic assessment and probabilistic assessment. It may be not entirely reasonable with extending each of criterions to everywhere, which could lead to imbalance in NPP design.

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APPROACHES TO PRACTICAL ELIMINATION

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APPROACH TO PRACTICAL ELIMINATION IN FINLAND

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Abstract

The concept of practical elimination has been an elemental part of the design of nuclear power plants. However, there has been a need to revisit this concept after the Fukushima Daiichi accident in 2011 and the lessons learned have lead to further elaboration of the concept and enhancement of related regulatory requirements in many countries. In Finland the updated requirements on practical elimination were issued in 2013. The paper will present the Finnish requirements related to practical elimination concept and their development as well as practical examples on how the concept is applied in the operating NPPs and NPPs under construction.

According to Nuclear Energy Act, the concept of continuous improvement of safety is applied in Finland and the safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further enhancement of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology. The practice of implementation of the updated regulatory requirements for the operating nuclear power plants and related safety enhancements will also be presented.

The Finnish approach for practical elimination of large or early releases is driven by limiting the overall frequency of initiating events that could lead to such a situation. Specific approach for practical elimination is addressing sequences, for which mitigation is not feasible due to the nature of the phenomena resulting from some specific initiating events. The possibility of occurrence of such sequences has to remain extremely low, which supports also achieving the probabilistic overall safety goal.

1. INTRODUCTION

The concept of practical elimination has been an elemental part of the design of nuclear power plants. However, as a concept the practical elimination was first introduced in INSAG-12 in 1999 in context of design requirements for new reactors. In the Finnish legislation and subordinate regulations the terminology is not used. This leaves room for interpretation on how the concept is applied. However, there has been a need to evaluate both national and international requirements after the Fukushima Daiichi accident and the lessons learned have lead to further enhancement of the regulatory requirements dealing with practical elimination in many countries.

In Finland the concept of continuous improvement of safety is applied and the safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further enhancement of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology. As the YVL Guides are written for the new nuclear power plant separate decisions are taken on the application of updated YVL Guides to operating plant or plants under construction.

In the new set of YVL Guides issued 2013 the requirements for practical elimination are presented systematically. The implementation of the updated regulatory requirements for the operating nuclear power plants has been completed in 2015-2016 and resulting safety enhancement projects are ongoing. Furthermore the licensees are required to analyse a wider spectrum of initiating events compared to examples given in Guide YVL B.1. The analysis with conclusions shall be submitted to STUK before the end of this year. The fulfilment of practical elimination concept and related requirements has been reviewed in the periodic safety review of the

Fortum's Loviisa 1 and 2 finished in 2017, and will be reviewed in the renewal of the operating licence of TVO's Olkiluoto 1 and 2.

The Olkiluoto 3 NPP unit is currently in operating license phase and the fulfilment of practical elimination concept and related requirements is being reviewed as part of licensing process.

Fennovoima's Hanhikivi nuclear power plant project is now in construction license phase and the application of practical elimination concept and related requirements are being discussed and reviewed as part of the licensing process.

2. THE DEVELOPMENT OF THE FINNISH REQUIREMENT ON PRACTICAL ELIMINATION

2.1. Overview of development of the concept

The term "practically eliminated" was originally introduced in the IAEA publications in INSAG-12 [1] in 1999 and introduced into the IAEA Safety Standard Series for the first time in 2004 in Safety Guide NS-G-1.10 [2] dealing with the containment systems of the nuclear power plants. The NS-G-1.10 explanation of "practically eliminated" state that "In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise". After the TEPCO Fukushima Daiichi nuclear power plant accident in March 2011 the concept of "practical elimination" of some specific initiating events leading to early or large releases has been dealt in several international publications.

In the IAEA Safety Fundamentals published in 2006 Principle 8 states that measures shall be taken to ensure that the likelihood of an accident having harmful consequences is extremely low. In the IAEA requirements level document the "practical elimination" of early or large releases was introduced in design requirements of the nuclear power plants SSR-2/1 [3] published in 2012. There has been a need to evaluate these requirements after the TEPCO Fukushima Daiichi nuclear power plant accident in March 2011 and the lessons learned have lead to further enhancement of the requirements. The updated IAEA nuclear power plant design requirements SSR-2/1 Rev.1 (2016) [4] systematically address the requirements for "practical elimination" of early or large releases. The ongoing updating of the IAEA Safety Standards introduces the concept explicitly to the safety guides dealing with design or safety demonstration such as deterministic analysis, containment system design or content of the safety analysis report [5, 6, 7]. More technical discussion of the concept is presented in IAEA TECDOC 1791 (2016) [8] on application of SSR-2/1 requirements.

The explanation of IAEA for the "practically eliminated" has been widely adopted. Footnote 4 in SSR-2/1 Rev.1 states "*The possibility of certain conditions arising may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise. In the safety demonstration the physically impossible is considered to be the preferred method. However, the distinction in between these two ways of demonstration is not always simple.*"

Practical elimination of early or large releases deals with the strength of the Defence-in-Depth and demonstration of the effectiveness of the protection, and on the other hand prevention of certain conditions from arising threatening the integrity of the containment. These conditions could be grouped into five categories as shown in the IAEA TECDOC-1791:

1. Events that could lead to prompt reactor core damage and consequent early containment failure:
    a. Failure of a large component in the reactor coolant system (RCS);
    b. Uncontrolled reactivity accidents.
2. Severe accident phenomena which could lead to early containment failure:
    a. Direct containment heating;
    b. Large steam explosion;
    c. Hydrogen detonation.
3. Severe accident phenomena which could lead to late containment failure:
    a. Molten core concrete interaction (MCCI);
    b. Loss of containment heat removal.
4. Severe accident with containment bypass;
5. Significant fuel degradation in a storage pool.

The safety goals for new reactors published by WENRA in 2010 [9] include safety objective for accident with core melt considering the “practical elimination” of early or large releases and WENRA considers in its reference levels (for the existing reactors) published in 2014 the concept of “extremely unlikely with a high degree of confidence” [10]. The concept of practical elimination is discussed more in the WENRA report on Safety of new NPP designs published in 2013 [11]. This report clarifies the concept of practical elimination as “*Accident sequences that are practically eliminated have a very specific position in the Defence-in-Depth approach because provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design*”. In addition, the paper takes a wider perspective: “*All accident sequences which may lead to early or large radioactive releases must be practically eliminated.*”, which arises directly from WENRA Safety Objective O3 “Accidents with core melt”.

The international discussion on the topic builds up on understanding of the different approaches as the integration of the concept into practice is done. This discussion reveals also the difficulty in building the common understanding around the concept. This is interesting acknowledging that “practical elimination” is crucial part of the safety demonstration of a nuclear power plant. OECD NEA Green Book [12] published in 2016 discusses the “practical elimination” of the significant releases through Defence-in-Depth.

2.2. Historical development of practical elimination concept in the Finnish nuclear safety regulations

The Finnish requirements were established right from the beginning of the use of nuclear energy in Finland. The fist 55 criteria were presented in the General Design Principles of a Nuclear Power Plant in 1976 [13]. The US requirements 10CFR50, Appendix A /14/ were used as a reference when developing the Finnish requirements. At that time no systematic approach was used in the requirement to quantify the need for safety demonstration. As an example the prevention of occurrence of a certain condition such as a criticality accident or loss of the integrity of the containment was required. The expression of “the possibility of … shall be extremely low” was used for the integrity of primary circuit and simultaneous loss of internal and external electric power. The Guide YVL 1.0 Design Principles of a Nuclear Power Plant published in 1982 [15] represented the state-of-the-art guidance at that time and good coverage of the safety issues. For instance, the severe accident management was required for the design of new nuclear power plants. From the middle of 1980s the IAEA Safety Standards were considered in the development of the YVL Guides.

The renewal of the Atomic Energy Act (356/1957) and publication of the Nuclear Energy Act (990/1987) and subordinate regulations systematized the approach for setting safety requirements at different levels in the Finnish regulatory framework. The safety requirements were issued for new reactors and principle that all the reasonable practicable safety improvements shall also be made for the already licensed nuclear power plants was applied. For the first time in 1991 the binding safety requirements were presented in the Government Decree on safety of a nuclear power plant (395/1991) [16]. In the legislation and subordinate regulations the terminology “practical elimination” was not used. This left room for interpretation in application of the concept. In Finland the terms such as “the possibility of … shall be extremely low” were used in this context.

The updating of the requirements in Government Decree on safety of a nuclear power plant (733/2008) [17] in 2008 considered the safety enhancements required for the new build and lessons learned form the start of the construction of the Olkiluoto 3 nuclear power plant. The update of the Government Decree on safety of a nuclear power plant (717/2013) took into account also the TEPCO Fukushima Daiichi accident in March 2011. These updated requirements are reflected in the total revision of the set of YVL Guides conducted during years 2006 to 2013. Government Decree (717/2013) [18] considered the updated WENRA safety goals and IAEA SSR-2/1 requirements for the design of nuclear power plants. The requirements indicate systematically needed strength of safety demonstration and particularly underline the requirements for “practical elimination”. The Defence-in-Depth principle in the design of nuclear power plants is strengthened by emphasizing the consideration of extreme external conditions. The new criteria take also into account social and economical aspects in addition to the risks based on doses in case of severe accidents with releases. The consideration of the need for large scale protective measures for the public strengthens the earlier requirements related to restrictions on use of land and water. Government Decree (717/2013) Section 10 “Limits for accidents” stated that:

“*The release of radioactive substances arising from a severe accident shall not necessitate large scale protective measures for the public nor any long-term restrictions on the use of extensive areas of land and water.*

In order to restrict long-term effects the limit for the atmospheric release of cesium-137 is 100 terabecquerel (TBq). The possibility of exceeding the set limit shall be extremely small.

The possibility of a release requiring measures to protect the public in the early stages of the accident shall be extremely small."

At the beginning of year 2016 the structure of the Finnish legal framework was changed and STUK Regulations were issued. Those regulations included technical requirements earlier covered by the Government Decrees. However, some requirements such as dose constrains needed to be presented in higher level legislation and were incorporated into Nuclear Energy Decree (161/1988).As an example the above mentioned Section 10 "Limits for accidents" is now presented in the Nuclear Energy Decree Section 22b. Regulation STUK Y/1/2016 [19] on the Safety of a Nuclear Power Plant supersedes Government Decree (717/2013). The Regulation uses the phrase "the possibility of … shall be extremely low" in cases that can be considered to be treated under "practical elimination" concept, although the general wording does not necessary imply this to be strictly followed. Cases to be "practically eliminated" are introduced in YVL Guides in more detail.

In Regulations items with extremely low possibility include for reactor design

— criticality accidents (§10, 3. a) iii); (nuclear reactor's physical feedback characteristics (§ 11, 1))
— integrity of (major) pressure bearing components (§10, 3. b) i.);
— containment leaks as a consequence of reactor pressure vessel failure (§10, 3. c) iii);
— and in connection of fuel handling and storage
— criticality accidents (§12, 4);
— severe accidents (§12, 5).

In the YVL Guides published in 2013 the "practical elimination concept" is further elaborated from the point of design and operation of a nuclear power plant reactor. In Guide YVL B.1"Safety design of a nuclear power plant" [20] presents the plant level requirements. The requirements for probabilistic safety assessment in the safety demonstration are presented in Guide YVL A.7 "Probabilistic risk assessment and risk management of a nuclear power plant" [21]. The assessment of releases is presented in Guide YVL C.3 "Limitation and monitoring of radioactive releases from a nuclear facility" [22]. Detailed requirements related to safety assessment, design of reactor, primary circuit and containment as well as specific provisions such as in-service-inspection programme, operating limit and conditions and ageing management programme that are needed to ensure safety during the operation are presented in YVL Guides.

In Guide YVL B.1, paragraph 423 deals with the "practical elimination" of the events leading to releases and a need to protect the population in early stage of an accident. The list presented in Guide YVL B.1 to be considered in the design of the nuclear power plant is not exhaustive and the licensees are expected to analyse more extensive set of events relevant for the reactor type chosen. Guide YVL B.1, paragraph 424 states that "*Events to be practically eliminated shall be identified and analysed using methods based on deterministic analyses complemented by probabilistic risk assessments and expert assessments. Practical elimination cannot be based solely on compliance with a cut-off probabilistic value. Even if the probabilistic analysis suggests that the probability of an event is extremely low, all practicable measures shall be taken to reduce the risk. As an example events to be practically eliminated include:*

1) a rapid, uncontrolled increase of reactivity leading to a criticality accident or severe reactor accident;
2) a loss of coolant during an outage leading to reactor core uncovery;
3) a load jeopardising the integrity of the containment during a severe reactor accident (e.g. reactor pressure vessel breach at high pressure, hydrogen explosion, steam explosion, direct impact of molten reactor core on containment basemat or wall, uncontrolled containment pressure increase); and
4) a loss of cooling in the fuel storage resulting in severe damage to the spent nuclear fuel.

Guide YVL A.7 paragraph 306 states that "A nuclear power plant unit shall be designed in compliance with the principles set forth in Section 10 of Government Decree 717/2013 in a way that

a) the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq is less than $5 \cdot 10^{-7}$/year;
b) the accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency.

Release assessments shall take into account all of the nuclear fuel located at the plant unit. A spent nuclear fuel storage external to the plant unit is considered a separate nuclear facility for whose analysis the aforementioned criteria apply.

It should be noted that the definition of the Design Extension Conditions in the Finnish regulations and YVL Guides is used for the accident conditions at Defence-in-Depth level 3b and as a result of mitigation a safe stable state of a reactor without severe fuel damage needs to be achieved. The Defence-in-Depth levels and the associated accident conditions with acceptance criteria are shown in Fig. 1. ”Practical elimination” of early or large releases consist of two aspects: for the first consideration of Defence-in-Depth effectively in design and operation of a reactor and for the second accident sequences for which provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design. As shown in Fig. 1. at Defence-in-Depth Level 1, the rupture of the pressure vessel and severe criticality accidents are “practically eliminated”, and at Level 4 the severe accident sequences leading to the loss of containment are to be identified and “practically eliminated”. These sequences are specific to the reactor type and design. The general goal for “practically eliminate” large or early releases is driven by the probabilistic limit for large and early releases above. The overall methodology of “practical elimination” cannot be the same for large amount of sequences, but the reactor unit is to be considered as a whole including all relevant combinations of events. This overall approach for “practical elimination” of large and early releases is supported by the approach of “practically eliminating” specific sequences and conditions threatening the containment integrity in severe accidents.

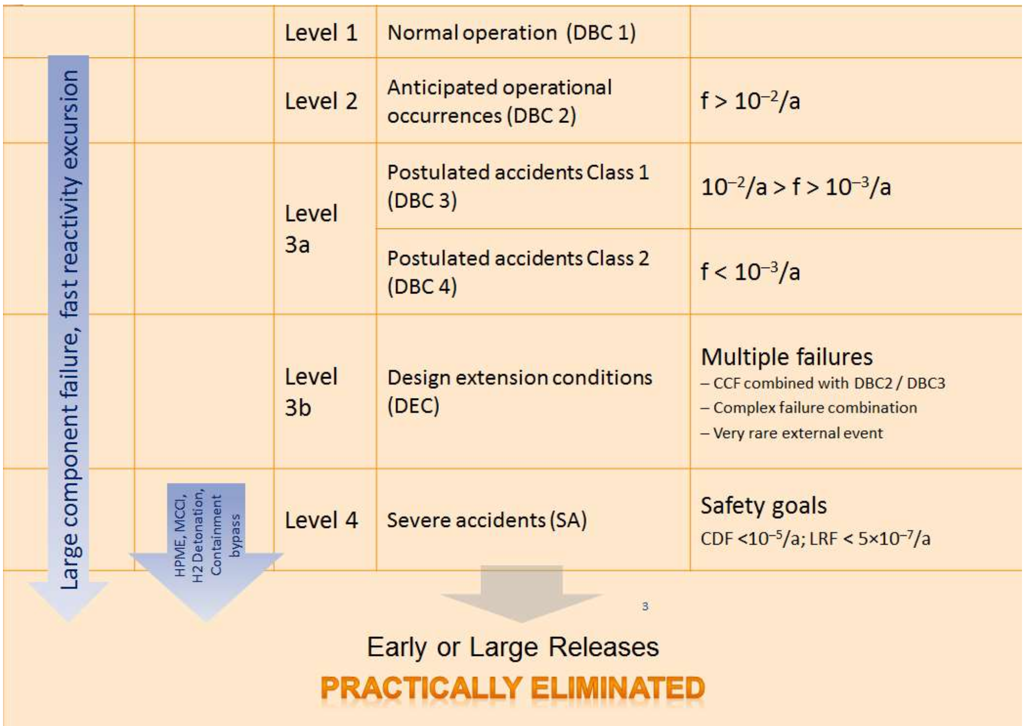

FIG. 1. The Defence-in-Depth levels, event categories, frequencies and relation to the phenomena to be practically eliminated in the Finnish context.

3. APPLICATION OF THE NEW REQUIREMENTS TO OPERATING NUCLEAR POWER PLANTS AND TO THE NUCLEAR POWER PLANT UNDER CONSTRUCTION

As the YVL Guides present the requirements for new reactors an implementation decision is made concerning the application of each new Guide YVL to operating nuclear facilities and facilities under construction. Before the decision the licensees are requested to evaluate their compliance with the guide. Based on the evaluation STUK may approve exemptions in the application of the guide. The implementation decision may also include requirements for improvement. In the judgement Section 7a of the Nuclear Energy Act is taken into account: The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified taking into account operating experience and safety research and advances in science and technology. The application of Section 7a of the Nuclear Energy Act and exemptions are explicitly described in building regulations.

The overview of the status of operating nuclear power plant and the plant under construction is given below. To limit the extensive topic for this paper an overview of "practical elimination" concept is made in context of Regulation STUK 1/Y/2016 § 10. Accident sequences for which provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design of a reactor are discussed in more detail in context of the operating nuclear power, a nuclear power plant under construction and a nuclear power plant to be built in Finland. The paper presents two examples of specific sequences to be considered in the concept of "practically eliminated" of early (or large) releases in different reactor types in Finland: for the first the prevention of the rupture of the reactor pressure vessel; and for the second the severe accident management in order to ensure containment integrity.

3.1. Loviisa Nuclear Power Plant units 1 and 2

The Loviisa plant comprises two PWR units (pressurised water reactors, of VVER type), operated by Fortum Power and Heat Oy (Fortum). The plant units were connected to the electrical grid in February 8, 1977 (Loviisa 1) and November 4, 1980 (Loviisa 2). The nominal thermal power of both of the Loviisa units is 1500 MW (109 % as compared to the original 1375 MW). The increase of the power level was licensed in 1998. The Operating Licences of the units are valid until the end of 2027 (unit 1) and 2030 (unit 2). According to the conditions of the licences, two periodic safety reviews are required to be carried out by the licensee (by the end of the year 2015 and 2023). The safety assessment of the first periodic safety review of current operating license period was completed by the licensee in 2015 and STUK gave its decision in early 2017.

Several plant changes have been carried out during Loviisa NPP plant lifetime. The most important projects since the plant commissioning have been modifications made for protection against fires, modifications based on the development of the PSA models, implementation of severe accident management systems, reactor power uprating, and construction of training simulator, interim storage for spent fuel and repository for reactor operational waste. More detailed discussion on the plant modernizations and the actions taken after the TEPCO Fukushima Daiichi accident can be found in reference [23] Appendix 2.

The chapter presents two examples of specific sequences: the prevention of the rupture of the reactor pressure vessel; and the severe accident management in order to ensure containment integrity.

The design of primary components of the Loviisa Nuclear Power Plant is mainly based on standards that were in force in the Soviet Union at that time. The design principles are similar to those in the ASME Boiler and Pressure Vessel Code, but the requirements are deficient with regard to brittle fracture. The design of the primary circuit is based on setting a sufficient safety factor for various types of failure mechanisms, such as general deformation (plastic instability), continuously increasing deformation during load variations, fatigue, brittle fracture and instability (compression stress). The defined design loads comprise normal operation, deviations from normal conditions and postulated accidents. Provisions have been made for a guillotine break of the reactor coolant pipe by constructing emergency restraints and jet shields.

The Guide YVL E.5 and the latest edition of the ASME Code, Section XI are used as the acceptance criteria for in-service inspection programs and procedures. All ultrasound, eddy current and visual inspection systems are in the qualification system. The in-service-inspections are continuously enhanced while one of most significant chances has been the commissioning of the qualification system. The inspections evoked by Belgian Doel 3 and Tihange 2 hydrogen flake finding showed no indication at either of the Loviisa Nuclear Power Plant reactor pressure vessels.

The Loviisa Nuclear Power Plant reactor pressure vessels have been subject to separate operating license till the end of years 2027 and 2030. Heat treatment (annealing) was applied to the embrittled weld in the reactor pressure vessel of the Loviisa 1 in 1996. Such treatment has not yet needed to be carried out for the reactor pressure vessel of Loviisa 2. STUK decision on the periodic safety review in February 2017 raised concern on embrittlement margins of Loviisa 2 nuclear power plant unit reactor pressure vessel towards the end of the operating license. There is need for assessment of the limiting case like an inadvertent start of containment spraying and changes to the water temperature at the start of spraying. The updating of the probabilistic safety assessment of the reactor pressure vessel is ongoing.

A comprehensive severe accident management (SAM) strategy has been developed for Loviisa NPP to mitigate the consequences of severe accidents. This strategy is based on SAM safety functions whose purpose is to ensure containment integrity and isolation. To ensure containment integrity, energetic events like direct containment heating, steam explosions and hydrogen detonations must be "practically eliminated". Direct containment heating and steam explosions can be prevented by ensuring in-vessel retention of corium by reactor pressure vessel external cooling and primary circuit depressurisation. Hydrogen management is a separate safety function. Mitigation of slow processes, like prevention of a slow containment over-pressurisation has to be ensured as well.

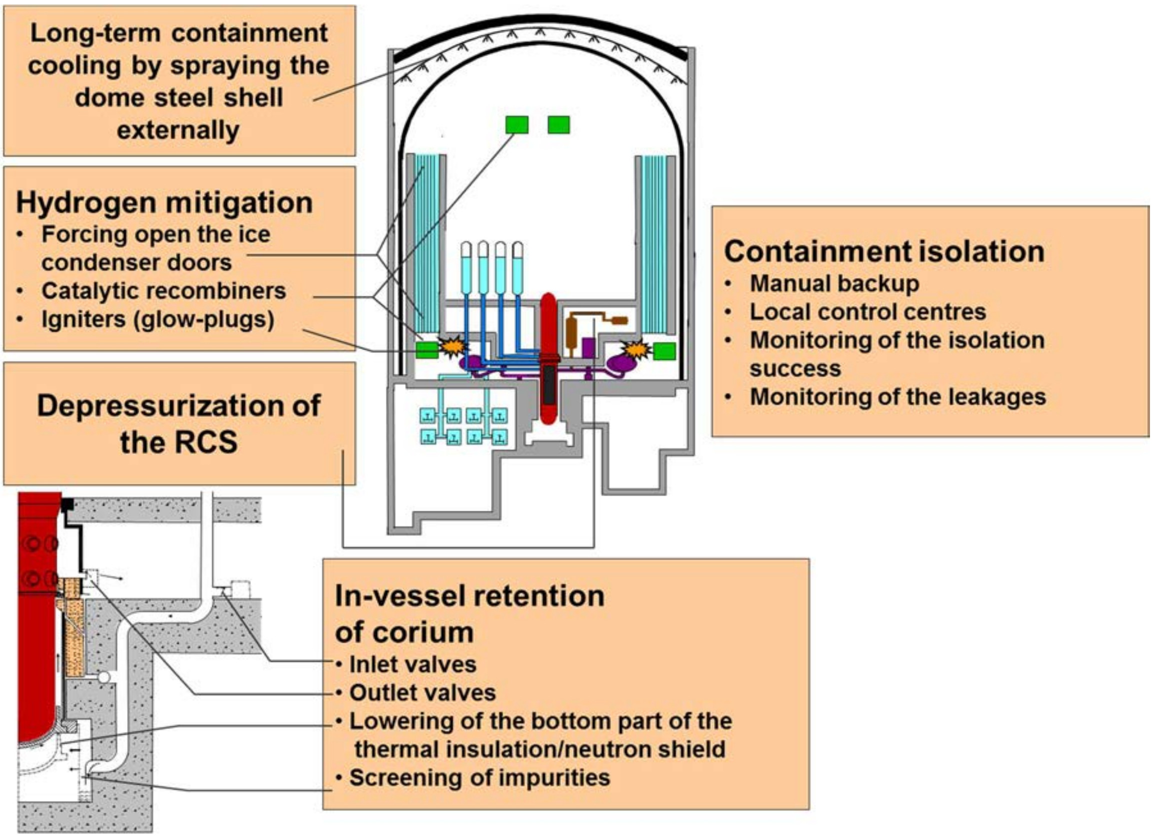

FIG. 2. SAM safety functions and the SAM systems of the Loviisa NPP.

Independent SAM systems were implemented at the plant mainly in 1990s, and the last installations were carried out in 2002. The SAM systems have independent power supply with dedicated diesel generators. Containment isolation is ensured with manual backups and additional local control centres to close the containment external isolation valves with SAM power supply. For RCS depressurisation two parallel depressurisation lines with two depressurisation valves in each of these have been installed. The water for cooling the RPV externally flows passively into the reactor cavity, and to ensure adequate flow paths, modifications to the structures were made. In addition, a hydraulic system to lower the thermal shield of the RPV lower head has been installed, and to avoid blockages also filtering of impurities in the water is applied with metallic screens on the cavity walls. Hydrogen management relies on efficient mixing of the containment atmosphere by forcing open the ice-condenser doors, and on passive hydrogen removal with catalytic recombiners. The glow plugs installed in the lower compartment of the containment ensure controlled hydrogen burn in case of fast hydrogen production.

Long-term heat removal from the containment and prevention of the containment overpressurization is managed by containment external spray system. A filtered venting system was not seen feasible for overpressure protection of the Loviisa NPP containments, as releasing non-condensable gases threatens the containment integrity in a situation where rapid steam condensation inside the containment could result in collapse of the steel shell due to its vulnerability to subatmospheric pressures. Base material attack by the molten corium is ruled out due to the in-vessel retention concept. Furthermore, pH control of the sump water is passively maintained well above 7 due to borax in the ice in the ice condensers.

SAM Procedures are based on the above SAM safety functions. Immediate SAM measures are carried out within the Emergency Operation Procedures (EOP). After carrying out immediate actions successfully, the operators concentrate on monitoring the SAM safety functions. The SAM guidance focuses on monitoring the leak tightness of the containment barrier, and on the long-term issues. The transition to SAM Procedures takes place when the reactor core is damaged or is close to doing so, and there is no return to EOPs thereafter. All operations are carried out by main control room (MCR) operators with the controls in the MCR. There is also a separate SAM control room with the same capabilities of SAM actions as the MCR, but with better protection against external radiation. The SAM actions can be backed up with local operations, in case control room operations are not functioning. To support SAM actions and monitoring, dedicated SAM measurements are implemented to monitor the successfulness of the needed actions.

3.2. Olkiluoto Nuclear Power Plant units 1 and 2

The Olkiluoto plant comprises of two BWR units that are operated by Teollisuuden Voima Oyj (TVO). The plant units were connected to the electrical network in September 2, 1978 (Olkiluoto 1) and February 18, 1980 (Olkiluoto 2). The present nominal thermal power of both Olkiluoto units is 2500 MW, which was licensed in 1998. The new power level is 115.7% as compared to the earlier nominal power 2160 MW licensed in 1983. The original power level of both units was 2000 MW. The Operating Licences of the units are valid until the end of 2018. According to the conditions of the licenses, the licensee carried out a periodic safety review and submitted it to the regulator at the end of 2008. The application for the operating license till the end of 2038 is under regulatory review at moment.

Several plant changes have been carried out during Olkiluoto NPP plant lifetime. The most important projects since the plant commissioning have been two reactor upratings, implementation of severe accident management systems, modifications based on the development of the PSA models, construction of training simulator, interim storage for spent fuel and repository for operational waste, and R&D program for disposal of spent fuel. The first power uprating project was carried out in 1983-1984. Thermal power was uprated from 2000 MW to 2160 MW (8%). The plant modifications included for example a new relief valve that was installed in the reactor primary system, changes in the reactor protection system, and increase of cooling capacity of some heat exchangers. More detailed discussion on the plant modernizations and the actions taken after the TEPCO Fukushima Daiichi accident can be found in reference [23] Appendix 3.

The chapter presents two examples of specific sequences: the prevention of the rupture of the reactor pressure vessel; and the severe accident management in order to ensure containment integrity.

The primary circuits of the Olkiluoto 1 and 2 NPP units include the reactor pressure vessel and its associated internal main cooling pumps and heat exchangers as well as the pipelines and fittings associated with the reactor pressure vessel up to the outer containment isolation valves. These components were designed and manufactured in accordance with the requirements of standards applicable to Safety Class 1 pressure vessels. The main standard applied was the US standard ASME Boiler and Pressure Vessel Code. Provisions have been made for ruptures of the reactor coolant pipe by providing anti-rupture supports and by carrying out the necessary dynamic analyses.

The Guide YVL E.5 and the latest edition of the ASME Code, Section XI are used as the acceptance criteria for in-service inspection programs and procedures. All ultrasound, eddy current and visual inspection systems are in the qualification system. The in-service-inspections are continuously enhanced while one of most significant chances has been the commissioning of the qualification system.

STUK safety assessment of the operating license application is ongoing. The plant will reach 60 year age at the end of the applied operating license period. One of the key questions is the adequacy of the safety margins of the reactor pressure vessel also at the end or this period.

The SAM systems were implemented in Olkiluoto 1 & 2 in 1980's, and some additional modifications were made in 1990s. The aim of these is to avoid energetic events, protect containment from overpressurization, ensure long-term containment cooling and minimize radioactive releases in course of a severe accident.

The original pressure relief system of the RCS has been modified to ensure depressurization in severe accidents with the objective to depressurize the primary system prior to the RPV breach. In addition to automatic actuation, actions for manual depressurization are included in EOPs. The valves to be applied for depressurization in severe accidents have been modified to fail safe in open position in case of loss of electricity, and the valves are kept open at low pressures hydraulically with nitrogen or water supplied from outside of the containment.

To guarantee the corium cooling after the RPV breach due to the core melt, the lower drywell is flooded prior to the RPV failure. Flooding prevents contact of hot debris with drywell wall penetrations, and the penetrations have been shielded as a plant modification. Flooding takes place passively after opening two pipelines with two valves in each of these to enable water flow from the wetwell into lower drywell. Opening of the valves is done manually either from the MCR or locally. Flooding of the drywell results in a 9 m deep water pool, and the melt discharged from the RPV is expected to solidify and fragment into particles when it falls through the pool. This, however, creates a possibility of a energetic melt-coolant interaction, and thus plant modifications have been made to strengthen the containment (the personnel access lock) against steam explosions.

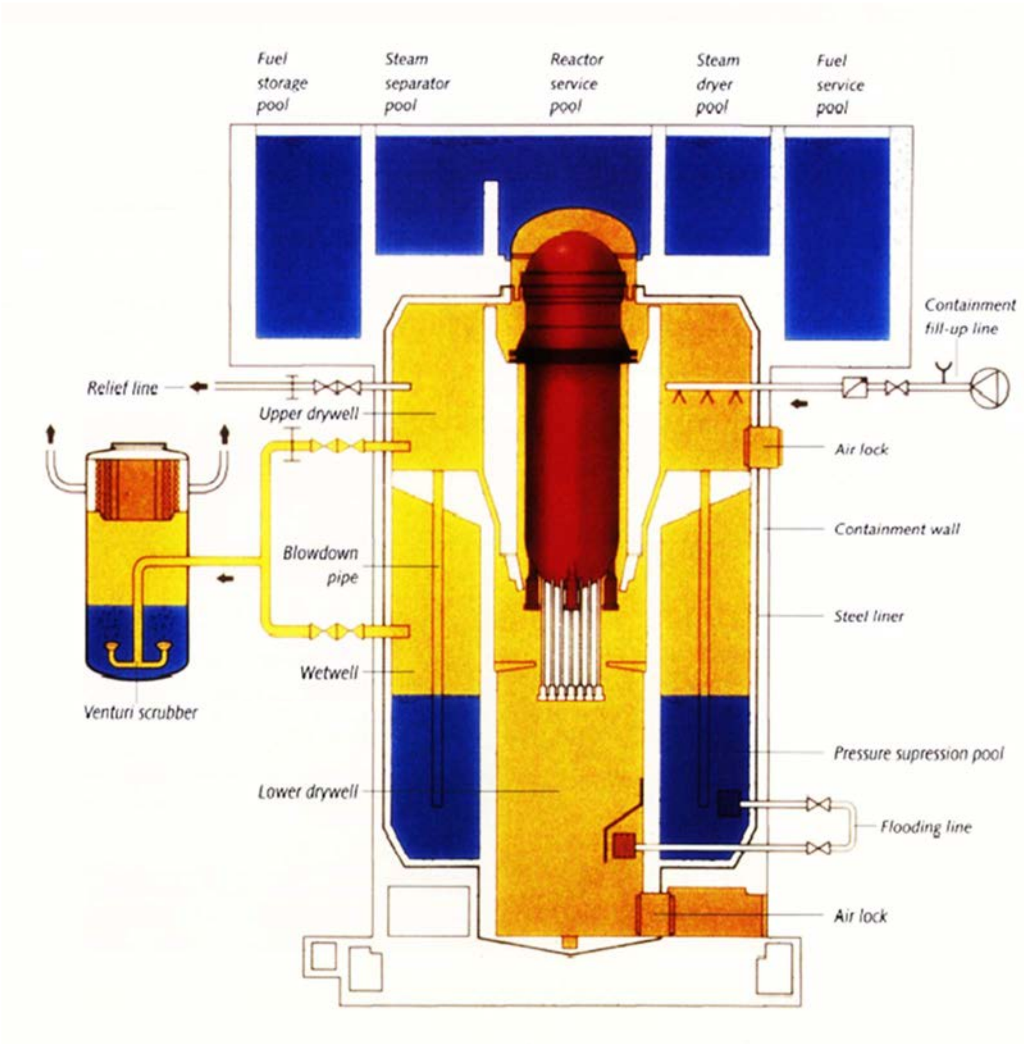

FIG. 3. SAM safety functions and the SAM systems of the Olkiluoto NPP units 1&2.

The containment can be further filled with water from an external source (fire water) to the level of the core top, and this ensures cooling of the core debris possibly remaining in the RPV. Filling of the containment takes more than 24 hours, and as the filling continues, compression of non-condensable gases will at some stage necessitate controlled containment venting.

Installation of the containment filtered venting system (FCVS) at both units was the largest plant modification for severe accidents in Olkiluoto 1 & 2. The FCVS is a venturi scrubber filter with an integrated metal-fibre filter on top of the scrubber pool. Water in the pool is alkaline to control gaseous iodine production. Venting actuates passively by bursting of the rupture disc at the drywell venting line when the containment pressure exceeds 0.55 MPa. The vent line can be closed by manually operated valves from the reactor building at a location providing adequate shielding against radiation from the radioactive material in the FCVS. The system is initially filled with nitrogen to prevent hydrogen burns or explosions, and no water addition is needed for the first 24 hours of operation of the system. From the inlet nozzles the gas enters the scrubber pool, and then flows through a filter of stainless steel wool removing droplets. The exhaust gas is released from the ventilation stack separate for each of the reactor units.

Retention of iodine inside the containment has been improved with a containment pH control system for severe accidents installed in 2001. The system is common for both units providing NaOH solution injection into the containment via the containment spray system.

Furthermore, specific containment instrumentation for severe accidents has been installed to enable monitoring pressure, temperature, water level, and the containment and the stack off-gas dose rate, and carry out sampling of the FCVS during severe accidents. The severe accident instrumentation is independent of other instrumentation, and battery back-up is able to supply power during the first 24 hours without any auxiliary power.

The SAM Procedures for the MCR operators provide for instructions to carry out all above actions needed during severe accidents. Transition to the SAM Procedures takes place when the actions in other EOPs fail to restore a safe plant state. The systems needed in severe accidents are single failure tolerant, with the exception of the FCVS vessel. The phenomena threatening the containment integrity, especially the possibility of a steam explosion exceeding the limits of the containment strength and core debris coolability in the bottom of the flooded drywell, have been analysed widely with support of national and international experimental activities. At the moment, the SAM approach has been considered acceptable by STUK, but the possible new information is taken into account when evaluating the adequacy of the provisions against severe accident phenomena.

3.3. Olkiluoto Nuclear Power Plant units 3

Decision-in-Principle procedure was carried out during the period November 2000 – May 2002 when Teollisuuden Voima Oyj (TVO) applied for a Decision-in-Principle for the fifth NPP unit in Finland and the Government approved it and the Parliament confirmed the approval. Construction license application for the Olkiluoto unit 3 was submitted by TVO to the Ministry of Trade and Industry (predecessor of the Ministry of Employment and the Economy) in January 2004.

The new unit, Olkiluoto 3 is a 1600 MWe European Pressurised Water Reactor (EPR), the design of which is based on the French N4 and German Konvoi type PWR's. A turn key delivery is provided by the Consortium Areva NP and Siemens. The technical requirements for Olkiluoto unit 3 were specified by using the European Utility Requirements (EUR) document as a reference. TVO's specifications complemented the EUR mainly in those points where Finnish requirements are more stringent. STUK gave its statement and safety assessment in January 2005 based on the review of the licensing documentation and the Government issued the Construction License in February 2005.

Construction phase of the Olkiluoto unit 3 still continues. In the turbine island, commissioning tests of the systems have been completed as far as possible without connection to reactor island. The turbine island is under preservation. In the nuclear island, the finishing of installation is going on and the commissioning tests have been started. Next major licensing step is the operating license. TVO submitted operating license application to the Ministry of Employment and the Economy in April 2016. Operating License is needed prior to loading nuclear fuel into the reactor core.

Following the accident at the TEPCO Fukushima Daiichi nuclear power plant on the 11th of March in 2011, safety assessment of Olkiluoto unit 3 was initiated. The topics included the preparedness against loss of electric power supply, loss of ultimate heat sink and extreme natural phenomena. As being a unit under construction, any immediate actions were not necessary, but STUK required the licensee to carry out additional assessment and present an action plan for safety improvements. Assessment was conducted and reported by the licensee to STUK on 15 December 2011. STUK reviewed the assessment and made decision on 19 July 2012 on the suggested safety improvements and additional analyses. More detailed discussion on the plant the actions taken after the TEPCO Fukushima Daiichi accident can be found in reference [23] Appendix 4.

The chapter presents two examples of specific sequences: the prevention of the rupture of the reactor pressure vessel; and the severe accident management in order to ensure containment integrity.

At the moment STUK reviews the operating license application Olkiluoto unit 3. The French RCC-M standard applicable to the design of nuclear facilities are used. The design criteria presented in the said standard are based on the ASME Boiler and Pressure Vessel Code, Section III, NB, Class 1 Components, Rules for Construction of Nuclear Power Plant Components (American Society of Mechanical Engineers), to which reference is also made in the Finnish YVL Guides. Provisions for primary circuit pipe breaks have been made in the design of the Olkiluoto 3 Nuclear Power Plant Unit. In the design of the Olkiluoto 3 Nuclear Power Plant Unit maintaining adequate safety margins throughout the contemplated 60-year service life of the facility is considered. The reactor pressure vessel is designed and manufactured in such as way as its rupture can be regarded as highly improbable. The plans for the in-service-inspections have been develop for Olkiluoto 3 Nuclear Power Plant Unit in line with YVL Guides.

With regards to severe accidents, the big difference between Olkiluoto 3 and the operating reactors in Finland is that SAM provisions have been integrated already in the basic design of the EPR, whereas for the operating reactors the effective measures have required wide plant modifications as described above. The main principles of the SAM at Olkiluoto 3 are rather similar to those in operating reactors in Finland: RCS depressurization, core melt cooling, containment heat removal, hydrogen management, and the containment filtered venting with dedicated SAM monitoring systems to support these functions.

RCS depressurization is carried out with two redundant dedicated lines with the aim of practically eliminating core melting and the RPV failure at high RCS pressures. The melt stabilization is based on melt retention in the reactor pit below the RPV lower head before allowing it to be spread into the specific cooling region. During the retention phase the melt mixes up with additional sacrificial material, and the properties of the melt become more favourable for spreading and cooling. The melt is spread into a specific spreading area, where its surface-to-volume ratio becomes larger in order to enhance cooling. In this area, the melt is cooled by the water flowing passively from the in-containment tank under the floor of the spreading area cooling the melt from below and further flooding the spreading area from the top ensuring efficient melt cooling.

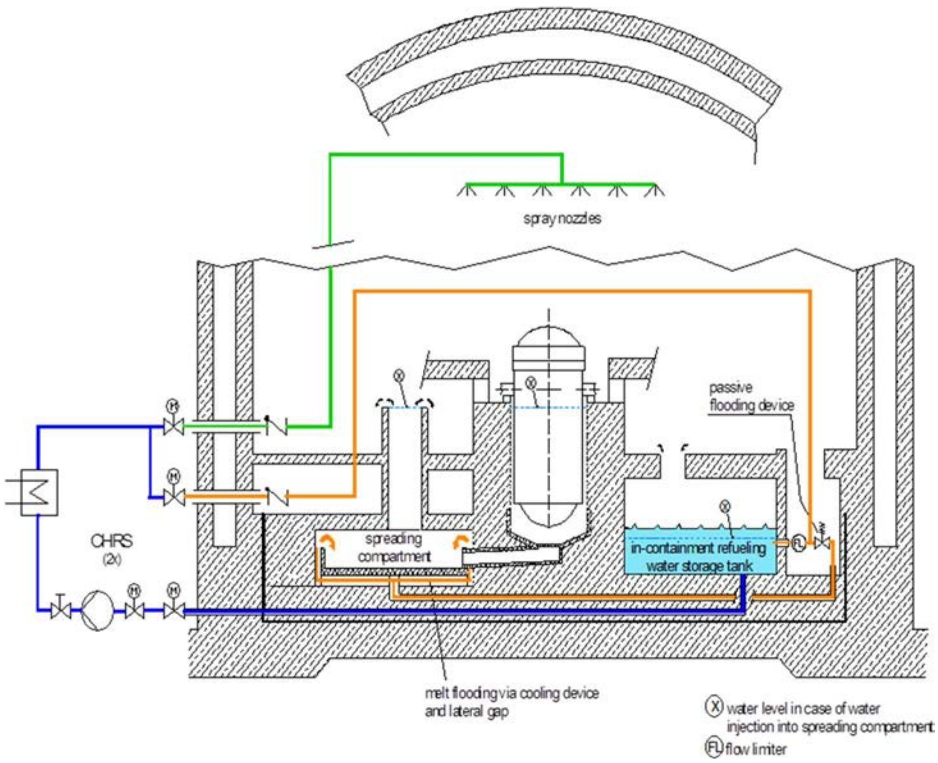

FIG. 4. SAM safety functions and the SAM systems of the Olkiluoto NPP units 3.

The heat from the containment is removed through the heat exchangers of the dedicated containment heat removal system that provides for containment spraying and recirculation of the water cooling the core melt inside the containment. Hydrogen removal from the containment is carried out by passive autocatalytic recombiners.

Olkiluoto 3 is equipped with a similar FCVS as those at Olkiluoto 1 & 2. At the time when Olkiluoto 3 was licensed in Finland, there still was an explicit requirement for FCVS in force. Although the SAM strategy of an EPR does not necessarily need the FCVS, STUK considered that it has to be installed at a new reactor design to be licensed. Nevertheless, the system ensures a controlled way to decrease the containment pressure in the long-term. Furthermore, it is used as an diverse path to release heat from the containment in specific accident scenarios during refuelling outages, in case other means of heat removal have failed.

No specific pH control for the containment has been considered necessary to be included in the SAM provisions of Olkiluoto 3. As the releases due to the SAM measures remain well below the limits set for severe accidents, additional reduction has not been required, in order to avoid possible drawbacks, e.g. containment sump clogging, in implementing pH additives into the cooling water inside the containment.

4. APPLICATION OF THE NEW REQUIREMENTS TO NEW BUILD

Fennovoima received positive Decision in Principle (DiP) in 2010 to construct a new nuclear reactor in Finland. Fennovoima is a new company established in June 2007 and it made a contract with Rosatom (VVER-AES2006) in December 2013. Plant design AES-2006 was not included in the original Decision in Principle, and therefore Fennovoima applied a complementary Decision in Principle in spring 2014. STUK submitted its statement with the preliminary safety assessment to the Ministry of Economic Affairs and Employment in May 2014 stating that the AES-2006 plant can be constructed to fulfil the Finnish safety requirements at the site Hanhikivi at Pyhäjoki. However, some changes are required to the plant design in order to meet Finnish safety requirements. Such issues are the measures against a large civil aircraft crash, internal flooding and fires, as well as severe accident management. STUK identified other technical details that require additional analyses or tests during the construction license phase. Fennovoima filed a construction licence application to the Ministry on 30th June 2015.

The AES-2006 is based on the VVER 91/99 plant, which is developed from the operating VVER-1000 plants. Plants of the VVER type have been constructed in Russia and many other countries for more than 30 years. The Loviisa 1 and 2 plant units are based on the VVER-440 plant type. A reference plant for Fennovoima is the Leningrad NPP-2, which is currently under construction in Russia. The Leningrad NPP-2 comprises two plant units, which, together with the Novovoronesh-2 plant unit (AES-2006/V392M), are the first AES-2006 type plants in Russia. The construction of the Leningrad NPP-2 was started in 2008. The design service life of the plant is 60 years.

The review of the construction license application is ongoing at STUK. However, during the Decision in Principle phase the following conclusions were made by STUK concerning the reactor pressure vessel and severe accident management.

- The design objectives and design principles of the main nuclear components of the AES-2006 are mainly in compliance with the Finnish safety requirements. The effects that the reactor pressure vessel material and especially its nickel alloys and impurities have on radiation embrittlement and the radiation embrittlement rate requires additional clarifications that shall be described in the construction licence application.
- STUK finds that it is possible to implement the systems and strategy for managing severe accidents in compliance with the Finnish safety requirements. The implementation of the functions for severe accidents by independent systems in compliance with the Finnish requirements shall be verified in the construction licence phase.

5. CONCLUSIONS

The ”practical elimination” of early or large releases consist of two aspects: for the first consideration of Defence-in-Depth effectively in design and operation of a reactor and for the second accident sequences for which provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design.

The idea of practically eliminate rupture of the reactor pressure vessel and fast increase of reactivity threatening the integrity of the containment and causing early or large releases have been in the design of nuclear power plants from the 1970s. Operating experience, research and development have shown the complexity of safety issues to be considered at different Defence-in-Depth levels in this context. The consideration of severe accidents has been implemented in the Finnish regulations in 1980s. As all the reasonably practicable modernizations need to be implemented into operating nuclear power plants in Finland severe accident management systems have been implemented to operating power plants. The consideration of conditions arising during severe accident needs to be assessed as part of demonstration to practically eliminate early or large releases. The phenomena to be assessed depend on the reactor type and the severe accident management strategy. The plant in different licensing phase can be adapted to new requirements. The practical elimination concept should be clearly described in the plant safety analysis report.

The Finnish approach for practical elimination of large or early releases is driven by limiting the overall frequency of accidents that could lead to such a situation. Specific approach for practical elimination is addressing sequences, for which mitigation is not feasible due to the nature of the phenomena resulting from some specific initiating events. The possibility of occurrence of such sequences has to remain extremely low, which supports also achieving the probabilistic overall safety goal.

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PRACTICAL ELIMINATION: ORIGIN, FOCUS AND IMPLEMENTATION IN THE SAFETY DEMONSTRATION

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Abstract

Severe Accident mitigation and prevention of large radioactive releases have been playing an increasing role in the safety philosophy and demonstration. This attention is significantly reinforced globally since the Fukushima Daichi accident.

Within the safety demonstration as regards severe accidents, the "practical elimination" approach appears of great importance. In order to contribute to the building-up of a common understanding and practice, this paper addresses some questions raised by this matter: (1) purpose and scope of the practical elimination at its inception: avoid with the greatest confidence some of the fast and energetic physical phenomenon which can occur in case of a severe accident and might jeopardize the containment; (2) respective role of the deterministic and probabilistic approaches; (3) the progressive generalization of the practical elimination objective from the prevention of large and early releases towards the large or early releases. The paper presents some examples of the practical elimination demonstration. It discusses the relevance and drawbacks of the practical elimination concept generalization as regards delayed large releases, as the basemat ablation by the corium. It concludes with a view on the specific role and focus of the practical elimination demonstration in the layered approach of defence in depth and safety demonstration.

1. INTRODUCTION

The "practical elimination" expression was first formulated in a joint meeting between the French Groupe Permanent and German experts from RSK, in the beginning of the 1990s. This meeting was part of an extensive work between the German and French safety authorities, dedicated to the definition of the safety objectives for a new generation of nuclear reactors[1].

2. PURPOSE AND SCOPE OF THE PRACTICAL ELIMINATION AT ITS INCEPTION

The "practical elimination" was defined as part of the safety objectives addressing severe accident provisions. Indeed, in the aftermaths of the Three Mile Island and Chernobyl accidents, a major step was decided in terms of Defence in Depth: in addition to the expected progress in the field of prevention, provisions should be included in the design in order to be able to mitigate a severe accident, should it occur.

The severe accident objective was therefore two folds: (1) the severe accident situations which would lead to large early releases should be practically eliminated; (2) the low pressure core melt should be mitigated in order that the maximum anticipated releases would require only protective measures towards the populations that remain limited in time and space.

The situations to be practically eliminated because they could lead to early and large releases were clearly targeted as involving fast and energetic physical phenomenon, likely to occur in the course of a severe accident, which might jeopardize the containment in a rapid and non-recoverable way and for which no reasonably achievable mitigation design solutions were foreseeable.

In practice, as regards the next generation of PWRs which was contemplated, the list of such situations was limited and well identified: high pressure core-melt (potentially leading to Direct Containment Heating), large scale hydrogen detonation, steam explosions, large reactivity accidents (for example resulting from the rapid injection of a non-borated or cold water slug into the core).

The course of the R&D and the progress of knowledge later allowed focusing the steam explosion problematics, in terms of practical elimination, on the ex-vessel steam explosions.

[1] This new generation of reactors was later called EPR (the model) or Gen 3 / Gen 3+ (the generic names).

3. DETERMINISTIC AND PROBABILISTIC APPROACHES IN THE PRACTICAL ELIMINATION

Practical elimination meant from its inception that the given situations, if they could not be considered as physically impossible, should lead to specific design provisions in order to exclude their occurrence with high confidence.

Although the intent was clear, the wording was not very easy to write and even somehow laborious; it ended, in French, with “elimination pratique”, translated as “practical elimination”. If “elimination” owns a clear and unambiguous meaning, especially in a post Chernobyl context, the word “pratique” or “practical” deserves some more explanation, as it is not necessarily intuitive and might even show some ambiguity.

Although one meaning of “practically” could be “almost”, which might be interpreted as a target somehow loose, the objective here was clearly to seek a very high level of confidence and certainty, which is on purpose embedded in the word “elimination”.

“Practical”, on its side, intended to refer to the physical world, which is the practical world, as opposed to the probabilistic dimension. It means that adequate, strong and very convincing engineering features were requested to achieve the objective.

In particular, for the phenomenon targeted, it was stated that the practical elimination demonstration cannot be based on a general probabilistic cut-off frequency (as considering a very low core melt frequency). Engineering provisions are in priority to be sought and a deterministic demonstration is requested. Their objective is, first, to endeavour to make the given physical phenomenon physically impossible by design, second, if not physically impossible, highly unlikely with a very strong confidence thanks to dedicated engineered design provisions, R&D and physics.

Probabilistic studies are to be used as a complement and verification. In practice, the balance in the roles of deterministic and probabilistic approaches may depend on the phenomenon and the provisions involved.

Importantly, the ways to ensure the practical elimination demonstration are established on a case by case basis.

4. SOME EXAMPLES OF SAFETY DEMONSTRATIONS APPROACHES

Here are the main lines of the EPR safety provisions and demonstrations as regards practical elimination.

4.1. High pressure core melt and DCH

Discharge lines dedicated to release the pressure in case of a severe accident are implemented on the pressurizer. They are qualified to severe accident conditions, their opening is clearly stated as the first action to be taken by the operators when entering severe accident management guidelines and the criteria to do so are simple and non-ambiguous: core temperature above a threshold or reactor building dose rate above a threshold. The deterministic safety case is supported by the sizing, the design, the qualification and the actuation of the discharge valves. The severe accident support studies prove that there is sufficient time for the operator to perform this action with a high confidence.

4.2. Hydrogen detonation

The principle is to show that at no time in the course of a severe accident the reactor building atmosphere's composition would be in the Schapiro diagram zone where a detonation could occur. This involves a comprehensive design approach including the mixing of the various gazes in the reactor building (with dedicated openings between the compartments and the building), hydrogen passive catalytic recombiners, complete modelling of the atmosphere composition along the course of the accident.

4.3. Steam explosion

The conviction that in-vessel steam explosions would not jeopardize the containment was achieved in the early 2000s, thanks to extensive R&D.

On the other hand, R&D could not, so far, guarantee that ex-vessel steam explosions would not jeopardize the containment if molten core breaching the vessel were pouring in a pit filled with water. This leads to design a vessel pit remaining dry during accidents, to make a large steam explosion physically impossible.

4.4. Large reactivity injection in the core

As regards the potential of injecting a large slug of non-borated / cold water in the core, dedicated provisions were taken by design, as a safety class bore meter on the injection lines and automatic stopping, on dedicated signals, of any dilution ongoing from the volumetric control systems. These provisions and the related safety cases are deterministic. In addition, as dilution could originate from diverse scenarios, the probabilistic studies play a greater role, as a powerful tool to address multiple and complex sequences. Finally, studies and modelling have been continuously progressing as regards the potential for inherent dilutions in LOCA conditions and the mobilization of consequently inhomogeneous primary water under natural circulation.

5. "LARGE AND EARLY RELEASES" OR "LARGE OR EARLY RELEASES"?

Several texts now refer to the practical elimination of large **or** early releases rather than the original focus on large **and** early releases. It is not sure that the reasons of this wording evolution have been comprehensively and consciously stated in terms of the safety philosophy and the role of the concept in the safety demonstration.

Obviously, this change reflects a very understandable will to consider with the greatest seriousness the potential consequences of severe accidents, to acknowledge their hardly bearable consequences, to loudly and publicly state that those should be avoided by all means. "Practical elimination" being the strongest words in the safety glossary, it tends to be used to express the objective to suppress all massive radioactive release scenarios.

However, such an objective seems well captured by the Vienna declaration: "Nuclear power plants should be designed, constructed and operated with the objectives of preventing accidents and, **should an accident occur, mitigating its effects and avoiding off-site contamination**[2]" .

It seems also well captured by the second part of the objective stated by the technical guidelines for the design and construction of a next generation of PWR NPPs, adopted by the French Groupe Permanent and RSK experts in 2000, as well as by WENRA: "for accident with core melt that have not been practically eliminated, **design provisions have to be taken so that only limited protective measures in area and time are needed for the public** (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures".

All in all, even without the expression "practical elimination", a major safety objective arose, spread and has been regularly stated in the last 25 years: that severe accident mitigation is to be ensured through the design and the accident management provisions, so that the confinement is maintained and large releases are prevented, should a severe accident occur. Such an objective requires, on new reactors, a dedicated line in the defence in depth, specific equipment and an appropriate safety demonstration, firstly based on deterministic studies. It seems that such an objective and its subsequent consequences in terms of engineering, safety demonstration and operations may suffice to ensure the seriousness to be placed on this matter.

For example, it can lead to the implementation of a core catcher and of a dedicated system to remove the residual heat outside the containment in case of a severe accident, based on extensive R&D, adequate engineering provision and a full deterministic safety demonstration.

On the other hand, the practical elimination concept seems indispensable for the phenomenon which would lead to massive release without any possible or reasonably practicable means of mitigation: avoiding massive releases fully relies, in this case, on prevention. Practical elimination seems all the more indispensable for such phenomenon than the corresponding releases might occur in a rapid manner.

In other words, focusing the practical elimination concept on the large and early releases offers the great advantage to lead to identify and to pay a specific attention to the very situations and phenomenon which, among

[2] Being there understood that off-site contamination relates here to large releases which would have long term consequences over large portions of territory, this declaration having been elaborated as a consequence of the Fukushima Daiichi accident)

the diverse phenomenon and scenarios regarding severe accident, bear the highest risks. It therefore allows an adequate focus on the limited number of these well identified situations and phenomenon. It brings an additional level of hierarchy and structure in the safety philosophy, which seems a benefit in terms of safety.

6. CONCLUSION

Historically, the concept of practical elimination focused on the fast and very energetic physical phenomenon which can occur in the course of a severe accident and could jeopardize the containment in a sudden and non-recoverable way[3]. It concerns a limited number of situation and phenomenon for which no mitigation seems possible or practically achievable. For those, the only way of avoiding large releases, which in addition could occur in a rapid manner, is to prevent the situation or the phenomenon to occur. The consequences at stake justify and require a specific approach. In particular, it is requested that the practical elimination is grounded on engineered solutions making the given phenomenon either physically impossible or very unlikely with a high confidence, thanks to the robustness of the engineered features and a deterministic safety demonstration.

Keeping this focus would, for the author, makes a lot of sense in terms of safety philosophy and structure of the safety demonstration. It gives a sense of hierarchy in the severe accident situations: the most challenging severe accident phenomenon should be well identified and specifically addressed. And it provides an adequate framework for the two categories of the severe accident situations to address in the design, operation and safety demonstration of the Nuclear Power Plants in order to avoid, should a severe accident occur, large radioactive releases and consequences which would not be limited in time and space: (1) the phenomenon which would jeopardize the containment and cannot be mitigated have to be practically eliminated (all the more than they can show rapid kinetics, leaving insufficient time to take protective measures towards the populations) (2) the other phenomenon have to be mitigated.

[3] This approach, focused on the reactor, was later extended to the spent fuel pool, where severe core degradation is to be practically eliminated, as means of mitigation would appear not practicable and reasonable.

SUMMARY AND CONCLUSIONS FROM THE INTERNATIONAL SEMINAR ON IN-VESSEL RETENTION STRATEGY

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Abstract

On the 6-7 June 2016, IRSN hosted an international workshop about the "Strategy of In-Vessel Melt Retention: Knowledge and Perspectives". The workshop was co-organized by JRC and IRSN, with the sponsorship of ETSON. With panel discussions and technical sessions, the workshop covered all the important issues related to in-vessel corium retention, from the physical understanding to regulatory frames. The major points of the safety demonstration were discussed. Some industrial aspects were also addressed. One of the objectives was to provide an orientation of R&D projects to strengthen IVR strategies, such as the H2020 IVMR project, coordinated by IRSN. The current approach followed by most experts for IVR is a compromise between a deterministic approach using the significant knowledge gained during the last two decades and a probabilistic approach to take into account large uncertainties due to lack of data for some phenomena and due to excessive simplifications of models. It was concluded that a harmonization of the positions of safety authorities on the IVR strategy is necessary to allow decision making based on scientific knowledge. For this, a consensus on several issues should be reached between R&D experts. This includes in particular the issues of the transient evolution of oxide and metal layers in the lower plenum and of the long term mechanical behavior of the thin "cold shell" resulting from vessel ablation.

1. INTRODUCTION

Gen III nuclear reactors are characterized by the consideration, at the design stage, of severe accidents with core melting and, therefore, the retention of corium in the containment. To reach this goal, the options selected for the design of those reactors may be divided into two different approaches. One approach is the stabilization of corium outside the vessel (EPR or some VVER cases, for example), the other aims at keeping the corium inside the vessel (IVR strategy). Thanks to panel discussion and technical sessions, the workshop covered all the important issues related to strategies of in-vessel corium retention, from the physical understanding to the safety demonstration and regulatory frames. Experts' presentations provided an overview of the current knowledge and understanding about the physical processes associated with corium and its interaction with the reactor vessel. The major points of the safety demonstration were pointed out and discussed. Some industrial aspects were also addressed and discussed by reactor designers.

One of the objectives of the workshop was to provide a basis for the orientation of new research projects such as the European project IVMR. This project started in 2015 with duration of 4 years and is coordinated by IRSN. It aims at developing knowledge and tools to estimate the efficiency of the measures dedicated to stabilize corium in the reactor vessel in case of severe accident with core melting in reactors of 1000MWe or more.

The workshop gathered more than 130 participants from most of the main actors involved in issues related to severe accident management in order to share knowledge and positions:

- Safety authorities: ASN – France, US NRC – USA, NSC – China, NRA – Japan, SEC-NRS – Russia, KINS – South Korea, STUK – Finland, ONR – United Kingdom, CNSC – Canada, SSM – Sweden, SNNI – Switzerland, SSTC-NRS – Ukraine, SUJB – Czech Republic, UJD – Slovakia;
- Technical Safety Organizations (TSO) and research institutes: IRSN – France, IBRAE – Russia, GRS – Germany, Bel V – Belgium;
- Utilities, Vendors: EDF, AREVA, Mitsubishi, SNERDI, Fortum.

The presence of organizations who are not directly involved in IVR strategy (because ex-vessel corium stabilization is the main option in their country) showed that, in the current post-Fukushima context, all safety options deserve to be studied and that safety issues in- and ex- vessel are closely connected.

2. THE IVR STRATEGY AS AN OPTION FOR SA MITIGATION

WENRA has recently initiated a work to harmonize the safety requirements at the European level. For new power plants, one of the safety objectives is to reduce potential releases, also in the long term which means that accidents leading to large or early releases must be practically eliminated and that other core melt accidents should cause only limited protective measures in area and time. The independence between all levels of defence-in-depth was also pointed out. In that respect, corium retention in the vessel should be examined, not as a unique option but as an option which would also increases the chances of success of the measures dedicated to protection of the containment. It was pointed out that deterministic analyses should cover core melt scenarios starting from all operational states. Postulated core melt accidents are typically considered with realistic assumptions and best estimate methodologies. The probabilistic safety assessment (PSA) is complementary to the deterministic analyses.

2.1. Panel discussion with Safety Authorities and TSO's

A first panel with the participation of ASN, US NRC, NSC, ONR, SEC-NRS, KINS, SSM and SSTC-NRS was organized to let safety authorities and TSOs express their position about corium retention in general and more specifically about IVR strategy. The discussion focused on the three following questions:

- General position on IVR strategy: it was acknowledged by all participants that, for a safety authority, the IVR strategy remains an option that must be examined on a case by case basis. No general conclusions can be drawn on simple criteria such as reactor power or design. Each reactor design has to be analyzed independently of conclusions drawn for other designs, even similar ones. The implementation of the various operations necessary to flood the reactor pit and ensure the stabilization of corium in the vessel all have an important impact on the safety margins. Of course, the larger the reactor power, the more detailed the safety analysis must be because the safety margins are expected to be lower. One key conclusion was that it is important to harmonize the basis of knowledge at the R&D level first, in order to allow safety authorities to define safety criteria for IVR strategy which can be shared at the international level.
- Methodology to address IVR safety assessment (deterministic or probabilistic approaches) - Main issues for a robust assessment: there was no general position about the methodology to be used because, as it was pointed out in the previous point, a common basis of knowledge is still incomplete for the issues related to IVR (despite the large number of data already obtained in previous R&D programs). It was mentioned by several participants that safety assessment should not rely only on probabilistic approaches but should include also a deterministic point of view, in order to understand the main issues related to corium retention in the vessel and make a reliable estimate of the risks. Criteria of acceptability: short term and long term assessment of vessel integrity. At this point, it is clear that the criteria of acceptability may vary according to the reactor design. Among the most important criteria, the minimum vessel thickness appears as a critical one. It is obvious that the vessel would be highly ablated in case of IVR, with just a few centimeters of solid steel remaining at some places. In principle, it was demonstrated that such thickness would be enough to resist the static load corresponding to the weight of corium and small overpressure in LB LOCA but it was never really demonstrated if it would be enough to resist a significant spike of primary pressure, following for example, a reflooding, especially in

case of SBO or SB LOCA scenarios. In addition, long term (i.e. several months) resistance of the ablated wall remains to be assessed. It was also mentioned that human factor might play an important role in IVR strategy because the decision of flooding rapidly the reactor pit is crucial for the success of IVR but is not reversible.

2.2. Examples of designs with implementation of IVR strategy

In spite of the difficulties to assess in a reliable manner the effectiveness of the IVR strategy, several vendors have designed reactors in which this strategy can be used, in case of severe accident with core melting. Some examples were given during the workshop for two large power PWRs (CAP-1400 and APR-1400), for one BWR (KERENA) and one PHWR (CANDU). Even though it is not the point of this paper to review all the elements which were given for the safety demonstration of those designs, it is interesting to identify the similarities between them and the design options which are considered as positive for IVR success.

— The passive reflooding of the reactor cavity, up to the level of the primary loops. This is even improved in the case of AP1000 and CAP-1400 where it is possible to flood a much larger volume covering most of the primary circuit, allowing a simultaneous in-vessel reflooding if the break is below the water level and the vessel depressurized. The design of vessel insulation in order to allow water circulation between the vessel wall and insulation and to allow steam venting in the top part. The simultaneous in-vessel water injection (direct such as in APR-1400 or not as in CAP-1400) in order to reduce the risk of focusing effect.The presence of a large mass of steel in the lower plenum (examples of KERENA and CAP-1400) which would most likely avoid the formation of a heavy metal layer and would increase the thickness of the top metal layer, thus reducing the risk of focusing effect.The absence of penetrations (as in several PWR and PHWR designs)The large water inventory (including safety injections and storage tanks) leading to delay the time of corium arrival in the lower plenum and therefore the residual power to extract (up to one day before corium arrival for the KERENA (see Fig. 1.) and CANDU designs, up to 3 days for VVER-TOI).

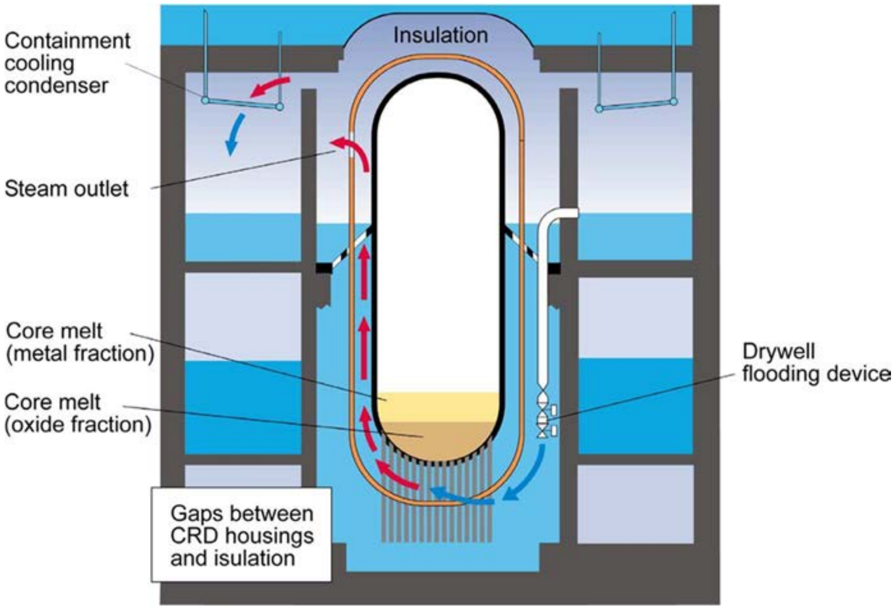

FIG. 1. Example of the KERENA design, from Fischer's presentation.

From those examples, it is worth noting that designers are aware that some uncertainties, such as complex physico-chemical phenomena in the corium still exist but they propose design options and measures which would, in any case, help to prevent large heat flux to the vessel. They also propose measures which should reduce the occurrence of fast transients which are critical for the safety assessment of IVR. In order to complete the international panorama given in that session, it is important to recall that In Russia, the IVR concept is not considered for high power VVERs, however for middle power reactors (VVER-600, 440) deterministic analyses demonstrated the effectiveness and reliability of decay heat removal by external reactor vessel cooling.

3. OVERVIEW OF MAIN RESULTS IN THE LAST DECADE AND R&D NEEDS

3.1. Corium pool

Several presentations were dedicated to the behavior of corium in the lower plenum and the resulting heat flux to the vessel, from the oxide pool and from the metal layer. Since MASCA results, it is well known that the thermochemical processes play an important role in the stratification of metal and oxide layers and hence on the heat transfers. The miscibility gap between molten fuel and molten steel may lead to various patterns of stratification (see Fig. 2.), with possible formation of a heavy metal layer for transient periods, which makes the thermal analysis more complex than in the standard analysis where the metal layer is always on top of the oxide pool. This is a major source of uncertainty. Further experiments are necessary to investigate situations which are more complex than in MASCA i.e.: situations with presence of an oxide crust separating metal and oxide, situations with oxidizing atmosphere, situations with solid debris in the molten pool.

FIG. 2. Examples of MASCA MA-3 and MA-9 tests with a heavy metal layer and a light metal layer, respectively (from Bechta et al. presentation).

The knowledge about turbulent natural convection in the oxide and metal layers, and the correlations for heat transfers were also presented. Currently, the correlations used in codes are based on the same set of experimental data obtained in the past for the highest Rayleigh numbers that could be reached. Discrepancies between correlations exist at very high Rayleigh numbers but this uncertainty is known and can be taken into account in the analysis. However, very large uncertainties remain in the analysis of the top metal layer when it is thin (focusing effect). Several issues including scaling, 3D effects, radiative heat transfers and conduction in the vessel wall make the evaluation of the heat flux along a thin metal layer difficult, with large uncertainties. At present, this is a key-issue for the improvement of the safety demonstration of IVR. Experiments will be proposed in the IVMR and SAFEST projects to study some of these issues. In parallel, CFD appears as a promising way of tackling this problem.

3.2. Corrosion of vessel steel by corium and molten metal

At very high temperature, vessel steel reacts with both the oxide corium and the molten metal phase.

Corrosion of the vessel by the oxide corium was studied during the METCOR project (phases 1 and 2). It was observed that there are two steel corrosion mechanisms depending on the oxygen content of the system. Those mechanisms could be described by models. The rate of vessel corrosion was evaluated. It becomes significant at temperature above 1100°C. Therefore, corrosion does not affect the cold part of the vessel which is most resistant one.

The interaction of vessel steel with a molten metal containing Zirconium is exothermic and might be self-propagating under some conditions. However, this effect could not be observed clearly in the experiments performed up to now (possibly because of other effects causing heat losses). In the absence of further evidence, it is recommended to take into account the heat produced by intermetallic reactions in the models used for IVR.

3.3. External cooling

External reactor vessel cooling (ERVC) was studied rather extensively in the past leading to several correlations for the maximum heat flux that can be extracted from the vessel wall (CHF) as a function of the water flow rate, the local void fraction, the angle of the wall and the water sub-cooling. CHF increases with the mass flow rate, sub-cooling and inclination angle. Predictions are more complex for the complete natural circulation loop consisting of the channel between the vessel and the insulation and the volumes and pipes driving water back to the bottom of the vessel. The strategy of demonstration followed up to now relies on the design of a full scale loop representing a slice of the real geometry, with prescribed heat flux profiles applied along the internal vessel wall. This was done for the VVER-440 reactors in Europe, and results for CHF were obtained in full-scale experiments ULPU (in a semi-elliptical design) and CERES. This was also done for the AP-1000 design, with results obtained in the ULPU experiment (in a hemispherical design). Currently, a comparable experiment is under preparation at UJV (Czech Republic) for the VVER-1000 design, as a preliminary work for a possible implementation of IVR strategy in VVER-1000 reactors in Europe and Ukraine. Similar experiments were built in China (REPEC-2, 3D-IVR) to optimize the design of Chinese reactors. Those experiments have shown that values of the CHF between 1.2 and 2 MW/m2 can be reached, depending on the level of optimization of the loop. Apart from the design of the cooling loop, the effects of the surface characteristics and of the composition of water still need to be investigated in order to obtain clear conclusions. It is generally considered that water chemistry effects are less important than surface effects. It was demonstrated that the roughness and “micro-porosity” of the surface play a role to increase the CHF. Such natural surface roughness comes from the oxidation of the external surface of the vessel under normal operation. However, this oxidation is not controlled and it might be uncertain to rely on it to justify increased performances of ERVC. Alternative options such as industrial coatings are also studied, to guarantee an increase of the CHF. Up to 30% increase can be expected for the CHF. It is important to remind that the remaining thickness of vessel (non ablated) is inversely proportional to the heat flux through the vessel, which would be of the order of the CHF value (but lower, considering the safety margin). For a heat flux of 1MW/m2, the residual thickness is about 3cm. Therefore, it does not appear as a good option to find ways to increase the CHF above 3MW/m2 because the residual thickness would become lower than 1cm which would raise serious issues for the mechanical resistance. Another significant issue for the realization of the external vessel cooling is a transient heat transfer during vessel flooding. In normal operation conditions the vessel wall is heated to the temperature well above the water saturation temperature at the containment pressure (which is rather low), and the transient heat transfer during flooding may be different from the steady state or even fall into film boiling region. Thus this transient heat transfer problem should be carefully analysed.

3.4. Mechanical resistance of ablated vessel

At steady state, when the maximum heat flux is reached, the vessel is significantly ablated in some parts (see Fig. 3). The residual thickness can be reduced to 3cm or less (starting from an initial intact thickness of 15 to 20cm). It was proved that such residual thickness is able to withstand static loads corresponding to the weight of corium and molten metal, with sufficient margin. This results from the “shell” of cold external steel preventing large deformation and mechanical damage. Hence, the mechanical properties (elastic-plastic) at moderate temperature are likely more important than creep properties, since RPV mechanical resistance is provided essentially by its cooled outer zone.

However, at higher pressure (above 30 bars over pressure), and depending on other hypotheses (residual thickness, pressure loading duration, reactor design…) mechanical failure is possible. The situation is even more complex when penetrations exist. This means that any significant over pressure must be avoided in order to keep the corium within the vessel. Therefore, events like a late in-vessel reflooding or sudden closure of RCS valves must be examined in order to evaluate the transient primary pressure increase. The thermal shock behavior of the vessel wall in transient conditions during cavity flooding should also be addressed.

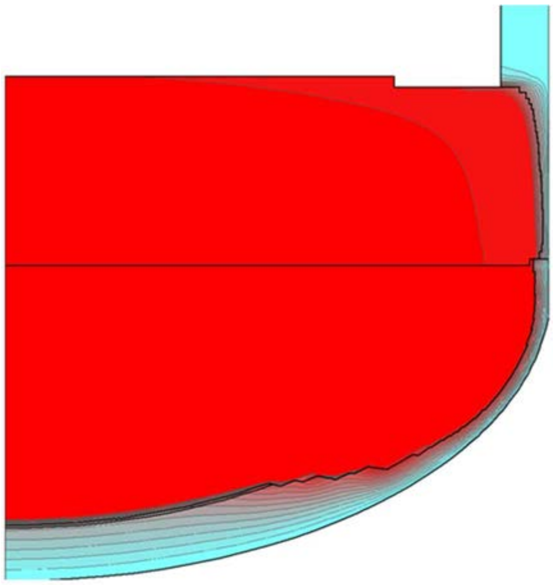

FIG. 3. Example of calculated ablation of the vessel (from Filippov et al. presentation).

3.5. Overview of analytical tools and codes

A session was dedicated to the analytical tools that are currently used for the analysis of IVR and the estimate of vessel failure probability. All the codes include similar approaches to couple the core degradation phase and the molten pool phase. Core degradation calculation provides the mass flow rates of core materials (fuel and cladding) which are used to calculate the transient evolution of stratified layers in the lower plenum. Some codes are more designed as a combination of a probabilistic method with lumped parameter models (such as SIMPLE-COMPASS presented by KAERI).

The consideration of transient evolutions of layers is a significant improvement compared to the earlier models which considered only static configurations representing steady state situations.

For the calculation of convective heat transfers in the pool, all codes use almost the same set of correlations established from data of large scale experiments (COPO, BALI …). All lumped-parameter codes also use similar simplifications for the thin metal layer (focusing effect) which can lead to a likely conservative evaluation of the heat flux along the metal layer. However, codes differ significantly in the way they take into account thermochemical effects such as non-miscibility or crusts. Some codes do not include models for the miscibility gap and some codes make different assumptions on the presence of oxide crusts at some boundaries of the pool. This results obviously in different evaluations of heat transfers.

Ablation of the vessel and mechanical resistance of the ablated wall are treated with various levels of accuracy: some models use a rather rough meshing and simple mechanical failure criteria whereas other models use a fine meshing and finite-elements calculation to estimate deformation and stress. This is a point which will require efforts in order to harmonize the models in different codes.

One positive point is that codes with the same level of development such as ASTEC and SOCRAT, for example, are able to predict similar transient phenomena and heat fluxes. A good example is the prediction of a transient peak of heat flux during unsteady vessel melting or after the inversion of stratification, with comparable values (above 3MW/m2 for several minutes), as shown in Filippov et al. and in Fichot et al. presentations. This indicates that a harmonization of models and codes could be reached in the future.

Another promising direction of R&D is the potential of CFD codes to calculate high Rayleigh number natural convection in molten pools (see Fig. 4). CFD could be the solution to avoid using conservative assumptions in excessively simplified models. In addition, CFD tools could bring insights (transient behaviour, local flow and heat transfer). CFD tools can also be used for the external cooling flow and may help for design optimization. However, CFD codes need to be validated. The accuracy of CFD predictions for the focusing effect and for the CHF will be investigated in the IVMR project.

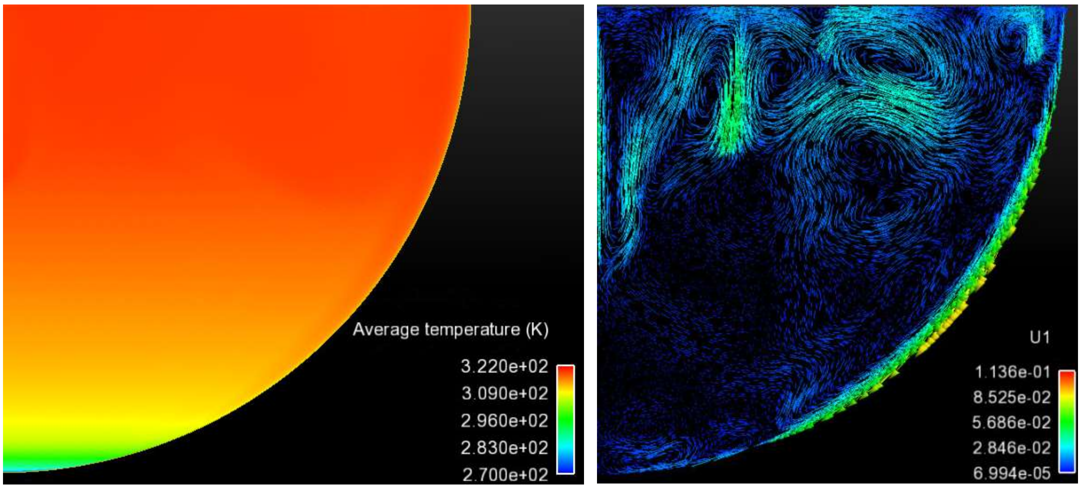

FIG. 4. Examples of flow field calculation with NEPTUNE-CFD in a large molten pool (BALI experiment), from Le Guennic et al. presentation.

4. PERSPECTIVES AND INNOVATIONS FROM INDUSTRY

The final session of the workshop was a panel session with industry and TSOs. Several items were discussed: decrease the probability of occurrence of transients leading to an early core melt (thanks to active/passive safety systems): This would improve the robustness of an IVR strategy for a PWR-1000. This was illustrated by AREVA who mentioned that in-vessel injection effectiveness depends on core degradation status: before corium relocation, it will delay core melting, and after corium relocation, it will cover the top metal layer and hence reduce focusing effect. In any case, it would enhance Zr oxidation and hence have an impact on corium stratification. Among the passive systems that could be implemented, there is a choice between dedicated accumulators (pressurized tanks), dedicated core make-up tanks (both sides of which are connected with the primary circuit) or elevated gravity tanks. It was pointed out that in several designs, the IVR strategy also relies on in vessel injection whatever the value of CHF may be. Decrease the maximum heat-flux to the vessel (dilution of corium, increase of mass of steel and water injection into RPV, etc.): The impact of steel mass was illustrated with the CAP1400 design by SNERDI. The low position of the core support plate leads to its early melting soon after corium relocation, resulting in a large mass of steel and avoiding transient situations with a thin metal layer. The total mass of molten steel is more than 70 tons which is enough to avoid the focusing effect (no formation of heavy metal layer). The discussion also focused on the fact that not only necessary sufficient conditions should be identified but also sufficient conditions to guarantee the success of the IVR strategy. Criteria and conditions for in vessel injection were identified as an issue for CAP1400 considering associated risks. Increase the efficiency of external cooling: Strategy to apply the “cold spray” as possible innovation to improve the ERVC efficiency, based on existing small scale experiments. This point was illustrated by UJV who presented a coating technology which can enhance the CHF by up to 60%, in particular for low inclination angles (bottom of the vessel). It should be noted that in the areas where a high heat flux is expected (vertical part of the vessel), the increase of the CHF would be lower (15% to 30% increase are expected). This technology appears to be robust and reliable. It could be used for semi-elliptical vessels in order to make sure that a sufficient heat extraction is provided at the bottom, in the flat part of the vessel. Mitigate nevertheless the consequences of vessel failure and corium interaction with water considering the defense in depth concept and to demonstrate that vessel failure does not lead to a cliff edge effect: Comparison of consequences of vessel failure with and without a flooded cavity. This point was illustrated by some studies made at IRSN. It was suggested that a good practice can be to postulate a failure of an IVR strategy and to design features so that the long term corium stabilization can still be reached with no containment failure. This means combining IVR strategy with ex-vessel stabilization configuration. Another subject of the safety demonstration of the IVR strategy that was discussed was the need to demonstrate that this strategy does not lead to a non coolable corium configuration for long term situations

5. CONCLUSIONS

The state of the art was presented and discussed. Though significant progresses were made, there is still lack of knowledge and a need to continue the research in different domains. Indeed, currently, there are still missing data to be able to completely understand the phenomenology of the processes involved in IVR. It includes thermochemical effects, high Rayleigh number turbulent convection, surface effects on the CHF and mechanical behavior of the thin “cold shell” resulting from ablation of the vessel. So, the standard analysis for safety demonstration of IVR suffers from two opposite drawbacks: it neglects transient situations which may lead to higher heat fluxes than steady-state situations and it uses excessively simplified models (such as focusing effect) which may be too conservative. The current approach followed by most experts for IVR is a compromise between a deterministic approach using the significant knowledge gained during the last two decades and a probabilistic approach to take into account large uncertainties due to lack of data for some physical phenomena (as listed above) and due to excessive simplifications of models.

It was concluded that a harmonization of the positions of safety authorities on the IVR strategy is necessary to allow decision making based on shared scientific knowledge and not on individual judgments. For this, a consensus on several issues should be reached between R&D experts. This includes in particular the issue of the transient evolution of oxide and metal layers in the lower plenum and the issue of the long term mechanical behaviour of the ablated vessel wall. Additional R&D should be encouraged, as well as international collaborations on the topic. In Europe, the IVMR project aims at providing new experimental data and a harmonized methodology for IVR. It will also include an activity on innovations dedicated to increase the efficiency of the IVR strategy. More generally, this workshop has shown that industrial actors make constant efforts to take into account existing knowledge and uncertainties in order to propose more effective implementation of the IVR strategy. There is a clear distinction between existing reactors for which the implementation of IVR strategy is a serious challenge and new reactor designs for which several options can be implemented to increase the safety margins, such as delaying corium arrival in the lower plenum, increasing the mass of molten steel or implementing measures for simultaneous in-vessel water injection.

THE "PRACTICAL ELIMINATION" APPROACH FOR PRESSURIZED WATER REACTORS

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Abstract

The reinforcement of defense-in-depth for generation III reactors has led vendors to define, at the design stage, provisions to prevent severe accidents and to limit their consequences should they occur. Nevertheless, taking into account the state of knowledge and techniques available during the design of a reactor, the definition of realistic and demonstrable mitigation provisions for all severe accident situations may be not possible. Therefore, an objective of "practical elimination" of situations that may lead to large early releases has been defined for reactors being designed in the 90's. On the basis of the experience gained since that time with new reactors projects, the paper focusses on the place of "practical elimination" in the safety case and how the "practical elimination" of situations can be justified, on a case-by-case basis, using deterministic and probabilistic considerations. The need to introduce robust and reliable lines of defense in the design to prevent such situations to occur is discussed. In this respect, implemented provisions shall fulfill requirements in terms of design, manufacturing, construction, qualification and operation.

1. INTRODUCTION

Following the Three Mile Island and Chernobyl accidents, the need of a significant improvement of the safety of reactors to be built from the start of the 21st century compared to the ones in operation or under construction at the end of the 20th century has been recognized at the international level. This improvement seemed to be achievable for water-cooled reactors, taking into account R&D works performed on core melt accidents.

Thus, provisions aiming to limit the releases outside the plant in case of severe accident have been introduced in the initial design of generation III reactors. However, in some core melt situations that could be envisaged at least theoretically, it seemed impossible to implement realistic measures that would reduce the radiological consequences at an acceptable level and to demonstrate their robustness. For this reason the concept of "practical elimination" was introduced.

The term "practical elimination" appeared in the beginning of the 90's in the definition of the general safety objectives for the future pressurised water reactors to be built in France and Germany. These objectives have been then developed in the technical guidelines for the design and construction of nuclear power plants with pressurized water reactors in 2000 [1].

In the meantime, INSAG 10 [2] stated in 1996 that: "*For advanced designs, it would be demonstrated, by deterministic and probabilistic means, that hypothetical severe accident sequences that could lead to large radioactive releases due to early containment failure are essentially eliminated with a high degree of confidence*".

Then, in 1999, INSAG 12 [3] indicated: "*27 - the target for existing nuclear power plants consistent with the technical safety objective is a frequency of occurrence of severe core damage that is below about 10^{-4} events per plant operating year. Severe accident management and mitigation measures could reduce by a factor of at least 10 the probability of large off-site releases requiring short-term off-site response. Application of all safety principles and the objectives of para.25*[1] *to future plants could lead to the achievement of an improved goal of not more than 10^{-5} severe core damage events per plant operating year. Another objective for these future plants is the practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic*

[1] « *25. For future nuclear power plants, consideration of multiple failures and severe accidents will be achieved in a more systematic and complete way from the design stage…* »

assumptions and best estimate analyses so that their consequences would necessitate only protective measures limited in area and in time".

The latest international texts state that "practical elimination" should be applied to situations likely to lead to early or large releases [4, 5], thus widening the range of situations potentially concerned. According to this wording, the "practical elimination" approach could be used for all situations that can cause large releases, whether or not they lead to early releases.

2. THE "PRACTICAL ELIMINATION" APPROACH: GENERAL CONSIDERATIONS

Nuclear facilities are designed according to the defence-in-depth principle: a series of measures are defined on the one hand to prevent accidents and on the other to limit the consequences of any accidents that do occur despite the measures taken to prevent them.

There are today five levels of defence-in-depth for nuclear reactors; each level aims to reduce the consequences of the failure of the previous level and to prevent challenging the next one, in compliance with the general safety objectives for these reactors:

- The first four levels rely on physical, human and organisational measures defined by the operator;
- The fifth level, of different nature, essentially consists of organisational measures aiming to protect the population and the environment in case of off-site releases.

The design mainly focusses on the first four levels of defence-in-depth, aiming to reduce as far as possible the risk of off-site radioactive releases. The safety approach used for the design of new nuclear facilities aims to enhance defence-in-depth, improving the robustness of the different levels, increasing their independence, improving the way hazards are taken into account, etc. However, it is important to point out that the implementation of defence-in-depth has its limitations and that it is not possible, for facilities where large quantities of radioactive substances coexist with an amount of energy able to disperse them, to limit the consequences to an acceptable level in all cases.

So situations likely to lead to large releases, because of the simultaneous or successive loss of integrity of all the containment barriers or because of the bypass of these barriers (containment bypass situations):

- Either lead to define provisions allowing to significantly limit their consequences;
- Or have to be "practically eliminated" where it appears to be impossible to define such provisions or to demonstrate their adequacy with the knowledge and techniques available at the time of the design orientations.

As far as possible, priority has to be given to the implementation of mitigation provisions for severe accidents; it generally increases the robustness of the overall design, preventive measures being already defined according to the defence-in-depth principle. Moreover, the guarantee to cover the different sequences that may lead to a dreaded situation is higher.

For French water-cooled reactors, for example, the situations likely to lead to large radioactive releases are:

- Core melt with loss of containment integrity – case 1;
- Core melt with containment bypass – case 2;
- Melt of spent fuel assemblies being handled or stored in the spent fuel pool – case 3.

Cases 1 and 2: the integrity of the containment could be affected suddenly by the occurrence of a high-energy phenomenon (e.g. massive and rapid reactivity insertion, hydrogen detonation, etc.), or more gradually due to the loss of the containment heat removal function (slow increase of temperature and pressure in the containment, erosion of the basemat by the corium, etc.). In the latter case, the uncertainties associated with possible evolutions of the plant conditions are smaller and the operator would have some time to act before large releases occur. The definition of measures to limit the consequences of these situations and the demonstration of their adequacy are possible. This is why only fuel melt situations that could lead to large early releases have to be "practically eliminated"; by extension, core melt sequences with containment-bypass are considered as situations to be "practically eliminated" as well. Generally, for other situations, measures have to be taken by the operator to limit their consequences.

Case 3: in this case, the implementation of measures to limit releases due to the melt of spent fuel assemblies could technically be envisaged. However, in France, the spent fuel pool is located outside the containment; thus, given the scale of the releases that could result from fuel melt in the spent fuel pool, the "practical elimination" approach has been used also for these situations.

Therefore, while defining the design orientations for a new water-cooled power reactor, vendors shall use the "practical elimination" approach for severe accident situations (in the reactor core or in the spent fuel pool) potentially leading to large early radiological releases, where it seems impossible to define realistic and demonstrable measures to limit their consequences according to current knowledge and the techniques available at the time.

The concept of "practical elimination" of accidental situations should not be extended beyond the framework described above. In any case, it should not be applied without discussions between vendors and safety authorities in the early stages of a new design, primarily with the intention of improving the defence-in-depth.

3. SITUATIONS FOR "PRACTICAL ELIMINATION"

It is crucial that all plausible situations for "practical elimination" are identified. The different modes of failure or bypass of the last barrier (or of its extension) in case of severe accidents (e.g. explosion, combustion, etc.) should be considered and the situations which can lead to the failure or bypass modes identified (e.g. steam explosion, release of mechanical energy during a reactivity insertion accident, hydrogen detonation, etc.).

These situations have to be characterised taking account of uncertainties due to limited knowledge of certain physical phenomena and relying as much as necessary on dedicated studies or research and development activities.

The event sequences representing all possible severe accident scenarios leading to situations for "practical elimination" have to be identified.

4. JUSTIFICATION OF "PRACTICAL ELIMINATION"

The justification of "practical elimination" of an accidental situation is based on a case-by-case analysis and relies on deterministic and probabilistic considerations. It has to be pointed out that such a justification cannot be claimed solely based on compliance with a general cut-off probabilistic value [1, 5].

An accidental situation can be considered "practically eliminated" if:

— It is physically impossible for the accidental situation to occur;
— The accidental situation can be considered with a high degree of confidence to be extremely unlikely to arise [4, 5].

The justification of "practical elimination" should preferably rely on the physical impossibility of the situation.

4.1. Physically impossible situations

Demonstration of the physical impossibility of a situation can be based on various considerations, for example:

— Intrinsic characteristics that guarantee the non-occurrence of certain dreaded phenomena (e.g. neutron feedback);
— Design choices limiting the quantities of substances likely to produce energetic phenomena (e.g. limitation of the capacity of deborated water tanks for the circuits connected to the reactor primary circuit in order to prevent heterogeneous dilutions, which would lead to a reactivity-initiated accident);
— Passive static systems that cannot fail (such as construction provisions allowing to prevent a heavy load drop from seriously damaging the structural integrity of the spent fuel pool and causing the uncovery of spent fuel assemblies or of an assembly being handled).

It should be noted that the demonstration of physical impossibility must not, under any circumstances, be based on measures requiring active components, since any active component has a non-zero probability of failure.

4.2. Situations extremely unlikely with a high degree of confidence

If physical impossibility cannot be demonstrated, it has to be proved that the situation is extremely unlikely with a high degree of confidence. The justification, assessed on a case-by-case basis, is based on a deterministic approach, generally complemented by probabilistic assessments and relies on the following principles:

- The deterministic justification for a situation being "practically eliminated" must be based on both the existence of a sufficient number of lines of defence consisting of material and organizational provisions and the robustness and independence of these different lines of defence.
- Material provisions defined within the scope of the "practical elimination" of an accidental situation shall be subject to requirements concerning the design (diversification, geographical separation, backup power, qualification, reliability, etc.), manufacturing (quality control) and operation (operation monitoring…). This also applies to the instrumentation used to check the functions fulfilled by these provisions. The more a provision or a set of provisions contributes to reduce the probability of a situation occurring, the more the requirements are stringent.
- Provisions defined within the scope of the "practical elimination" of an accidental situation shall be tolerant to human actions and errors:
  - The operator has to prove that these measures cannot fail because of human errors prior to the accident (e.g. maintenance) or that it takes all necessary measures to limit the probability of such errors;
  - If the justification for "practical elimination" is partly based on human actions (e.g. the depressurisation valves in the Flamanville 3 EPR primary circuit, which have to be opened manually by the operators to "practically eliminate" high-pressure core melt situations), the operators must have the necessary information for carrying out these actions without misunderstandings and for checking their effectiveness. The detection systems must be reliable, clear and explicit, and the time scales within which the actions must be carried out following the alert shall be long enough, particularly given the conditions of intervention, so that the actions can be considered to have a very low probability of failure.
- Provisions defined within the scope of the "practical elimination" of an accidental situation shall be tolerant to internal and external hazards; in particular, the occurrence of rare and severe external hazards shall not call into question a "practical elimination" justification. Preference should be given to the implementation of measures that are tolerant to the loss of support functions.
- Although probabilistic safety assessments (PSA) can be used to assess the exhaustiveness of the measures taken to avoid certain accidental situations (in particular their investigation methods can identify situations resulting from multiple failures not identified deterministically (support system failures, common cause failures, human errors, etc.)), they should be used with care for assessing the extreme unlikeliness of situations for "practical elimination" because of the impact that the models used and the assumptions taken into account can have on the results.

5. CONCLUSION

The "practical elimination" approach is used in the design of pressurised water reactors for severe accident situations that can lead to large early releases, where it appears to be impossible to define realistic and demonstrable measures to limit the consequences of these situations.

The situations for which this approach can be used shall be discussed by vendors and safety authorities from the initial design stages of a new reactor type. The technical dialogue should continue throughout the entire licensing process.

However, any agreements resulting from these discussions are time limited given that they take account of current knowledge and technical possibilities. Although using the "practical elimination" concept means that the consequences of the accidental situations concerned are not studied in detail and are not included in the formal safety demonstration, this should not stifle continuous efforts to improve safety, further reflection leading to potential safety improvements may still be carried out later on, after the initial design phase, for example during periodic safety reviews.

In order to "practically eliminate" an accident situation, the vendor shall first examine the possibility for making it physically impossible. If this cannot be achieved, measures have to be taken that enable the situation to be considered extremely unlikely with a high degree of confidence. These measures shall be subject to requirements in terms of design, manufacturing and operation; they shall be tolerant to human actions, human error or hazards. The justification of "practical elimination" should be examined on a case-by-case basis, using deterministic considerations, completed by a probabilistic assessment.

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SAFETY AND SITTING ASSESSMENT FOR NPPS DEPLOYMENT IN INDONESIA

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Abstract

Study on national economic and energy sources of Indonesia had shown that nuclear energy will be included as part of energy sources in our national energy mix policy in addition to fossil energy source, gas, water and other new and renewable energy. Fukushima accident made people doubt about the safety of Nuclear Power Plant (NPP), which impact on the public perception of the safety of nuclear power plants. Lesson leant from Fukushima accident, the safety of nuclear power plants need to be highlighted back to improve the public acceptance of nuclear safety and security in Indonesia. Preliminary safety assessment and sitting preparing for Large Reactor (LWR), and technology assessment on various technologies of SMR has been started. The paper will describe the assessment of safety and radiation consequences on site for normal operation and Design Basis Accident postulation of PWR-1000 MWe in Bangka Island. Consequences of radiation for normal operation simulated for 2 units NPPs. The source term was generated from an inventory of peak radioisotope activities released by using ORIGEN-2 software and the consequence calculated by PC-Cream (routine release) and PC Cosyma (accident release). The adopted methodology used was based on the predominant site-specific meteorological data. The results of calculation are: doses on public for normal operation below 1 mSv/yr, and no evacuation of emergency planning for DBA postulation.

1. INTRODUCTION

As an archipelago country, Indonesia consists of five big islands and more than thousand medium and small islands with various electricity ratio and industrial development. Total area of 1.9 million square miles including the ocean. Population is around 250 millions people with rate of 1.21% (2015). National economic and energy assessment has shown that nuclear energy would be part of energy sources in national energy mix policy simultaneously with fossil energy source, gas, water and other renewable energy. The National Energy Policy has increased share of new and renewable energy at 31% by 2050 to reduce dependence on fossil & to diversify energy (Presidential Regulation No. 5 Year 2006). National Long-Term Development Planning 2005 – 2025 that new energy resources can be produced by new technology, i.e. renewable or non-renewable energy, such as nuclear energy (Act No.17 and 30 Year 2007). Projection of energy supply by 2050 is 430 GWe (low scenario) and 550 GWe (high scenario). Projected role of nuclear energy ~55 reactors by 2050. High demand and supply of energy in the future has chosen that SMR technology is much more appropriate for small and medium islands to support their development and Large NPP for Big Islands. In order to accelerate national development, President of Republic Indonesia issued Presidential Instruction No.1/2010 regarding Accelerating National Development Including Nuclear and Government Regulation No. 5/2010 on National Medium Term Development that Include Nuclear Power Plant as part of alternative energy. In line with Indonesia nuclear Infrastructure development progress in phase 1 and phase 2, main activities are recently in progress i.e.: Feasibility Study and site evaluation in Bangka-Belitung island. Based on the assessment result, Indonesia planned to be built large NPP on the Bangka Belitung Island.

Fukushima accident made people doubt about the safety of Nuclear Power Plant (NPP), which impact on the public perception of the safety of Nuclear Power Plants. Lesson leant from Fukushima accident, the safety of nuclear power plants need to be highlighted back to improve the public acceptance of nuclear safety and security in Indonesia. To prove that the operation of a large nuclear power plant in Bangka Belitung required analysis of radioactive dispersion into the environment under normal conditions and postulation accident. The paper will describe the assessment of safety and radiation consequences on site for normal operation and DBA (Design Basis Accident) postulation of PWR-1000 MWe in Bangka Island. Consequences of radiation for normal operation simulated for 2 units NPPs. The source term was generated from an inventory of peak radioisotope activities released by using ORIGEN-2 software and the consequence calculated by PC-Cream (routine release) and PC Cosyma (accident release) [1]. The adopted methodology used was based on the predominant site- specific meteorological data and spatial data.

2. METHODOLOGY

The assessment approach in the paper use based on atmospheric dispersion calculation has shown in Fig. 1 [1-4]. The reactor inventory use ORIGEN2.0, the dose calculation use PC-CREAM 08 for normal operation [2] and PC Cosyma for DBA calculation [1].

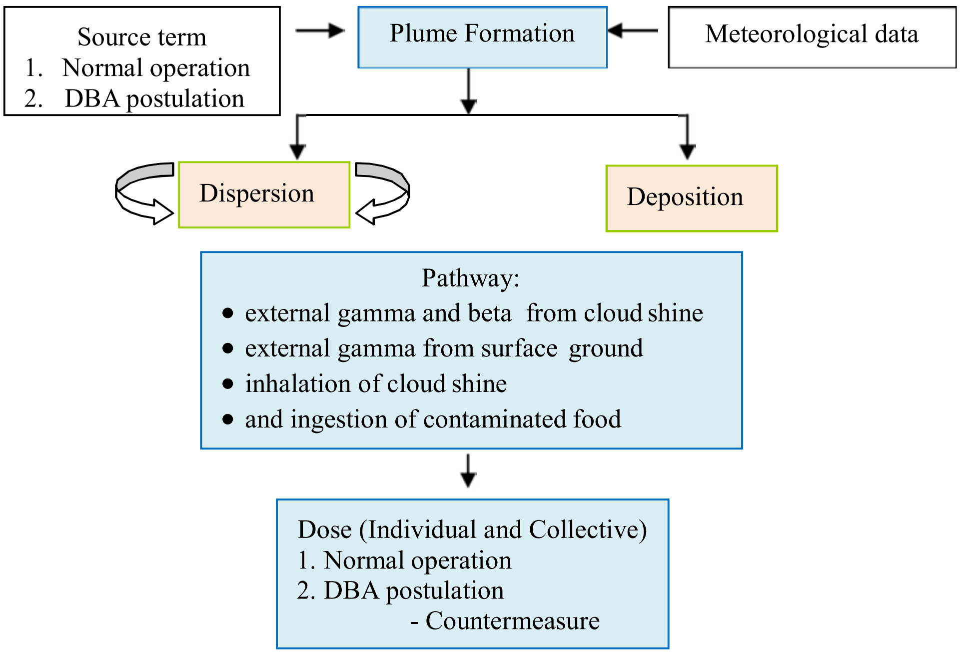

FIG. 1. Methodology approach of radiation dose calculation.

2.1. Inventory and source term calculation

The core nuclide inventory data were calculated with ORIGEN-2 code package [1-2], a zero-dimensional isotope decay and transmutation code. The calculation is done using the ORIGEN2 program based on reactor data. The condition of PWR-1000 MWe are: 3400 MWt; UO2 fuel; Number of fuel assembly 157; Enrichment (%) 2.35; 3.40; 4.45; Number of rod/assembly: 264; Fuel density (gram/cm^3): 10.28; Fuel weight as UO_2 (gram): 95 975; Operating cycle /year: 1.5; and average burn up 50 GWd/tU [5].

The source terms are calculated for normal and accident releases. The calculation of source terms begins with calculation of reactor inventory. After the inventory is obtained, the calculation of source term release from inventory until release to environment is conducted. The normal/routine release is assumed that the release of fission products from pinholes reached 0.1%. In addition to the fission products of the porosity of the cladding, the fission products in the primary coolant contamination also comes from natural uranium and enriched uranium on the outer surface of the cladding. Uranium contaminants in surface cladding can reach 10 microns uranium weight [6].

For the accident cases, scenarios were postulated by large break loss of coolant accident (LOCA). The enveloping scenarios of postulated initiating events (PIEs) are analyzed with conservative input conditions and assumptions for system availability; The PIEs to be covered in the design have been considered to constitute the Design Basis Accidents (DBAs), including a double-ended guillotine break of the main reactor coolant system piping or a design basis LOCA. The LOCA scenario involves calculation of radioactivities in the gap of fuel, in the reactor coolant, in the containment and at the outlet of reactor stack [7]. The source term is calculated based on reactor inventory with the assumption core damaged reaches approximately 33% which is the optimal damaged based on results of the study and an agreement for LB-LOCA accident [8].

2.2. Dose Calculation

A Gaussian atmospheric dispersion code PC Cream and PC Cosyma were used to estimate the radiological doses of the released radionuclides. Although the two-dimension Gaussian model cannot handle complex terrain and the spatial variability of meteorological fields, its fast runtime and simple algorithm makes Gaussian models an effective tool for sensitivity and statistical studies [1-2,9-10]. The model requires the following input data and parameters: time and location, atmospheric data (wind speed and wind direction, air temperature, relative humidity, cloud cover, mixing layer height), stability class of the atmosphere (proposed by the model based on meteorological data), roughness length and information about the release (air pollutant, duration of release, amount of released material, source height). The distribution of Pasquille Guifford atmospheric stability classes as an annual base of the West Bangka site which were processed from hourly. The most popular scheme of stability classes known as Pasquill stability used to calculate a stability category ranging from A to F was used to help predict how well the plume will disperse. There were nineteen distances chosen for the calculations: 800m-80km with sixteen sectors at different wind speed. Radioactives are released through the ventilation stack of 100 m height to the environment.

3. RESULT AND DISCUSSION

3.1. Core inventory and reactor source term

TABLE 1. CORE INVENTORY FOR PWR 1000 MWE

Nuclide	Bq	Nuclide	Bq	Nuclide	Bq
Kr 85m	1.67E+18	Y 92	7.92E+18	Te131	4.59E+18
Kr 87	3.34E+18	Zr 95	8.47E+18	Te132	7.33E+16
Kr 88	4.74E+18	Zr 97	8.99E+18	I131	5.14E+18
Xe133	1.07E+19	Nb 95	8.47E+18	I132	7.44E+18
Xe135	3.43E+18	Mo 99	9.58E+18	I133	1.07E+19
Rb 88	4.77E+18	Tc 99m	8.40E+18	I134	1.15E+19
Sr 89	5.96E+18	Ru103	8.95E+18	I135	9.92E+18
Sr 90	6.25E+17	Ru106	5.29E+18	Cs134	2.40E+18
Sr 91	7.66E+18	Rh105	6.48E+18	Cs137	9.14E+17
Sr 92	7.92E+18	Sb129	1.79E+18	Ba139	9.69E+18
Y 90	6.62E+17	Te129	1.77E+18	Ba140	9.32E+18
Y 91	7.25E+18	Te 129m	2.62E+17		

Table 1 gives the activity of core inventory for PWR-1000 MWe. Baed on the data on Table 1, the source term for normal/routine release and for LB-LOCA accident have been calculated. The result estimation shows on Table 2. The source term calculation in the normal/routine operations is activity for a year. Estimation of LB-LOCA source term is calculated using the following assumptions: 33% cores damaged, the gap release of noble gases; Kr = 7.5% and Xe = 2.15%, for Iodine = 0.65%, other nuclides = 0.0051. Release core inventory: Iodine is 0.22%, Cs-137 is 0.5%, and other nuclides 0.06% Reduction in containment for nuclide Iodine is 0.46. The efficiency of the filter in the chimney of the reactor was taken to 0% noble gases, iodine (organic) 90%, and other nuclides (Br, Te, Cs, Rb) 99% [5]. Based on the amount of activity, and a half-nuclide nuclide, and then selected 20 nuclides as the source term.

TABLE 2. SOURCETERM FOR NORMAL AND ACCIDENT

Nuclide	Normal, Bq/a	LOCA, Bq	Nuclide	Normal, Bq/a	LOCA, Bq
Kr 85m	7.27E+14	1.78E+16	Cs137	2.08E+04	1.69E+10
Kr 88	2.04E+15	5.00E+16	Te132	1.45E+04	2.36E+11
Xe133	4.43E+15	3.07E+16	Ba139	5.14E+05	3.39E+11
I131	5.06E+08	4.15E+14	Ba140	4.98E+05	3.28E+11
I132	7.49E+08	6.16E+14	Sr 90	1.89E+04	1.54E+10
I133	1.14E+09	9.41E+14	Mo 99	4.94E+05	3.25E+11
I134	1.26E+09	1.04E+15	Ru106	2.14E+03	3.80E+10
I135	1.06E+09	8.76E+14	Rh105	1.45E+05	9.23E+10
Rb 88	1.70E+10	1.72E+11	Y 90	1.99E+04	4.95E+08
Cs134	6.64E+03	5.31E+09	Y 91	4.97E+05	3.82E+10

3.2. Individual and collective dose

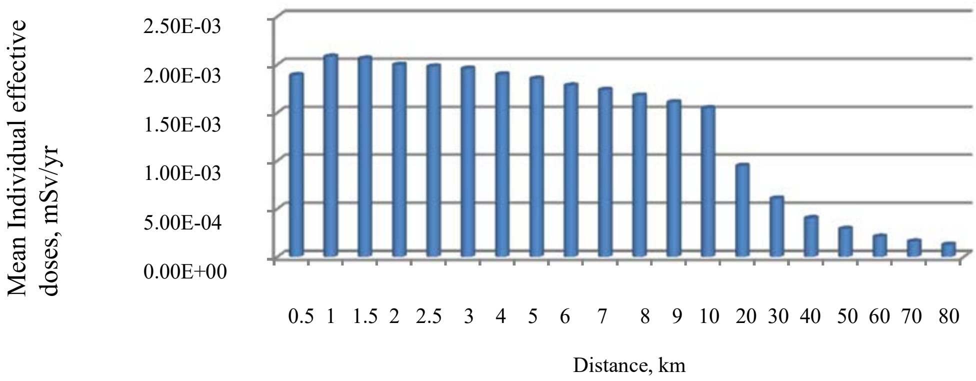

FIG. 2. Individual effective doses for normal operation.

The term dose for normal dose used in the paper is therefore the sum of the annual external and internal effective doses to individuals received over 1 y. The individuals doses at each sector were assessed assuming that the individuals occupancy indoors is 90%. According to calculation by PC-CREAM 08 computer code, the highest individual dose in terrestrial area for adults is 2.08E-02 mSv/y in NW direction and 1 km distance from stack. The highest mean individuals effective dose is 2.08E-03 mSv/y within 1 km distance from stack (Fig. 2). It can also be concluded that the estimated effective doses are lower than the dose constraint of 0.3 mSv/y (BAPETEN, 2014) associated with this plant.

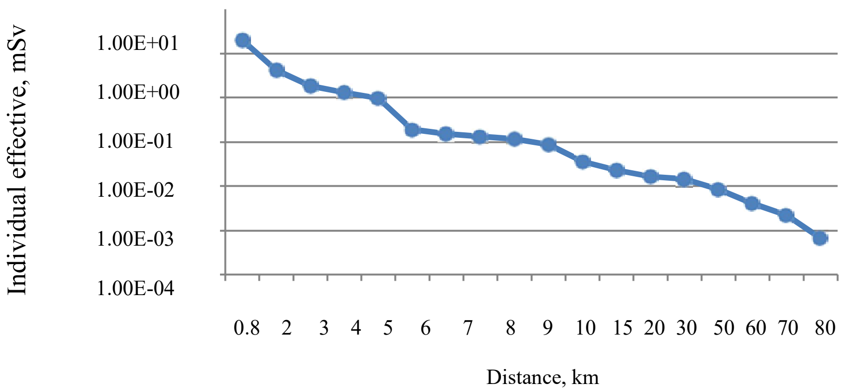

FIG. 3. Individual effective doses for LOCA in NW direction.

Fig. 3 shows the highest individuals' effective dose for LBLOCA is 2.00E+01 mSv in NW direction and 800m distance from stack. Results show that there are no evacuation countermeasure will be taken based on the regulation of emergency. Evacuation shall be taken if the dose have accepted by public is ≥ 50 mSv (BAPETEN). From Fig. 3 can be seen that the doses decrease when the increase distances. The total collective dose is 1.18E+00 manSv. The results showed that for LBLOCA postulated accident at 1000 PWR-MWe, does not lead to evacuation countermeasures on public in West Bangka site.

The assessment have been done to prove that the construction of large NPPs in West Bangka safe and meets the safety standard for normal operation and the impact in DBA conditions. This study used as supporting data for the dissemination of nuclear power plants in Indonesia. The purpose of dissemination of nuclear power plants is to increase public acceptance of nuclear energy. From the last survey shows that a total of 77.53% of Indonesia expressed its support for the construction of the Nuclear Power Plant (NPP) in Indonesia. The survey was conducted from October – December 2016 by distributing questionnaires to 4,000 respondents in 34 provinces throughout Indonesia (BATAN, 2017). The survey results shows the level of public acceptance of nuclear power development plan in 2011 (49.5%), 2012 (52.9%), 2013 (64.1%), 2014 (72%), and 2015 (75.3%).

4. CONCLUSION

Based on the routine discharge of normal operation, the highest individuals dose in terrestrial area for adults is 2.08E-02 mSv/y in NW direction and 1 km distance from stack. The highest mean individual effective dose is 2.08E-03 mSv/y within 1 km distance from stack. It can also be concluded that the estimated effective doses are lower than the dose constraint of 0.3 mSv/y associated with this plant (BAPETEN, 2014). The highest individual effective dose for LBLOCA is 2.00E+01 mSv in NW direction and 800m distance from stack. Results show that there are no evacuation countermeasure will be taken based on the regulation of emergency. The assessment have been done to prove that the construction of large NPPs in West Bangka safe and meets the safety standard for normal operation and the impact in DBA conditions. This study used as supporting data for the dissemination of nuclear power plants in Indonesia.

ACKNOWLEDGEMENTS

The study is supported by BATAN as part of the research activity on safety analysis of NPPs of PWR-1000 MWe. Authors thank to colleagues from Reactor Safety Technology Division for very helpful discussion.

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CHALLENGES DUE TO DIFFERENT REGULATORY FRAMEWORKS

Chairperson

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HARMONIZED EUR REVISION E REQUIREMENTS CORRESPONDING TO CURRENTLY AVAILABLE TECHNICAL SOLUTIONS

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Abstract

The revision E of “European Utility Requirements” (EUR) is under publication. Chapter 2.1 “Safety Requirements” has undergone major changes, which induced minor corrections in other chapters. The review consists of comprehensive analysis of IAEA Safety Standards for Design: namely IAEA SSR 2/1 Rev 1, GSR 4 Rev 1, IAEA SSG 2 Deterministic Safety Analysis and IAEA Specific Safety Guide SSG-30. Additionally, WENRA Report on Safety of new NPP design (March 2013) was considered. Among many improvements the defense in depth and the practical elimination concepts have been worked out including the design principles to prevent rare and severe external hazards. The list of external hazards has been extended, strict requirements for strengthening plant autonomy have been formulated. To include the lessons learned from Fukushima Daichii accident was particularly emphasized. New ideas concerning the safety classification have been implemented both for standardization and for extending the compatibility to a broad variety of mobile equipment. The terminology has been updated also in accordance with the IAEA Glossary and the TECDOC-1791 about SSR-2/1. The authors hope that the current revision will contribute to build new power plants withstanding most of the foreseen challenges and to reach the reduced limits of environmental impact.

1. INTRODUCTION

The nuclear safety requirements in the European Utility Requirements for LWR Nuclear Power Plants (EUR) Document should be in accordance with the European regulators licensing position and be a guide to the Designer/Vendor to understand these requirements. The common European regulatory requirements and guides are defined through WENRA and IAEA. This must be reflected in the EUR Document in order to make clear the safety level pursued in Europe. The nuclear safety requirements in the EUR Document must also be well-structured, and preferably make up a harmonized set of requirements, in order to facilitate assessments and biddings, of a new Nuclear Power Plant (NPP) in Europe. The present EUR Document (Revision D) suffers from some shortcomings in this regard.

The nuclear safety requirements were refined at three inherent levels of the EUR Document: *Safety fundamentals*, *Safety requirements* and *Design of systems, structures and components*. The safety fundamentals highlight the EUR nuclear safety ambition and declare that this is in concordance with the safety principles and objectives stated by IAEA and WENRA. The safety requirements are essential in defining the safety level that should be reached by the new NPP. Requirements for *design of systems, structures and components* are claimed as contributing to the achievement of the safety requirements, i.e. requirements on design- and engineering principles, necessary to demonstrate that the safety level above is reached throughout the lifetime of the new NPP.

The paper illustrates the work done for improving the EUR Document. First, the revision method is presented. Then the changes regarding the fundamental level of safety are shown. After that the improvements induced by the Fukushima Daichii accident are described briefly. The following main part presents the revised concept of safety classification. And finally, the description about physical protection relevant modifications and the interconnection between chapters can be found.

2. REVISION METHOD

Since the European regulatory documents have gone through improvements and new approaches have been defined, it was essential to start the work with a comprehensive examination of the relevant currently available publications. EUR Rev. E was intended to be up to date, therefore new versions of IAEA documents at their early draft stages were also considered with the objective of collecting ideas. The following regulatory requirements and guides were considered during the preparation of EUR Rev. E: IAEA SF-1 [1], IAEA GSR Part 4 Rev.1 [2], IAEA SSR 2/1 Rev.1 [3], IAEA SSG 2 Rev.1 (DS491, Step 8a) [4], IAEA SSG 30 [5], IAEA-TECDOC-1791 [6], IAEA-TECDOC-626 [7], WENRA Booklet (March 2013) [8] , WENRA Revised Reactor Safety Reference Levels (with Fukushima Lessons Learned) [9] . The origin of the new EUR version was EUR Rev. D [10], however the comparison of IAEA SSR 2/1 [11] and EUR Rev. C [12] was also taken into account. The Rev. E also reflects the legislative directives of EURATOM [14-15].

The elaboration of EUR rev. E was performed by the Topical Working Groups (TWG) who prepared the drafts of the chapters. The Technical Coordination Group (TCG) was assigned to perform a technical review of a draft chapter. The overlapping and synchronization between the TWG topics was also discussed by the TCG. The Administration Group (AG) focused on the content of the draft chapters in terms of quality, clarity and expressiveness. In addition, AG was in charge of validating the main technical evolutions of the chapter and the technical topics for which a consensus has been found during the TCG review. The Steering Committee was the highest level of the management team who served as a final approver.

Regarding the formulation of the requirements the following criteria were applied. First, the requirements have to be generally and functionally phrased. Second, the requirements of IAEA and WENRA have to retain their original wording as widely as possible for traceability reasons. Third, the EUR terminology is preferred. However, where contradictions occurred among the requirement sources, the IAEA terminology was used. And fourth, the number of the requirements has to be similar to the number of requirements in EUR Rev. D. Elimination of requirement redundancies and the merging of requirements were allowed, while restructuring the requirements for better readability was highly encouraged.

3. CHANGES IN BASIC SAFETY CONCEPT

3.1. Levels of defence in depth and design approaches

Defence in depth (DiD) concept in EUR Rev. E corresponds to DiD concept Approach 1 in IAEA-TECDOC-1791 see in Sec. 4.2.3.1 of Ref. [6]. The main problem to be solved was that the off-site emergency response needs to be assigned to level 5, however, actually 6 levels could be identified due to the analysis results of the investigation of postulated initiating events (PIE). More precisely *single failures*, *multiple failures* or *complex sequences* (without core melt) and *core melt accidents* form three different categories due to their nature, but they need to be assigned to DiD levels 3 and 4. In addition, there is a fundamental difference between the design approaches of systems coping with Design Basic Conditions (DBC) and systems coping with Design Extension Condition (DEC), in which Engineering design rules 1 (EDR 1) and Engineering design rules 2 (EDR 2) shall be applied according to EUR Rev. E, respectively. Design approaches EDR 1 and EDR 2 are superimposed to the DiD concept. Furthermore, WENRA prescribed O1, O2 and O3 radiological safety objectives [8], of which O2 refers to accidents without core melt. Taking into account the nature of initiating event categories, the engineering design rules and the radiological safety objectives, in EUR Rev. E DiD level 3 is split into sublevels 3a and 3b, and level 4 is kept for core melt accidents.

- **Sublevel 3a**. DBC-3 and DBC-4 events belong to this level, EDR 1 is applied (Safety systems are credited), and O2 objective needs to be fulfilled.
- **Sublevel 3b**. DEC-1 events (without core melt) belong to this level, which are named *Complex Sequences*. EDR 2 is applied (Safety features for DEC) and O2 objective needs to be fulfilled.
- **Level 4**. DEC-2 events (core melt is supposed) belong to this level, which are named *Severe Accidents*. EDR 2 is applied (Safety features for DEC) and O3 objective needs to be fulfilled.

With such an approach the 5 levels of defence are kept, sublevel 3b is assigned to DEC, and the WENRA approach about radiological consequences is integrated. Nevertheless, the failure of a safety system at level 2 leads to sublevel 3b, so no level is bypassed as written in [8].

Design concepts of redundancy, diversity, independency and separation are emphasized in the requirements. EDR 1 and EDR 2 differ from each other in many points, for example in the application of single failure criterion, analysis method (conservative or best estimate) etc. However, there are no requirements regarding the safety demonstration of independence between systems and system components (SSC) credited at different DiD levels, because they cannot be consistently formulated together with the IAEA requirements and such an aim seems to be economically not feasible.

3.2. Introducing the term "practical elimination"

The term "practically eliminated" was mentioned for the first time in INSAG-12 [12] in 1999. For new NPPs there is a limit on economic, societal and environmental risks that is acceptable concerning severe accidents and releases. Sequences that are to be practically eliminated are the ones that are leading to early or large releases. In EUR Rev. E the latter limit is interpreted as the sequences that exceed the Criteria for Limited Impact (CLI). The identification of accident sequences should be based on deterministic analysis supported by engineering judgement, and on probabilistic assessment. The practical elimination of an accident sequence can be based on either the physical impossibility of the sequence to occur, or on a demonstration about that the sequence is extremely unlikely to arise with a high degree of confidence. In both instances any claims should be well substantiated and justified. In EUR Rev. E the requirements of practical elimination of sequences leading to early or large releases are included. The requirements are fully compliant on a principal level with both IAEA SSR 2/1 [3] and WENRA Position 5 [8]. The application of practical elimination is an enhancement of the strength of DiD level 4, as it implies that more justification is required for sequences that were previously considered beyond design (cf. Sec. 4.3.).

3.3. Description of acceptable deterministic safety analysis methods

Deterministic safety analysis methodologies have been harmonized with IAEA SSG-2 Rev. 1 guide [4]. There are two main aspects of changes. One is the introduction of three graded methodology options for DBC 2-4, while the other is the rules of best estimate analysis in DEC. In DBC 2-4 the conditions of "*best estimate computer codes with conservative and/or realistic input data*" coupled with uncertainty analysis is the most preferred method, however, conservative approaches both in input data and in the codes themselves could be applied. This concept shows discrepancy with SSR 2/1 Rev. 1 Req. 5.26 [3], which asks for conservative approach. EUR position, similarly to the one in SSG-2 Rev. 1 [4], is that inserting as much information as possible into the analysis induces less difficulty than the conservative manner.

In WENRA report [8] best estimate methodologies are favourable for DEC, however with less stringent rules than in DiD level 3a, but the robustness needs to be shown. This opinion is integrated into EUR Rev. E, requiring best estimate analysis together with sensitivity analysis. Such approach in DEC 1 aims at preventing core melt at adequate level of confidence and at assuring that "*there is adequate margin with regard to cliff edge effects*", while in DEC 2 the robustness of the analysis is emphasized by applying best estimate approach in full range of the analysis (hypothesis, input data, initial and boundary conditions, code algorithms etc.). A new definition of best estimate analysis has also been introduced for clear understanding. Nevertheless, requirements dealing with break preclusion and leak before break have been also introduced both in the scope of deterministic safety analysis and in the scope of design approaches.

3.4. Improvements in the terminology

The definitions are of crucial importance in a regulatory document which comprehends a whole design of new NPPs. Changing the meaning of requirements induces changes in definitions in most cases. Terminology has undergone a major revision based on IAEA and WENRA used terms. Important modifications were that the term accident conditions has been extended to DEC, and operational states has been extended from normal operation to anticipated operational occurrences. New definitions have been formulated concerning the topics of hazards, security, power supply, SSCs, DiD, loss ultimate heat sink, EDR and safety classification.

4. CHANGES RELATED TO THE TOPIC OF FUKUSHIMA LESSONS LEARNED

4.1. Strengthening plant autonomy

Implementing the Fukushima lessons learnt autonomy requirements were strengthened significantly.

4.1.1. *Non-permanent equipment*

No on-site non-permanent equipment shall be credited (irrespectively whether it is light or heavy) during the first 72 hours following the accident initiation (it was requested only for light mobile equipment and for DBC in EUR Rev. D). Use of light non-permanent equipment after 24 hours in case of complex sequences can be an exception when safety classified. Off-site non-permanent equipment can be credited after 7 days only (in EUR Rev. D the limit was 72 hours). Notion of "non-permanent" is a refinement of the earlier used "mobile" equipment recognising that some equipment on the location of its operation requires certain action enabling of its functioning and this is not considered as mobile, which is assumed to be transported to its functional place in certain designs. Also, precise definition of "light" and "heavy" non-permanent equipment is added.

4.1.2. *Ultimate heat sink and electric power supply*

Loss of the ultimate heat sink is one of the main outcomes of Fukushima Lessons Learned. It is important to take into consideration that the loss of the ultimate heat sink may not have only been caused by unavailability of water intake but also by the failure of the system that transfers the heat to the sink. The heat removal capability of UHS should include not only the heat generated by the radioactive decay, but also the heat from system, structure and components which are required to remain operable such as spent fuel pool. This capability should be ensured for 7 days without off-site support in all plant states (it was required for 72 hours in EUR Rev. D). Similarly, plant independency from the off-site electrical power supply needs to be assured for 7 days (instead of 72 hours in EUR Rev. D). It is also required, that raw water reserves shall ensure for 30 days autonomy in case the UHS is provided by water reservoirs. Moreover, these autonomy objectives are also valid for the conditions following the occurrence of a rare and severe external hazard (RSEH) whereas it was not specifically required in Rev D (cf. Sec. 4.2.).

4.2. Extension of the scope of the external hazards to be considered

Although, the avoidance of the "cliff edge effect" was a definite design objective in the EUR Rev. D when the possible design basis hazards were considered too little help was provided to the designer about how it should be taken into account. In order to implement the lessons learnt from Fukushima Daiichi accident in this aspect two levels of severity were defined for the relevant external hazards: the design basis external hazard (DBEH) and the rare and severe external hazard (RSEH). Specific requirements were formulated then in the EUR Rev. E about how to take this into account in the Design in terms of meeting autonomy objectives, provision of environmental categorization, safety analysis rules, acceptance criteria, to name just a few.

Moreover, as an addition to the EUR Standard Design DBEH values and RSEH magnitudes were also defined for a "typical" European site. The determination of this RSEH set of values is based on statistical analysis of the country specific values provided by the member states. See some examples below:

— Seismic design level value for RSEH (peak horizontal ground acceleration) is 1.5*DBEH, where PGA_{DBEH} = 0.25 g;
— Tornado value for DBEH is 65 m/s, for RSEH it is 84 m/s;
— Cooling (sea) water temperature for DBEH: - 0.5˚C to + 30˚C, for RSEH: - 0.5˚C to + 32˚C.

Specific attention was also paid to the hazard analysis of multiple unit sites. It is now required to take into account that external hazards would probably affect several or even all units on the site simultaneously.

4.3. Spent fuel pool cooling

The Fukushima Daiichi accident highlighted the need of robust design of spent fuel pools (SFP) also. This implies that single initiating events, multiple failure events, and internal hazards as well as external hazards should be properly addressed. In particular, the structural integrity of the spent fuel pool needs to be ensured with adequate margin in case of external hazards. Several improvements have been carried out concerning the SFP cooling taking the international recommendations into account. It is prescribed, that during the identification of *Complex sequences* failures of SFP systems and system components need to be considered. As the spent fuel could be a potential contributor to early and large releases, the safety level of SFP design increased from two points of view.

On the one hand, the uncovering of fuel assemblies needs to be practically eliminated. On the other hand, failure of containment integrity following the failure of fuel in spent fuel pool also needs to be practically eliminated.

5. RENEWED SAFETY CLASSIFICATION

The aim of the revision of the *Safety Classification* section of Chapter 2.1 EUR Rev. D was to give a comprehensive and systematic approach to safety classification and environmental categorization of SSCs, which is also harmonized with the IAEA Specific Safety Guide No. SSG-30 [5], and reflects related Fukushima lessons learnt. In addition, it specifies basic engineering rules proportionate to the safety classification of SSCs. It is supposed that the Designer will apply the recommended methodology otherwise it shall demonstrate the equivalency of its safety classification system.

The formerly used safety function categorization was redefined using the introduction of four factors for safety significance analysis. As a consequence of this new approach direct link between safety function categories and safety classes of the SSCs performing them could be established (three safety classes are defined whereas there were only two in EUR Rev. D).

The timescale within which the safety function is required to perform is shown in the Fig. 1. below. Introducing a new threshold value - 7 days after which non-categorized functions can be credited - strengthens the robustness of the plant design and meets more demanding autonomy objectives as well (cf. Sec. 4.1.).

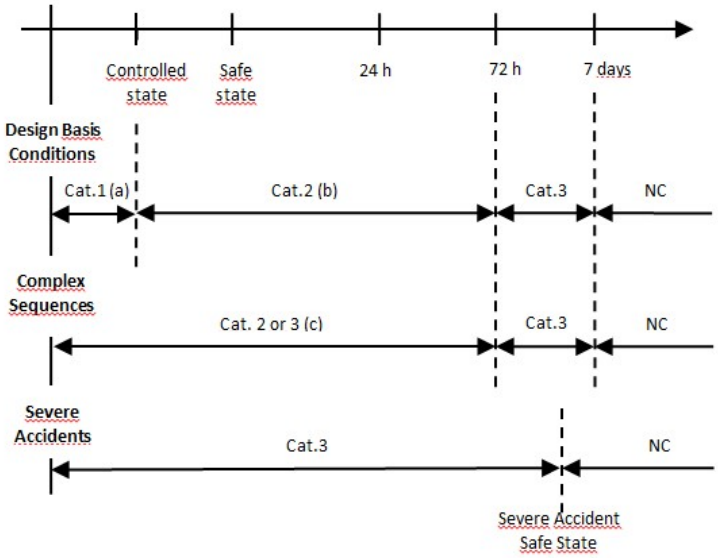

FIG. 1. Timescale of safety functions
(a) Cat.2 functions can be credited if it is demonstrated that the consequences of their failure are of "medium severity"; (b) Cat.3 functions can be credited if it is demonstrated that the consequences of their failure are of "medium severity"; (c) Cat.2 if the function is designed to provide a backup of a Cat.1 function; Cat.3 if the function is designed to provide a backup of a Cat.2 function.

Notion of design provisions was also introduced which simplifies the safety classification of those SSCs which are designed for use in normal operation and on the reliability of which plant safety is highly dependent. For the safety classification of design provisions categorization of their function is not required, safety class can be determined on the basis of the consequences of their failure. According to the new revision the safety classification, which is based primarily on deterministic methods can be refined further by in depth functional analysis based on engineering judgement and also probabilistic methods for verification. It is worth to note, that probabilistic safety analysis (PSA) has been updated based on IAEA SSG 3 and 4.

Important value is added in subsection of the environmental categorization. The recommended approach gives flexibility to the Designer to define those hazards which are relevant to the given site and elaborate specific environmental categorization for each significant hazard. Also, the recommended approach facilitates integrated application of safety classification and environmental categorization. Environmental condition resistance levels were introduced enabling the Designer to take into account the role of SSCs during and after a hazard and events leading to DBC2-4 and DEC. Differentiated engineering rules apply to SSCs credited for DBC2-4, complex sequences and design basis hazards, to SSCs credited for severe accidents and RSEH. Also, distinction is suggested when applicable requirements are defined for SSCs that must remain functionally operable, must not operate spuriously and for those which must retain its pressure boundary function and/or leaktightness.

6. OTHER SPECIFIC TEXT IMPROVEMENTS

6.1. Physical protection

The Security requirements in EUR Rev. E are mostly summarized in two chapters regarding general considerations, including intentional aircraft crash issues and plant layout arrangements issues, respectively. The new and updated requirements are primarily based on the recommendations of IAEA-NSS-13 (INFCIRC Rev. 5) [13] and WENRA Report on Safety of new NPP design, Position 7: *Intentional crash of commercial airplane* [8]. Although the requirements on the security of NPPs are primarily in the responsibility scope of national regulators (definition of Design basis thread) and provided by Security forces of States (Secret services), EUR Rev. E provides some requirements on categorization of Nuclear material (Category I, II and III) and Nuclear facilities and on the distribution of Nuclear facilities into Limited access area, Protected Area, Inner area and Vital areas according to the categories of nuclear material or attractiveness of sabotaging nuclear facilities. The physical protection of the plant relays on three fundamental functions (detection, delay, response). Each function uses DiD principle and applies a graded approach. The other currently important issue EUR Rev. E dealt with extensively is the intentional crash of commercial airplane.

6.2. Interconnection between chapters

There are many connection points and overlapping topics between chapters which cannot be perfectly separated. Nuclear safety aspects are important in every stage of the design so necessary to demonstrate it in several chapters. Chapters impacted by the requirements of chapter 2.1 were chapter 2.2 (fuel requirements from SSR 2/1), chapter 2.4 (specific hazards requirements), chapter 2.7 (components specific requirements from SSR 2/1), chapter 2.8 (SFP, emergency power supply) and so on. Also the formal and linguistic properties of the document needed to be in accordance with each other. While the TCG carried out the synchronisation of the technical issues, the AG done the latter.

A section called Design of Specific Systems was established to present safety requirements for specific systems very important in terms of nuclear safety. This way, safety requirements for these systems can be found together in one place making the requirement system more transparent and design works less difficult. These sections contain many citations pointing to other chapters, where the other requirement concerning the given system is described. The section prescribes requirement for a wide variety of systems from the reactor core and safety systems through I&C systems to electrical and radioactive waste treatment systems. Furthermore this section together with the other connecting chapters have been revised to be fully in line with ENSTO-E Grid Code and IEC Standards such as 61513, 60880, 62138, 61226.

7. CONCLUSION

EUR Rev. D has undergone a major revision, and a new version of EUR requirements named EUR Revision E is under publication. The new version is intended to be up to date, therefore new versions of IAEA documents at their early draft stages were also considered. To include the lessons learned from Fukushima Daichii accident was particularly emphasized. The latter induced major changes in plant autonomy. Nevertheless, the Safety Classification concept was deeply investigated and harmonized with IAEA-SSG-30. The Defence in Depth concept has also been revised according to IAEA-TECDOC-1791 and the WENRA report [8], where the nature of events, designing rules and radiological consequences have been taken into consideration.

The grandiose work was planned carefully aiming both at considering the opinion of each participant and at improving the content of Revision D without the loss of any safety related aspect. The state of the art nuclear technologies were taken into account during the formulation of the requirement but keeping in mind the feasibility from the economic point of view. Revision E is expected to be a valuable standard for new NPP designers with the highest possible level of nuclear safety.

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CONSIDERATIONS ON HARMONIZATION OF SAFETY ASSESSMENT OF NEW REACTORS' DESIGN

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Abstract

High-level safety objectives and requirements between countries are comparable (notably in European countries) and in general do not show important gaps. However, the application of specific national regulatory documents and the impact of specific safety assessment processes can have an important impact on the design of a reactor. Indeed, even with similar intention and definitions, a regulation in a country can lead to a different design than in another country due to scientific cultural background and/or NPP experience feedback of each country. As a matter of fact, several differences could be highlighted between on-going EPR projects, coming from the safety assessment approach of the national Safety Authority more than differences coming from strict regulatory requirements. We could for example present the postulated failure of safety digital I&C that leads to require a back-up system independent and diversified. In some countries, the term 'diversified' allows to rely on another type of digital I&C, in other countries it is understood as being a non-digital I&C system. After developing typical examples, this paper will discuss how harmonization of practices is desirable to increase standardization of reactors' design that will facilitate licensing processes and bring benefits to safety.

1. INTRODUCTION

High-level nuclear safety objectives and requirements have reached a certain level of harmonization and are comparable between countries (notably in European countries). Nevertheless, going deeper in the analysis of the methods and practices followed by different national regulatory bodies to evaluate the respect of similar high-level safety requirements, puts in evidence that the safety evaluation culture of each country has an impact on the final safety evaluation. The objective of the article is to point out some differences in the safety evaluation approach followed by different regulatory bodies, but not to express a judgment about these approaches. The objective is to highlight, through examples based on the current EPR projects licensing feedback, that it still exists areas where safety evaluation approach and interpretation of safety concepts are not harmonized.

2. TYPICAL EXAMPLE OF DESIGN DIFFERENCES COMING FROM COUNTRY SPECIFIC' SAFETY AUTHORITY ASSESSMENTS

2.1. I&C architecture

The I&C architecture differs from one EPR project to the other and this paragraph focuses only on the difference in the implementation of the back-up of the I&C Protection System.

On all the EPR projects there are two I&C digital platforms. An I&C platform is a set of equipment compatible between them. The safety I&C platform is an AREVA equipment ("TelepermXS") specifically designed according to nuclear safety requirements having the highest safety classification. The Protection System is one of the systems implemented in the safety I&C platform and houses the highest safety functions. The operational I&C platform is a Siemens equipment ("SPPA-T2000") designed as per good industrial practices with a good experience feedback from the industrial sector. I&C is involved in the safety design and demonstration for all the design conditions of the safety referential, from the Design Basis Conditions (categories 2 to 4) to the Design Extension Conditions A (prevention of core-melt) and B (mitigation of core-melt). For the Design Extension Conditions A resulting from Design Basis Conditions of category 2 cumulated with the loss of the Protection System (covering Anticipated Transient Without Trip cases), the safety demonstration proposed by EDF and AREVA credit the availability of the Safety Automation System implemented in the operational I&C platform. This system has been specifically designed and validated to meet the requirements of an intermediate safety classification level (in particular IEC 62138 [1] and the French Fundamental Safety Rule II.4.1.a [2]). To justify this safety position EDF & AREVA performed very refined studies to show that the design and the verification and validation process of the overall I&C architecture conform to the applicable nuclear safety requirements. The French Safety Authorities evaluation concluded that such I&C equipment and architecture areable, through the way the different equipment are grouped into systems or at the contrary are independent (either through physical separation, electrical separation, technology diversity or by logic), to ensure that no design condition would lead to a situation not covered by the safety demonstration. For the OL3 (Finland) and HPC (UK) EPR projects, the regulatory bodies considered that it is not possible to credit the Safety Automation System in case of the loss of the Protection System. Indeed, whatever the reliability of the Safety Automation System and its design difference with the Protection System, the complete loss of all the digital I&C systems had to be postulated. The argument of physical separation between the Safety Automation System and the Protection System as well as the difference in terms of design is not considered sufficient by the UK and Finnish regulatory bodies to conclude that the I&C reliability is at a sufficient level to meet Design Extension Conditions A criteria. Then, to meet the safety criteria of these operating conditions, an additional I&C system had to be implemented called “hardwired Backup system” in Finland or “Non Computerized Safety System” in UK whose design is based on a non-digital technology.

This example shows that the notion of diversity and overall reliability of a complex architecture like the I&C is not evaluated with the same criteria even in countries having the same overall safety requirements. This example shows that, even though the treatment of common cause failure caused by software within digital safety systems has already been examined within a specific MDEP working group [3], this topic could be proposed for further harmonization between countries to get a better mutual understanding of the rational of the situations to be covered.

2.2. Identification of the list of the Design Basis Conditions and Design Extension Conditions

The identification of the list of the Design Basis Conditions and Design Extension Conditions differ from one country to another, and this paragraph will only focus on a specific aspect.

For all EPR projects, the Design Basis Conditions of categories 2 to 4 are identified to determine, on the basis of a conservative approach, provisions enabling to limit the effects of the selected initiating events. Then, depending on their occurrence probability, these events are distributed in three categories (2 to 4) and studied with conservative assumptions as for example the application of an additional single failure, and the consideration of the preventive maintenance leading to the systematic unavailability of equipment possibly in maintenance when the event occurs. Then, to ensure fulfilment of the probabilistic core melt frequency target, more complex accident sequences are taken into account and grouped in a category called Design Extension Conditions A, which cover notably Design Basis Conditions with additional failures. These Design Extension Conditions Aare evaluated using less stringent rules and requirements than the ones used to evaluate the Design Basis Conditions. As for example, Design Extension Conditions A are evaluated without taking into account an additional single failure (as the considered event is already a multiple failure event). This approach is commonly adopted even if some differences exist in the level of conservatisms applied in terms of coverage of the main transient parameters.

For FA3 (France), TSN (China) and OL3 (Finland) EPR projects, the Design Basis Conditions are based only on single initiating events, and the multiple failures are only addressed in the Design Extension Conditions A. For HPC (UK)project, the identification of the Design Basis Conditions is performed using mainly the occurrence probability of the initiating event, independently of its nature (single or multiple events). As a consequence, some multiple initiating events must be studied following the stringent conservative rules of the Design Basis Conditions, such as the additional single failure and the combination of simultaneous preventive maintenance. This specific methodology of identification of the Design Basis Conditions leads, for example in the design of the HPC HVAC system, to implement additional safety chillers and to upgrade the classification of two non-classified preventive maintenance HVAC trains. (Indeed, the loss of HVAC in 2 safety divisions as though a multiple initiating event had to be analysed as a DBC event)

This example shows that the identification of Design Basis Conditions (depending on the definition retained for the associated initiating events, only single events or possibly multiple events, which corresponds to different approaches between deterministic and probabilistic aspects) could be proposed as a topic for harmonization between countries.

2.3. Rules used to evaluate Design Basis Conditions

This paragraph addresses some specifics about differences between countries in the rules used to evaluate the Design Basis Conditions of category 2 (also called Anticipated Operational Occurrences).

For the FA3 project, the French practice imposes to credit only the safety grade systems (with exceptions concerning some main plant controls). Then, taking into account the required conservative assumptions (additional single failure, preventive maintenance considered simultaneously with the event occurrence) leads to transients that significantly deviate from the ones that would normally result from the initiating events.

For OL3 project, the Finnish practice allows to credit the operation of the main plant controls and limitations in addition to the safety grade systems. That allows sticking to more realistic transient evolution. In addition, the Finnish practice requires to complement the analyses by addressing the same initiating events while crediting only safety grade systems and to meet safety criteria corresponding to Design Basis Conditions of category 3, instead of category 2.

The difference between the two approaches leads to very different transient evolutions for the same initiating events and for the same reactor design. It can artificially lead to conclude that safety margins are considerably different from one EPR project to the other, although differences in the results and consequences of the transients are often only due to different levels of conservatism taken into account for the same criteria and physical phenomena. This example shows that the rules to be used to analyse Design Conditions could be proposed as a topic for harmonization between countries.

3. CONCLUSION

High-level nuclear safety objectives and requirements have reached a certain degree of harmonization and are comparable between countries (notably in European countries). As a matter of fact, the IAEA Safety Requirements Series, the WENRA reference levels, and the will to compare safety evaluations performed in different countries for the same reactor design through the MDEP organisation, help harmonizing the safety assessment. Moreover, the World Nuclear Association (WNA) established the Cooperation in Reactor Design Evaluation and Licensing (CORDEL) Working Group with the aim of stimulating a dialogue between the nuclear industry (including reactor vendors, operators and utilities) and nuclear regulators (national and international organizations) on the benefits and means of achieving a worldwide convergence of reactor safety standards for reactor designsm [4].

Nevertheless, it appears that the application of specific national regulatory approaches and safety assessment processes can have an important impact on the design of a reactor. As illustrated in the paragraphs above, the licensing experience feedback of the current EPR projects show examples of design modifications that had to be implemented depending on the understanding/interpretation of a safety requirement and also on the importance given in terms of probabilistic and/or deterministic rules. The differences in interpretation of a same safety requirement can be explained by scientific cultural background or specific Nuclear Power Plants experience feedback.

These examples show that even if a number of organizations already promote the cooperation between Safety Authorities and nuclear industry players (IAEA, WENRA, MDEP, WNA) and played an important role to improve the standardization of the regulatory framework in force in each individual country, this cooperation needs to be carried on. Indeed, even keeping specific national regulatory frameworks, a common understanding of the safety evaluation and safety concepts and requirements would be beneficial both for lighting the regulatory bodies workload and the predictability of the licensing of a reactor design, and of the cost to completion for new nuclear projects.

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IMPLEMENTATION DESIGN EXTENSION CONDITIONS (DECS) WITHIN THE SCOPE OF RISK MANAGEMENT REGULATORY FRAMEWORK

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Abstract

How to balance the approach for DEC consideration between NRC and European? It is benefit for the design approach for the design and assessment of new nuclear power plant to form the common agreement about DEC. West European Nuclear Regulation Association (WENRA) considered the DEC and assessed those conditions by deterministic analyses method as well as the updated version of SSR2/1[1] did. According to the requirement of EUR, the off-site release targets for DEC (complex sequences) should meet the acceptances criteria relevant to Design basis category 4 condition (DBC). Meanwhile, American NRC made a decision to maintain the existing regulatory framework, not plan to establish a formal design basis extension category, for the nuclear power reactor safety program area on March 9, 2016 as SECY-15-0168 [4]. The category is still considered by implementation of a risk management regulatory framework. For example, the selected events such as SBO, ATWS are considered in the deterministic approach, other DECs (complex sequences) and severe accidents are analyzed using the probabilistic approach. Although the requirements relate to DEC is different from SSR-2/1, the DECs could also be fully evaluated following the US NRC regulatory framework. The current risk management regulatory framework can meet the requirement of DEC with a little improvement according to the assessment.

1. BACKGROUND

Nuclear safety especially severe accidents risks are of great concerns of nuclear power plant. Design consideration of severe accident prevention and mitigation is generally required by various nuclear safety authorities worldwide. However, those requirements related to severe accidents consideration are somewhat different from country to country. Recently the International Atomic Energy Agency (IAEA) updated and published a safety code on Specific Safety Requirement of Nuclear Power Plant Safety: Design (SSR-2/1) [1]. Meanwhile the Chinese National Nuclear Safety Administration (NNSA) also revised and updated the safety code on Requirement of Nuclear Power Plant Safety in Design (HAF102) [2]. In these two codes, both IAEA and NNSA established some new requirements, among which two are of great concern, one is Design Extension Conditions (DEC) for consideration of those conditions traditionally called Beyond Design Basis Accidents(BDBA) in design of nuclear power plant, another is requirement of practically elimination of large release of radionuclide.

In the other hand, improvement activity 1 in SECY-13-0132 [3] recommended that the NRC adopt a new term – “design-basis extension” – to define and describe the events and requirements for nuclear power plants that have typically been characterized as “beyond-design-basis” events and accidents. But the NRC staff determined that a new category of events should not be established in SECY-15-0168 [4].

Some new design nuclear power plants are being designed by SNERDI and their documentation was prepared mainly in accordance with US NRC RG-1.206 [5] (Combined License Applications for Nuclear Power Plants). This Guide does not use the concept of DEC. So a gap analysis between current safety documentation and the requirements of DEC is being carried out.

2. THE REQUIREMENTS OF IAEA TECDOC-1791

IAEA TECDOC-1791 [6] was issued in 2016. This document is to provide insights and approaches in support of the practical application of the new crucial requirements introduced in SSR-2/1 [1] and subsequently reinforced in SSR-2/1 [1]. According to this document, the requirements of DEC mainly include following elements:

- Definition of DECs;
- Determination of DECs;
- Acceptance criteria for DECs.

2.1. Definition of DECs

According to SSR-2/1 [1], DECs are: "Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with core melting."

TECDOC-1791 [6] also points out "DECs are those conditions not included in the DBAs, and which have a frequency of occurrence that cannot be neglected and in some cases comparable with the frequency of some DBAs."

Although current documentation does not include DEC, analyses have been provided in the current safety documentation for the conditions exceeding DBAs which is similar to DEC. This will be discussed in the next section.

2.2. Determination of DECs

The DECs can be divided into DECs without significant fuel degradation (DEC-A) and DECs with core melt (DEC-B) according to SSR-2/1 [1].

For DEC-A, at least three types of DECs can be considered according to the postulated assumptions:

- Very unlikely events that could lead to situations beyond the capability of safety systems for DBAs.
- Multiple failures (e.g. CCFs in redundant trains) that prevent the safety systems from performing their intended function to control the PIE.
- Multiple failures that cause the loss of a safety system while this system is used to fullfil the fundamental safety functions in NO.

Typically, it may include:

- ATWS;
- SBO;
- Los of core cooling in the residual heat removal mode;
- Extended loss of cooling of fuel pool and inventory;
- Loss of normal access to the ultimate heat sink;
- Total loss of feed water;
- LOCA plus loss of one emergency core cooling system (either the high pressure or the low pressure emergency cooling system);
- Loss of the component cooling water system or the essential service water system (ESWS);
- Uncontrolled boron dilution;
- Multiple steam generator tube ruptures (MSGTR) (for PWRs);
- Steam generator (SG) tube ruptures induced by main steam line break (MSLB) (for PWRs);
- Uncontrolled level drop during mid-loop operation (for PWRs) or during refueling.

In fact, all of these DECs are analyzed in current risk management regulatory framework. For example, ATWS, identified in 10 CFR50.62 [7], usually is presented in chapter 15.8 of preliminary/final safety analysis report (P/FSAR). The other example is that SGTR induced by MSLB is considered in PSA, which was included in chapter 19. So does SBO.

In addition, some plant specific initiating events such as spurious operation of automatic depressurization system, which was induced in some passive NPPs, are also included in the analysis, which can be treated as DECs too.

For DEC-B, it is necessary to select a representative group of severe accident conditions to be used for defining the design basis of the mitigatory safety features for these conditions. In general, it is similar to the severe accidents in current risk management regulatory framework.

Current risk management regulatory framework has included a detail analysis for severe accidents. PSAR chapter 19.2 of an example NPP is shown as follows:

c) The sub-section of chapter 19.2 presents the selection of significant accident sequences. The accident progressions of these significant accidents are analyzed using severe accident analysis code, including in-vessel and ex-vessel melt progressions. Key events evaluated for the in-vessel melt progression are core uncovery, core damage, molten core relocation to lower plenum, hydrogen combustion, in-vessel steam explosion and RPV failure. Key events evaluated for the ex-vessel melt progression are MCCI, DCH, ex-vessel steam explosion and long-term containment pressurization. The severe accident mitigation measures are described in sub-section of chapter 19.2, including in-vessel retention (IVR) of core debris, hydrogen control, mitigation of high-pressure core melt ejection (HPME), containment pressurization from decay heat, molten core concrete interaction (MCCI) and steam explosion.
d) A more specific example is given below. The evaluation of hydrogen control is described in another sub-section of chapter 19.2. Hydrogen generation and mixing are analyzed for typical severe accident classes with igniters unavailable, in order to determine probable regions of hydrogen risk. In order to validate the effect of hydrogen control system, a bounding case, initiated by large break LOCA with failure of accumulator, generating the most hydrogen is selected and analyzed also. 100% cladding of active fuel is oxidized, which is mainly occurred during reflooding. According to the analysis results, igniters can control hydrogen concentration effectively, thus the design of hydrogen control system is effective.

As mentioned above, current risk management regulatory framework has included almost the DECs should be considered although the concept of DEC is no appeared.

2.3. Acceptance criteria for DECs

SSR-2/1 [1] sets out the general requirement for DECs (Reg. 20) where it states that "A set of design extension conditions shall be derived on the basis of engineering judgment, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant 's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures".

Base on this, the Appendix 2 of TECDOC-1791 [6] gives examples of acceptance criteria, as Table 1 shown.

In addition, safety features for DECs shall as far as is practicable be independent of safety systems and the integrity or operability of equipment for DECs shall be justified under the DECs environmental conditions. In current risk management regulatory framework, the severe accidents with the phenomena challenging the containment integrity are analyzed in chapter 19 of safety analysis report.

PSAR chapter 19.2 of an example NPP is shown as follows: Sub-section 19.2.2 of chapter 19.2 provides the principles and methods for sequence selecting together with 12 accident categories, and different scenarios are determined for different purpose by combination of the probabilistic and deterministic and the engineering judgment. The hydrogen risk and the effectiveness of the hydrogen control system were assessed. The qualitative analysis of each category are exhibited and the results is that the containment cooling system provides well mixing in the containment for the open compartment and the hydrogen risk is restricted in the confined compartment because of the high steam concentration. Additionally, the hydrogen control system is evaluated by the amount of hydrogen that generated from the 100% active Zriconium oxidation, and the results show that the ignition system can effectively eliminate the hydrogen risk. The containment pressurization analysis without containment cooling system water cooling was performed, and the result shown that the failure probability in the 24h is less than 0.01% and the vent setpoint is reached in 37h, there is at least 4h left for the operator to consider before it reaches 5%. The MCCI analysis without water circulation was performed for the consideration of basemat melt-through challenge and pressurization threaten challenge. The results are that in the first 24 hours, the cavity basemat penetration or containment over-pressure will not occur, and the containment integrity will not be challenged with plenty margin. All the above evidence shows that the containment will keep integrity under the conservative condition after accident in 24h or longer and can reasonably deduce that it can maintain in the long term.

Moreover, the assessment of severe accident equipment survivability, which is required by risk management regulatory framework, is similar to those of DECs.

Based on the above explanations, the risk management regulatory framework is similar to those of DECs. Furthermore, if the frequency of the single selected severe accidents causing large radioactive release is lower than 1E-7/y, these conditions can be treated as “practically eliminated” conditions.

In general, the analysis of current risk management regulatory framework almost covers the acceptance criteria for DECs. But something should be done to indicate that the requirements of DECs have been considered:

— Give an explicitly description about compliance to DEC acceptance criteria in the safety analysis report.
— Make sure that all of the equipment for DEC is qualified based on the equipment survivability assessment during severe accidents.

3. CONCLUSIONS

In the future, the design of Chinese nuclear power plants should consider the requirement of DECs in SSR2/1 [1] and HAF102 [2] but it is not necessary to make a big change to current risk management regulatory framework because most contents of DECs have been covered by PSA or SA. The requirement of DECs can be met through supplementing some contents about compliance to DEC acceptance criteria and DEC equipment qualification.

4. TABLES

TABLE 1. ACCEPTANCE CRITERIA OF DECS

Level of defence	Objective	Criteria for maintaining integrity of barriers	Criteria for limitation of radiological consequences
Level 3b	Control of DEC-A	No consequential damage of the reactor coolant system, maintaining containment integrity, limited damage of the fuel.	The same or similar radiological acceptance criteria as for the most unlikely design basis accidents.
Level 4	Control of DECs-B	Maintaining containment integrity	Only emergency countermeasures that are of limited scope in terms of area and time are necessary

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CONTRIBUTIONS TO THE HARMONIZATION OF APPROACHES AND METHODS AT THE MEMBER STATE LEVEL

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MIXING OF THE ATMOSPHERE WITHIN THE EPR DESIGN CONTAINMENT IN DESIGN BASIS & SEVERE ACCIDENT CONDITIONS

IAEA International Conference - Vienna - June 2017

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Abstract

This paper presents the outcome of regulatory assessment of the generic EPR design relating to the performance of the atmospheric mixing measures within the containment during design basis and severe accidents. The EPR containment differs from many typical PWRs in that it uses a two room design concept. Equipment rooms immediately surrounding the reactor coolant system are isolated from the rest of the containment. Beyond this inner region, personnel access can be provided during certain maintenance tasks. The EPR design includes features that promote mixing within the containment during accident scenarios. Heat transfer to the containment heat sinks is facilitated by the "CONVECT system", consisting of passive rupture and convection foils, active mixing dampers, and related instrumentation and control equipment. Opening of the foils and dampers is designed to set up circulation patterns in both the accessible and inaccessible areas to increase the heat transfer surface areas in design basis accidents, and promote mixing of hydrogen released into the containment during a severe accident. The performance of the CONVECT system has been the subject of independent confirmatory analyses by participating regulators and their technical support organisations using lumped parameter codes and detailed CFD computer codes. For severe accidents scenarios which remain on the predicted accident progression path, the CONVECT system enables hydrogen released to be mixed efficiently within the containment. Despite some temporary high local hydrogen concentration in some accident scenarios, the containment integrity is not expected to be threatened. This paper has been facilitated by the Multinational Design Evaluation Programme, an NEA initiative set up to enhance standardization of safety assessment of new reactor designs by the national regulatory and safety authorities.

1. INTRODUCTION

1.1. MDEP Objectives

The Multinational Design Evaluation Programme (MDEP) is an OECD NEA initiative set up to enhance standardisation of safety assessment of new reactor designs by the national regulatory authorities in order to:

- Promote understanding of participating countries' regulatory decisions and basis for these decisions;
- Enhance communication among the members and with external stakeholders;
- Identify common positions among regulators reviewing new reactor designs (including EPR);
- Achieve or improve harmonisation and convergence of regulations, standards, and guidance.

1.2. Background

The EPR containment is a new design, different from many typical Pressurised Water Reactor (PWR) containments in that it uses a two room design concept. Equipment rooms immediately surrounding the Reactor Coolant System (RCS) are isolated from the rest of the containment. Beyond this inner region, personnel access can be provided during certain maintenance tasks. Separation is provided by structures and closed portals to minimise radiation exposure in the accessible space areas. During power operation, inaccessible areas inside the containment experience higher temperatures and radiation than accessible areas. The EPR design includes a number of features that promote mixing of the environment within the containment in accident conditions.

The CONVECT system consists of rupture foils, convection foils, mixing dampers, and the related instrumentation and control equipment. Rupture foils and convection foils are placed in the ceiling of each Steam Generator (SG) compartment. There are eight mixing dampers located in the lower part of the containment in the In-containment Refuelling Water Storage Tank (IRWST) wall above the water level.

Opening of the foils and dampers is designed to set up circulation patterns in both the accessible and inaccessible areas to increase the heat transfer surface areas in design basis accidents, and promote mixing of hydrogen released into the containment during a severe accident. The rupture foils are passive components which will burst open if the pressure differential on the foils exceeds a predetermined value. The rupture foils can burst in either direction. The passive convection foils are rupture foils placed in a frame, which is kept in the closed position by a fusible link. Should temperature rise to a set level, the link will melt with a short delay, and the frame will swing open by gravity. The result is that a convection foil will open on a pressure differential and will also open if the local compartment temperature reaches a certain level.

The mixing dampers are either opened manually from the main control room or are automatically activated on an absolute containment pressure signal set just above atmospheric, providing early opening of the mixing dampers for most accident scenarios. When closed, the mixing damper is held in position by an electromagnet against a compressed spring. In case of a power failure to the solenoid of the electromagnet, the spring will drive the mixing damper open.

The performance of the CONVECT system has been the subject of independent confirmatory analyses by a number of participating regulators and their technical support organisations using a variety of analytical tools such as lumped parameter codes and detailed Computational Fluid Dynamics (CFD) codes.

These analyses examined the effectiveness of the CONVECT system in facilitating general mixing in the containment atmosphere and in preventing the containment design pressure from being exceeded for design basis events. The outcome of this work has been shared amongst the participating regulators wishing to assess this feature of the EPR design, leading to improved understanding of its performance in accident conditions.

This paper presents the outcome of the regulatory assessment of the generic EPR design relating to the performance of the atmospheric mixing measures within the containment during design basis and severe accident conditions, which has been the subject of a common position paper[1].

2. SUPPORTING ANALYSIS

2.1. General

The CONVECT system performance has been independently assessed by a number of regulators reviewing the EPR design to evaluate containment mixing in the EPR for two key accident types:

- **Design Basis Accidents**: during which high energy water and steam is released from the RCS where the containment heat sinks (containment wall, internal structures) play a vital role in removing steam from the containment atmosphere. This supporting analysis is primarily to investigate the extent of any potential stratification within the containment which could prevent steam in the containment atmosphere from coming into contact with the relatively cold structures;
- **Severe Accidents**: during which, in addition to two phase flow, hydrogen is also released into the containment. A poorly mixed or stratified containment could allow accumulation of hydrogen in localised areas, detonation of which may put the containment integrity at risk.

The regulators note that the plant designers have conducted extensive analysis of the CONVECT system performance using lumped parameter models and CFD tools. This analysis has been used to demonstrate the role of natural circulation in temperature evolution and profile, evaluate pressure trends and calculate hydrogen concentration and distribution within the containment.

A number of regulators have performed confirmatory studies exploring mixing under design basis conditions using multi-node lumped parameter codes. Whereas, some others have performed studies using both lumped parameter and CFD models for the severe accident scenarios. These studies complement each other in that they explore similar phenomena using different tools and for different scenarios.

These confirmatory analyses and the EPR designer's results have been discussed at the MDEP EPR technical expert subgroups on Accidents and Transients and Severe Accidents.

[1]MDEP Common Position, "Common Position on EPR Containment Mixing", Version 1 (2015) developed by the MDEP EPR "Accident & Transient" and "Severe Accident" Technical Expert Working Groups.

2.2. Design Basis Accidents

The EPR designer has developed a model of the EPR containment using lumped parameter codes for analysis of design basis events. The model represents the different compartments, dome region and includes explicit modelling of the CONVECT system's foils and mixing dampers. The supporting justification argued that the nodalisation approach is sufficiently detailed to permit development of natural circulation patterns in the containment. Both local atmosphere and local wall temperatures are calculated and used in heat transfer predictions. Whilst the design analysis appeared to demonstrate adequate mixing, the two-room containment concept and lack of active containment atmospheric heat removal raised novel issues not studied previously:

- Two-room concept responding as a single volume during the initial pressure rise in accident conditions;
- Effectiveness of open dampers and foils in creating circulation patterns within the containment;
- Effectiveness of passive heat sinks in limiting, and subsequently reducing, containment pressure.

A set of confirmatory calculations were performed using a multi-node APROS model. Heat and mass transfer to the containment internal structures is calculated using natural circulation heat transfer correlations and condensation determined from the heat/mass transfer analogy. The APROS approach is similar to the "Diffusion Layer Model" programmed into GOTHIC and to the heat/mass transfer package in MELCOR.

The APROS study developed a model that included explicit representation of the CONVECT system to predict flow distribution, flow mixing, and heat transfer in the containment following design basis accidents. Sensitivity studies were carried out to evaluate; effectiveness of the foils and dampers, heat transfer assumptions, and the effect of break elevation. A single-node and a 41 cell multi-node model were developed to analyse Large-break Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) accident scenarios and predict peak containment pressure and temperature (P & T).

The sensitivity studies assumed a double-ended guillotine break in the cold leg pump suction line (2A CLB), providing the following useful insights:

- Foils burst in all four SG compartments, and mixing dampers opened within a few seconds of the postulated event. The entire free volume of the containment participated in the initial pressure rise. The first pressure peak was reached within 30 to 40 seconds;
- There was good mixing in most parts of the containment during the first hundred seconds of the accident. After which, the pressures equalised, and most local flows subsided. Mixing was limited in the pressuriser (PRZ) compartment which does not have ceiling foils, and in some equipment rooms;
- The start of safety injection had a very pronounced effect, almost immediately terminating the rise in containment pressure and temperature, followed by a rapid fall in pressure;
- The time history of the analysed LOCA shows two pressure and temperature peaks. The first is determined by the energy stored in the RCS that releases into the containment during the first 40 seconds. The second peak occurs later, dependent on the time at which safety injection starts;
- Should dampers fail to open, peak containment P & T were observed to increase by 55 kPa and 14°C.

The postulated break location for the main analysis was at a low elevation. An additional case was run placing the break in the top node of the SG compartment. The result was an increase of 68.9 kPa in peak containment pressure. This result led to the selection of the MSLB for analysis, as these could occur at higher elevations. Thermal stratification was observed to depend upon the break elevation and was most pronounced between the dome and the lower volumes. Accumulation of non-condensable gases (air, nitrogen) was observed to reduce the rate of condensation on the walls.

Two MSLB confirmatory calculations were performed with the 41-node model. A double-ended break was postulated at the highest point of the steam lines within the containment dome. The purpose of the MSLB calculations was to gain an understanding of flow distribution and mixing, heat transfer in the containment, thermal stratification, and the effectiveness of the CONVECT system. The following points were observed:

- Foils ruptured and mixing dampers opened in the first few seconds of the postulated transient;
- During the first 100s of superheated steam release, warm steam remained in the dome. Some of the air from the dome was displaced into the lower part of the containment through vertical flow paths;

- A clear vertical stratification formed during the short term between the dome and lower parts of the annular region. The stratification subsequently became stronger and lasted for more than 24 hours;
- Over the long term, a circulation pattern formed in the upper part of the containment; however, due to large temperature differences between the upper and lower part of the containment, the buoyancy effect inhibited gas flow into the lower parts of the containment;
- The single-node representation of the dome region prevented the expected circulation and stratification. A more detailed model of the dome would be needed to study this phenomenon;
- Containment pressure peaked around 70 seconds then decreased slowly. Following heating of structures a reduction in the condensation rate, and a second smaller pressure peak, were observed;
- Operation of the CONVECT system produced relatively uniform containment pressure, however it was less effective in setting up circulation patterns between the containment equipment space and the service area which would act to reduce stratification;
- Gas temperatures in the upper regions of the containment were significantly higher for the MSLB than for LOCAs. The difference existed for the duration of the calculation (24 hours).

A MELCOR model was developed based upon the 41-node APROS model, and the predications of this model were compared against results from the APROS and GOTHIC models for the postulated 2A CLB. Similar results to those from both models were obtained, with the EPR designer's multi-node GOTHIC model shown to be conservative in predicting peak containment P & T. The consistency of results between the reactor designer's GOTHIC model, and the independently developed APROS and MELCOR models gives confidence in the representative nature of the predictions.

Confirmatory sensitivity studies were performed on the damper flow resistance and flow area, by varying these parameters. In addition, sensitivity studies were carried out to examine the impact of the nodalisation on the recirculation patterns between different volumes. The results indicated that additional nodes and flow parameter changes each had a minimal effect on the peak containment P & T which remained below that calculated by the EPR designer using GOTHIC.

2.3. Severe Accidents

A series of independent confirmatory studies using both CFD and lumped parameter models have been undertaken by the regulators using the MELCOR, ASTEC/CPA, TONUS and FLUENT codes, to evaluate severe accidents. One of the main objectives of the studies was to evaluate the efficiency of hydrogen mixing measures in the containment and efficiency of recombiners in reducing hydrogen concentrations.

2.3.1. *Pressuriser SB-LOCA*

This study considered a SB-LOCA on top of the PRZ using MELCOR, with failure of the emergency core cooling and containment spray, but with successful partial and fast secondary cooldown and emergency feedwater operation. The results showed similar H_2 concentrations in different rooms, indicating good mixing of the containment atmosphere, when the rupture foils and mixing dampers operate as intended. The exception is the spreading room, which has a single opening, and exhibits a more stable H_2 concentration.

FLUENT simulations giving a more detailed spatial H_2 distribution suggest that burning in the PRZ volume is unlikely. The first prevailing steam release reduces O_2 concentration in the room around the leak and the room below. In the lower parts of PRZ volume there is still oxygen, but H_2 does not spread into this lower region. Due to separation of O_2 and H_2 the recombiners in the PRZ zone are not effective in this case.

Noting that both codes exhibit similar trends, FLUENT simulations predicted higher H_2 concentration than MELCOR at the top of the SG room due to stratification; between 4 - 8%, higher during release peaks. In the dome and lower volumes, the concentration remained below 4%, excluding the plumes rising from the SG rooms. In the dome area mixing is very efficient due to the natural convection caused by cooling structures.

A series of sensitivity studies were performed with a small break LOCA and Loss of Offsite Power. The case has been simulated both with and without the recombiners being credited. Both simulations show that hydrogen released will be mixed quite efficiently around the containment and provided an indication of recombiner performance, which reduced the hydrogen level below 4% within 3 hours in this case.

In summary, the results showed well mixed hydrogen concentrations in most compartments, in agreement with the EPR designer's numerical predictions. The agreement in the outcome of the postulated scenarios using different tools provides confidence in this aspect of the EPR design.

2.3.2. *Cold Leg SB-LOCA*

This study considered a SB-LOCA in the cold leg, postulating failure of safety injection and containment spray systems using a lumped parameter code (ASTEC/CPA) for the early phase to evaluate the pressure, steam/air distribution and the wall temperature profiles, and a CFD code (TONUS) to simulate the H_2 release phase. Local values have been produced to evaluate the containment atmosphere mixing during this latter phase.

Two kinds of scenarios were selected for these studies; with and without core reflooding. In previous studies, it was observed that, for transients with low Mass and Energy Release (MER) in the containment such as SB-LOCA, only a few rupture foils opened even on the break side as this balances the containment pressure. As a result, the assumption that all rupture foils remain closed was considered.

A number of sensitivity studies were therefore carried out to examine performance of the CONVECT system with different configuration of dampers and foils failing to operate as per the design intent.

Scenarios without core reflooding assumed the partial and fast secondary cool down and the emergency feed water system were operational, unlike those with core reflooding which also assumed unavailability of fast secondary cool down. The opening of the primary depressurisation valve was simulated as delayed for the core reflooding scenario, leading to discharge of the accumulators onto the damaged core.

During the core degradation phase, H_2 was released at three different locations: the break and the lower part of two pump rooms via the Pressurizer Relief Tank. A series of sensitivity cases were therefore studied to examine the performance of convection foils in different scenarios for various locations of steam and H_2 release.

Whilst the total mass of H_2 released is similar for both cases, the release rate is much faster for the scenarios with re-flooding. Therefore, for the scenario with reflooding, the impact of recombiners is less significant and the hydrogen mass in the containment is more important.

For the scenario without core reflooding, the rate of hydrogen release is not very high, so the recombiners are effective, and local hydrogen concentrations are unlikely to lead to a risk of flame acceleration.

For the scenario with re-flooding, the rate of H_2 release is high. Despite good mixing, the results show high H_2 local levels at the top of the SG compartments and in the dome for a short period. The CONVECT system has no influence on this phase because even with a large opening at the top of these compartments there is a potential risk of flame acceleration; although the risk from flame acceleration in such infrequent scenarios has been analysed by the EPR designer to demonstrate that the containment integrity would not be threatened.

3. CONCLUSIONS

The independent confirmatory analyses commissioned by the regulators concluded that the CONVECT system is effective in facilitating mixing in the containment atmosphere and preventing the design pressure from being exceeded for design basis events. Temperature stratification is possible for steam line breaks occurring at high elevation in the containment, the occurrence of which does not challenge the design pressure. However, the uncertainties relating to stratified conditions may potentially lead to challenges in temperature qualification of instrumentation within the containment, which must operate following a design basis event.

With regards to severe accidents, the results of the independent confirmatory analysis performed show that for scenarios which remain on the predicted accident progression path, the CONVECT system enables hydrogen released to be mixed efficiently within the containment. Despite some temporary high local hydrogen concentration, the containment integrity would not be threatened.

Overall, the effectiveness of CONVECT system has been confirmed by regulators and their TSOs as well as the EPR designer, and the studies have confirmed that there is sufficient mixing within the EPR containment after an accident to provide mitigation against design basis and severe accidents.

ACKNOWLEDGEMENTS

This paper has been prepared and presented on behalf of the Multinational Design Evaluation Programme (MDEP), Accident & Transient, and, Severe Accident EPR Expert Working Groups. MDEP members contributing and supporting this work include P.R. China, Finland, France, the United Kingdom and the United States. The contribution to the development of this paper is greatly appreciated and the author would like to express his gratitude to MDEP members for their help and support in preparation of this paper.

CONSIDERATIONS ON THE CONCEPT OF REFERENCE PLANT

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Abstract

The concept of the Reference Plant [RP] is mentioned in INSAG-22 and further in INSAG-26 – but is currently not addressed in any of the IAEA Safety Standards. It has primarily been used in two different contexts:

(1) to describe a plant design which has been licensed in an initial country (the vendor country in many cases), which assessment is later relied upon by the regulatory body of another country when licensing that plant design (the 'Regulatory RP' concept); and

(2) to describe a plant already constructed, which is used in a commercial new build arrangement between an operator and a contractor to set contractual baselines (the 'Contractual RP' concept).

Experience has shown that the 'Regulatory RP' concept has been introduced into the regulatory framework of some embarking countries with stringent conditions, which raised difficulties in the practical implementation for specific projects.

Equally, there has been some experience of confusion between the 'Regulatory RP' concept and the 'Contractual RP' concept.

The objectives of the article are to:

— present practical examples where Reference Plants have been used in history;
— seek to clearly define the 'Regulatory RP' concept and the 'Contractual RP' concept;
— highlight and discuss the impacts of specific conditions in national regulatory frameworks seeking to deploy the Reference Plant concept.

1. INTRODUCTION

The concept of reference plant has been used for long in the safety justification and evaluation files of nuclear power stations as well as for contractual purposes between vendors and future operators.

The paper aims at identifying the different uses of the reference plant concept and will focus on the reference plant concept used for regulatory purposes. In that context, the impact of specific conditions in national regulatory frameworks intending to deploy the reference plant concept for regulatory purposes will be discussed.

2. BACKGROUND AND DEFINITIONS

2.1. Background

The concept of reference plant has been used for long in many countries with the main following objectives:

— To get a guidance in the implementation of a nuclear project based on another coherent nuclear project more advanced in time and that brings experience feedback,
— To get a reference in terms of safety concepts that have been already assessed by a nuclear regulatory body and then maximize the confidence in the level of nuclear safety of a new nuclear project,
— To minimize the risks of a nuclear project schedule by using established plant designs with a background of manufacturing and erection experience and to make savings in capital cost by adhering closely to already established design, manufacturing and erection techniques.

Besides, it must be noticed that a complete adherence to a reference plant is neither always possible nor desirable. Indeed, site specific conditions and respect of national regulations will lead to deviations from the reference plant. Moreover, in some cases the chosen reference plant is still a living project and it is necessary to adopt as a reference a coherent status of the project (e.g. design and plant configuration of the reference plant as submitted for the construction license application) that will be different from the reference plant "as built". For this reason a reference plant can only be a starting point for a nuclear project with a number of deviations that will have to be identified, justified and documented.

In the case where a vendor/designer proposes a new reactor design even if the design is innovative and evolutionary it is often based on previous reactor designs that have been, most of the time, modified to improve safety and economics. In that case, the "reference design" of the new nuclear project is derived from one or several nuclear power plant models that can be called "Parent design".

The following table gives examples of the use of the reference plant concept in France for the national and exporting markets:

TABLE 1.

NPP identification	Operator	NPP type	Reference NPP
Fessenheim 1,2 (CP0 reactor fleet type)	EDF (France)	PWR, 3 loops 2660 MWth	Beaver Valley (Westinghouse design)
Bugey 2,3,4, 5 (CP0 reactor fleet type)	EDF (France)	PWR, 3 loops 2785 MWth	Fessenheim 1, 2 & North Anna 1 (Westinghouse design) for the power uprate compared to Fessenheim NPP
Tricastin 1,2,3,4 (CPY fleet type)	EDF (France)	PWR, 3 loops 2785 MWth,	Bugey 4, 5
Paluel 1, 2, 3, 4 (P4 reactor fleet type)	EDF (France)	PWR, 4 loops 3817 MWth	South Texas 1 (4 loops PWR, Westinghouse design)
ChoozB 1, 2 & Civaux 1,2 (N4 reactor fleet type)	EDF (France)	PWR, 4 loops 4250 MWth,	Previous French 4 loops PWR plants
Flamanville 3 (EPR type)	EDF (France)	PWR, 4 loops 4500 MWth	N4 & Konvoi (Siemens/KWU design)
Koeberg 1, 2	ESKOM (Republic of South Africa)	PWR, 3 loops 2785 MWth	Tricastin 1
Ulchin 1, 2	KEPCO (South Korea)	PWR, 3 loops 2785 MWth	Le Blayais 3, 4 –EDF operator PWR, 3 loops 2785 MWth
Daya Bay 1, 2	CGN (China)	PWR, 3 loops 2905 MWth	Gravelines 5, 6 PWR, 3 loops 2785 MWth
Olkiluoto 3	TVO (Finland)	PWR, 4 loops 4300 MWth	N4 & Konvoi (Siemens/KWU design)
Taishan 1, 2	CGN (China)	PWR, 4 loops 4590 MWth	Flamanville 3

2.2. The concept of reference plant in IAEA safety publications

In 2008, INSAG 22 [1] introduced the reference plant concept to encourage new entrant countries to base their first construction on this concept: "*Many mature nuclear countries used a so-called "reference plant" concept for their first nuclear units. Under this approach, an imported plant has the same design and safety features as a plant already licensed by the regulatory body of the exporting country. However, care should be taken to ensure that the selected site and the reference plant site have similar characteristics or that any significant differences have been taken into account…. Also any construction by a new entrant will likely be based on the well proven technologies of an exporting country. It might be expected that the design has been licensed by the regulatory body in the exporting country, perhaps with the benefit of analysis by other regulatory bodies… It is highly recommended that the regulatory body in the importing country establish and maintain a knowledge transfer relationship with the regulatory body in the exporting country…*". Then, in 2012 INSAG 26 [2] indicated that, whenever possible, when a new entrant country chooses as its first nuclear power plant a power plant that would have essentially the same design as a nuclear power plant already licensed by an experienced regulator, the use of the reference plant concept would be desirable : "…*During the design safety review process for issuance of the construction licence for the first nuclear power plant, use of the design safety review conducted earlier by an experienced regulator for the reference plant could be appropriately made. However, it is essential that the* *regulatory body has a good understanding of the design and due attention is paid to the design differences on account of factors such as site related parameters, plant layout and incorporation of new design features based on operating experience and advancement in technology. This strategy is proposed primarily to ensure a high level of safety which incidentally, may also help expediting the licensing process*".

Further development about the concept of reference plant as discussed below in this paper may in time allow introducing it in the IAEA safety standards. As a matter of fact, this concept is neither introduced in the Safety standard SSG-12 [3] 'Licensing process for nuclear installation', nor in the Draft Standard DS 473 [4] 'Functions and Processes of the Regulatory Body for Safety.

2.3. Tentative definitions

2.3.1. The reference plant concept used for contractual purposes

The reference plant concept used for contractual purposes can be defined as a plant of same or similar design as that of the nuclear project to be built, having readily a level of detailed design sufficient to secure the project and limit the provisions for industrial risk. Experience of the vendor and of the buyer as well as the maturity of the proposed technology are important factors to limit industrial risks.

2.3.2. The reference plant concept used for regulatory purposes

The reference plant concept used for regulatory purposes can be defined as a plant of same or similar design and safety features as that of the nuclear project to be built, having already been licensed or certified in the vendor country or possibly in another country. In that case the concept of reference plant is used to help securing the licensing process and minimizing the risk of dead ends during safety evaluation process. Indeed, one can suppose that a reactor design that has gone through a thorough safety evaluation by an experienced regulatory body will comfort the feasibility in a reasonable time frame of the licensing of the same or of a similar design in another country. Especially in new entrant countries, the existence of a reference plant would facilitate the licencing process as the regulatory body (which has never licensed a nuclear power plant before) could learn considerably from the existing safety evaluation reports (SER) written as part of the licensing process of the reference plant and could obtain important insights from the results of various safety analyses that were carried out for the reference plant. The concept of reference plant could also be extended to a reference design at the level of a detailed design having strong technical similarities with an existing model using proven technologies and safety concepts already assessed by an experienced regulatory body.

3. IMPACT OF SPECIFIC CONDITIONS IN THE REGULATORY FRAMEWORK OF THE REFERENCE PLANT CONCEPT

Up to now, specific conditions related to the use of a reference plant concept in national regulatory frameworks is very seldom. Certain countries are currently revising their requirements to allow for more flexibility and to facilitate the completion and the submission of the safety case to be provided along with the chosen reference plant for regulatory purposes.

Even if such requirements, mainly for new entrant countries, are intended to bring more confidence about the new projects to be built in terms of support brought by an experienced regulatory body it could also lead to undue or unexpected difficulties. Indeed, imposing for example that the reference plant for regulatory purposes be already a plant in operation or under construction could considerably limit the number of potential adapted nuclear power plant models and associated vendors. Moreover, the requirement linked to the transmission of the entire licensing case could pose intellectual property issues. Finally, it must be also kept in mind that a new entrant regulatory body will also have to go through a rigorous licensing process that will ensure:

- Full endorsement of the safety case by the operator and the regulatory body,
- Full understanding of the safety case for the complete life of the plant,
- Full consideration of the national regulations.

4. CONCLUSION

The use of the reference plant concept for licensing purposes has proven its efficiency in terms of transferring knowledge between regulatory bodies and ease of the licensing process through a close collaboration between the regulatory body of the new entrant country and the regulatory body of the country of origin. Nevertheless, the experience shows that introducing requirements, if any, in the national regulatory framework about a reference plant for regulatory purposes should be done in a manner that leaves sufficient flexibility not to unduly limit the potential choice of reactor models acceptable in the recipient country.

As a matter of fact, the concept of reference plant could also be extended to a reference design at the level of a detailed design having strong technical similarities with an existing model using proven technologies and safety concepts already assessed by an experienced regulatory body.

It should be noted that the licensing process is not limited to the granting of the construction license and the concept of reference plant is also intended to provide insights to the regulatory body in the recipient country regarding the oversight of the construction and commissioning phases of the nuclear power plant to be built. This aspect was intentionally not developed in this article and further discussion should be planned to better explain it.

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MANAGEMENT OF ENGINEERING AND SAFETY KNOWLEDGE ALONG REACTOR LIFETIME

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Abstract

Often, current engineering approaches for nuclear power plants (NPPs) are still very similar to those put in place decades ago. Such approaches might have been adequate in the past, where regulatory and operational requirements were less stringent than they are today. They may still be adequate for evolutionary designs, where a proven-in-use design is improved but not radically modified. They are not optimal for innovative designs such as highly manoeuvrable reactors, hybrid reactors (combining production of electric power with other uses of energy to cope with demand variations), SMRs (Small Modular Reactors) or Gen IV reactors. In particular, engineering and safety knowledge could be better structured and systematic, and specific information easier to retrieve and understand. The approach proposed in this paper combines and takes advantage of research work currently being done at EDF R&D on several subjects: the engineering of complex systems (typically, cyber-physical or even socio-cyber-physical systems, and systems of systems), the design of innovative reactors, and safety justification.

1. INTRODUCTION

Nuclear power plants (NPPs) are subject to extremely stringent and ever increasing constraints covering a wide range of issues: safety (particularly after majors accidents have shaken public confidence), security and computer security, cost (for design, construction, operation, renovation and deconstruction, particularly in the light of cheap or subsidized alternative sources of energy) and operation (with, increasingly, the need to take into account a high percentage of intermittent renewables in the energy mix).

In many cases, current engineering approaches for NPPs are still very similar to those put in place decades ago. Such approaches might have been adequate in the past, where constraints were less stringent than they are today. They may still be adequate for evolutionary designs, where a proven-in-use design is improved but not radically modified. They tend to be less than optimal for innovative designs such as highly manoeuvrable reactors, hybrid reactors (combining production of electric power with other uses of energy to cope with demand variations), SMRs (Small Modular Reactors) or Gen IV reactors. In particular, engineering and safety knowledge could be better structured and systematic, and specific information easier to retrieve and understand.

2. GENERAL APPROACH

This paper proposes an approach that combines and takes advantage of three complementary elements:

- Safety & security requirements from national regulations and international standards are essential inputs. Their interpretation and reconciliation (when coming from different sources) are a challenge. The proposed approach first relies on a database organising and clarifying all applicable requirements.
- An NPP is a complex system that requires the cooperation of many engineering teams and disciplines covering a very wide range of topics and the complete lifetime of the plant, from initial prospective studies (to determine the main plant characteristics) to deconstruction. Thus, the approach also relies on high level, formal assumptions, requirements and behavioural modelling, facilitating teams coordination and enabling massive simulation for design verification and operator assistance.
- Lastly, the approach integrates a Claim-Argument-Evidence justification framework: whereas the formal modelling of the previous bullet represents objective engineering goals and facts, this framework organises and structures human reasoning and subjective judgement.

3. REGULATORY & STANDARD REQUIREMENTS

The design, construction and operation of an NPP are in a very large part guided and constrained by safety and security considerations. Many international standards and guides regarding these issues have been developed by bodies such as the IAEA (International Atomic Energy Agency), the IEC (International Electrotechnical Commission) and the IEEE (Institute of Electrical and Electronics Engineers), some specific to NPPs, others (like IEC 61508) not. Even though these standards bodies make efforts to coordinate their works, the current set of documents does not, by far, constitute an integrated whole: each body has its own way of organising a given topic into documents, and documents from a given body are often difficult to relate and compare with those of other bodies. Sometimes, even the approaches and principles are different, making comparison even more difficult.

In addition to international standards, most countries with NPPs have developed nuclear specific national regulations and guidelines. Like for international standards, national documents are organised and follow principles that differ from country to country. Also, the same international standard can be interpreted differently in different countries. Here again, there are ongoing harmonisation efforts, with bodies such as MDEP (Multinational Design Evaluation Programme) or WENRA (Western European Nuclear Regulator Association). However, the nuclear industry is still far from the level of agreement and consensus reached by other industries such as the aviation industry, and a plant design accepted in one country may very well be rejected by another. An NPP designed for construction and operation in a single country already faces a significant challenge understanding and meeting the requirements and practices of that country. NPPs designed for international markets face an even greater challenge.

The approach proposed can be decomposed into the following steps:

- Determination of the country (or countries) of interest for a given project, and of the standards and the international consensus bodies to be taken into consideration.
- Collection the corresponding regulatory, standard and consensus documents. Sometimes, national practices also need to be collected: what has been required, or on the contrary rejected, in recent projects.
- Extraction, from these documents, of the significant statements, in particular of individual definitions, requirements, recommendations and practices.
- Organisation of these statements into a structured database for easy retrieval of all the statements concerning a specific topic (e.g., defence in depth or data communication), country, engineering discipline (e.g., I&C or human factors), safety class, etc. The database can also facilitate the retrieval of statements a given level of "urgency" (e.g., requirements vs. recommendations, or safety vs. safety related). One can then determine which of these statements will be considered as design requirements. This will in general include all regulatory requirements, but can also include many if not all standard requirements, and a number of recommendations and practices (which are often de facto requirements).
- Development of unambiguous rules for engineering and operation, covering the selected design requirements. As previously noted, many are expressed in general terms and allow different interpretations. Also, as they come from different sources and may address similar goals using different terms, concepts or solutions, the rules provide an interpretation that is easier to manage by the subsequent engineering activities.
- Justification that compliance to these rules implies compliance with the original design requirements collected in the database. This justification can be organised as proposed in Section 5 Justification Framework.

When the plant progresses in its lifecycle, one will need to justify that its design, construction and operation are indeed in compliance with the rules. Sometimes, exceptions with respect to the rules might be necessary. Then, one will need to provide a direct justification that these exceptions are indeed consistent with the original requirements (see Fig. 1).

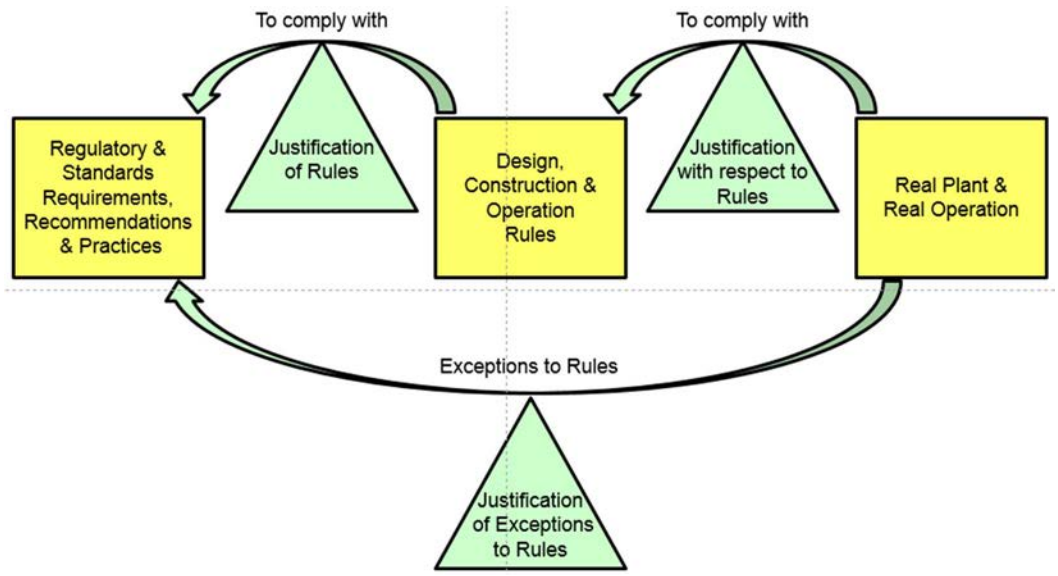

FIG. 1. Justification of the NPP with respect to regulatory and standard requirements.

4. ADVANCED MODELLING FOR THE ENGINEERING OF COMPLEX SYSTEMS

A complex system can be defined as a system that cannot be fully understood and mastered in all necessary aspects and to the necessary degree by a single individual or from the standpoint of a single discipline. On the contrary, it needs the cooperation and coordination of many disciplines and teams covering a very wide range of topics, from so-called "hard sciences" (such as physics, materials, civil engineering and computer engineering) to "softer sciences" (such as psychology and sociology). Coordinating these disciplines and their many teams is a difficult but essential issue, but as different disciplines often rely on specific concepts, terms and tools, they tend to have difficulties understanding one another. Even teams that practice the same discipline may have difficulties coordinating with one another: too much coordination could lead to paralysis; too little rapidly leads to chaos. Thus, the essence of systems engineering for complex systems is the art of coordinating many teams and very different disciplines just as necessary (see Fig. 2).

FIG. 2. A neutral coordination ground for the teams & disciplines working on the system.

Coordination is not only necessary at specific instants in the lifetime of an NPP: it is also necessary to ensure it over the lifetime, so that the engineers and operators of the future will know and understand the principles that guided the design of the plants, and the details necessary for correct and safe operation, retrofit, renovation and upgrade.

Together with a number of other industrial organisations, tool providers and academic and scientific bodies, EDF R&D is developing an innovative approach for the engineering of complex systems. This approach is based on the modelling and simulation of the dynamic phenomena affecting such systems, covering their complete lifecycle from prospective studies aiming at defining the nature and scope of a system to be developed, down to system operation and maintenance, retrofits and modification (see Fig. 3).

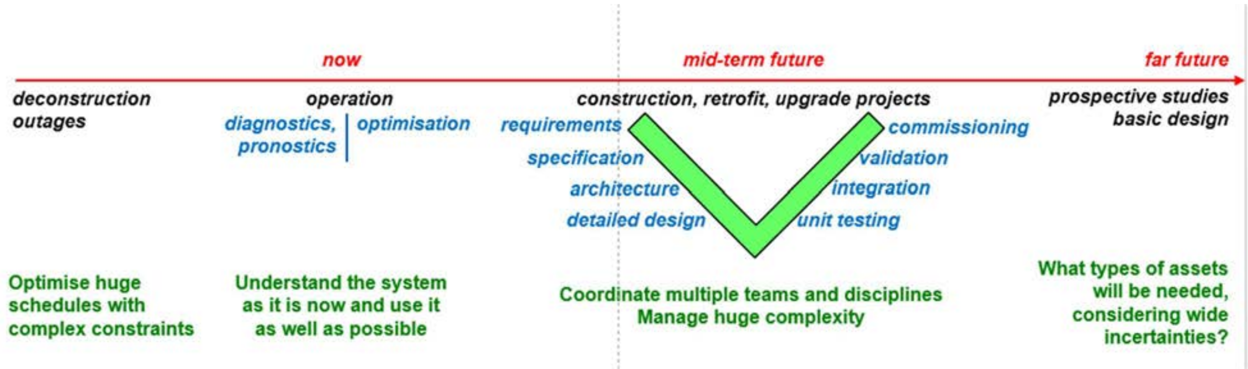

FIG. 3. Coverage of the system lifecycle.

The methodology is based in a large part on the FOrmal Requirements Modelling Language (FORM-L) that has been specified in the framework of the ITEA2 project MODRIO. Its scope is the formal modelling of properties, in particular of requirements and assumptions, in the form of envelopes of dynamic, time-dependent phenomena (see Fig. 4). Formal here means that the properties have rigorous syntax and semantics that can be interpreted by software tools to actively support a variety of systems engineering activities.

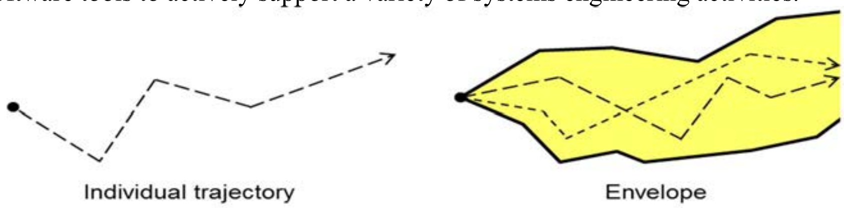

FIG. 4. Individual trajectories, versus envelopes allowing multiple trajectories.

The approach aims at the following objectives:

- Multi-aspect modelling, covering the needs of various disciplines such as physics, instrumentation and control, human behaviour, costs and revenues analyses, operations and maintenance, probabilistic analyses, complex tasks scheduling (e.g., for outages), etc. Each discipline can still use its methods, tools and models of choice, but it expresses its assumptions and its needs regarding the other disciplines in FORM-L, generally in the form of formal contracts.
- Multi-phase modelling all along system lifecycle, to allow step-by-step verification and validation and to reconcile safety and dependability with innovation. In particular, even for a given discipline, the system may be represented by different models, each representing different stages of development or operation. The first models represent preliminary and high-level views, and can be expressed as FORM-L envelopes. The subsequent ones represent more detailed views of the solutions chosen, and can be expressed either in FORM-L or in discipline-specific formats. Bindings allow information exchange between FORM-L models and non-FORM-L models.
- Multi-aspect and multi-phase modelling provide powerful means for the management of size and complexity. It can provide a complete view of the system, but also focused views on specific issues or subsystems. In particular, the formal modelling of contracts can help understand the complex webs of interdependencies between disciplines and between subsystems.
- Massive simulation in order to efficiently explore many different normal and failure situations. Tools are currently being developed at EDF to automatically generate test cases that are consistent with the assumptions modelled in FORM-L, to automatically verify the satisfaction of requirements also modelled in FORM-L, and to enable co-simulation of FORM-L models with

non-FORM-L models. Simulation can also be a very effective help in the understanding of complex models and systems, and can be used for training or for what-if analyses.
— Formal analysis, in favourable cases, to verify that in all modelled situations and under the specified assumptions, the specified requirements will be satisfied.
— The ability to efficiently assess a solution with respect to specified assumptions and requirements means that at each step in the development of solutions, one can explore multiple options (including innovative ones), eliminate those that do not meet the requirements, and then select the best remaining one(s) for the next steps.

5. JUSTIFICATION FRAMEWORKS

An assurance case can be defined as a documented body of evidence that provides a convincing and valid argument that a specified set of critical claims regarding a system's properties are adequately justified for a given application in a given environment. Initially, they have been applied mainly to safety cases, but they are now applied to security cases and to cases for business critical systems. ISO/IEC/IEEE 15026 is about systems and software assurance, and its Part 2 is about assurance cases, and in particular of structured assurance cases. Such cases are structured sets of claims, arguments and evidence, where:
— A claim is a proposition made about a system of concern.
— An argument is a reasoning of WHY a claim is true. It is usually supported by one or more pieces of evidence.
— A piece can be a fact, a datum, an object, a claim that is not further justified (and that is then an assumption), or (recursively) an assurance case supporting a sub-claim.

Justification frameworks have been introduced to help develop, verify and maintain complex assurance cases. They also facilitate the reuse of assurance case structures to similar systems or issues. The proposal here is to apply the justification framework developed by the Euratom FP7 project HARMONICS to justify that a solution under study not only complies with given regulatory or standard requirements as discussed in Section 3, but also that it meets the other types of requirements discussed in Section 4 (e.g., functional requirements, performance requirements or economic requirements).

Among many other things, the HARMONICS justification framework introduces 5 basic justification blocks:
— Concretion. This block is used when some aspect of the claim needs to be given a more precise definition or interpretation. It is needed when a claim refers to a requirement expressed in abstract terms, as is often the case of the regulatory or standard requirements of Section 3, is ambiguously worded or lacks precision.
— Substitution. Another common type of claim expansion involves transforming a claim about an object into a claim about an equivalent object, which can be viewed as a form of substitution. For example, one might claim that the evaluated system specimen has a certain property, and therefore the production system has this property too, provided that the production system is equivalent in some clearly defined way to the evaluated one. When engineering makes use of the advanced modelling introduced in Section 4, and when one wishes to an assurance case early in the system lifecycle, preliminary models (generally in FORM-L) can be used as substitutes to real systems, provided that the real systems are consistent with these models.
— Decomposition. This block is concerned with structure. Many arguments are based on the partitioning of some aspect of the claim, for example, according to the functions or the architecture of the system, to the properties being considered, or with respect to some sequence such as lifecycle phases or modes of operation. The basic idea is very simple: for example, in order to make a claim about a property of a system, we need to investigate whether the system has this property by evaluating its subsystems. To do this, we need to be clear about what the property is and we also need rules about how we view the system as being composed of subsystems and how the properties of these subsystems can be combined. Double decomposition is the most general form of decomposition in which a claim that a property of a system can be deduced from other properties of the constituting subsystems. Single decomposition is when either the property or the

system is being decomposed, not both. Preliminary evidence for decomposition can often be based on the advanced modelling introduced in Section 4.

— Calculation. This block is used to claim that the value of a property of a system can be computed from the values of related properties of other objects (e.g. its sub-systems). One application of the block is to provide a quantitative argument when the value of one property can be calculated from the values of other specific properties. For example, the availability of a system can be calculated from its reliability and its failure recovery time, or the average time of data retrieval from a database can be calculated from the probability that the data is in the cache and the time of data retrieval if it is not in the cache.
— Evidence incorporation. This block is used at the edge of the case tree to incorporate the evidence elements. The block demonstrates that a sub-claim is directly satisfied by its supporting evidence.

6. CONCLUSION

Even though these three techniques have been developed separately and are individually beneficial, they complement one another in a very effective manner, and when integrated, they constitute a powerful and effective means for representing safety and engineering knowledge on such a complex system as an NPP. Indeed, many of the limits and weaknesses of one can be compensated by the others:

— Regulatory and standard requirements are often expressed in general and abstract terms. Concretion with the justification framework may be used to trace the precise interpretation made by a project. That interpretation is explicit and can be subjected to reviews and scrutiny.
— Early justification of engineering rules with respect to regulatory and standard requirements may be associated with modelling in preliminary development phases. This could enable early safety and / or security assessments, when there is still time design changes and when that is not too costly.
— More generally, some of the evidence needed for safety or security justification may be taken from the advanced modelling of assumptions and solutions. Conversely, the fact that particular assumptions or solution features are cited in safety or security justification means that changes will need to be rigorously controlled.
— The advanced, progressive modelling approach proposed, together with simulation, can help understand the WHY of some of the design decisions (in order to satisfy requirements stated at earlier stages, in the framework of explicitly stated assumptions). In complex cases, the justification approach, with its decomposition and calculation blocks, provides additional help with much more explicit and informative traceability means than the traditional links, which are devoid of any semantics. This is particularly evident for requirements / claims that need to be allocated to a large number of components (e.g., response times or reliability).

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OVERVIEW AND COMPARISON OF INTERNATIONAL PRACTICES CONCERNING THE REQUIREMENTS ON SINGLE FAILURE CRITERION WITH EMPHASIZE ON NEW WATER-COOLED REACTOR DESIGNS

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Abstract

The Single Failure Criterion (SFC) ensures reliable performance of safety systems in nuclear power plants in response to design basis initiating events. The SFC, basically, requires that the system must be capable of performing its task in the presence of any single failure. The paper provides an overview of the regulatory design requirements for new reactors addressing the SFC at international level. It considers the SCF in relation to in-service testing, maintenance, repair, inspection and monitoring of systems, structures and components important to safety, as well as to severe challenges not included in the design envelope. A comparison is provided of the current SFC requirements and guidelines published by the IAEA, WENRA, EUR and nuclear regulators in several selected countries. Also, paper addresses some specific issues concerning the application of SFC such as the allowable equipment outage times and exemptions to SFC requirements. Based on the comparative evaluation, the conclusions are provided concerning the need for the harmonization at the international level.

1. INTRODUCTION

The Single Failure Criterion (SFC) ensures reliable performance of safety systems in nuclear power plants in response to design basis initiating events. The SFC, basically, requires that the system must be capable of performing its task in the presence of any single failure.

When applied to plant's response to a postulated design-basis initiating event, the SFC usually represents a requirement that particular safety system performs its safety functions as designed under the conditions which can include:

- All failures caused by a single failure;
- All identifiable but non-detectable failures, including those in the non-tested components;
- All failures and spurious system actions that cause (or are caused by) the postulated event.

In practice, the SFC exists in two major contexts: (1) system design requirements, largely associated with the general design criteria which require designing safety-related systems to perform safety functions to mitigate design-basis initiating events, assuming a single failure (and calling for the principles of redundancy, independency, separation, and those associated), and, (2) guidance on design-basis-accident (DBA) analysis, directed towards demonstrating adequate design safety margins based upon defined acceptance criteria.

2. OVERVIEW OF INTERNATIONAL REQUIREMENTS CONCERNING SFC

The IAEA, in the major document related to the design of the nuclear power plants, SSR-2/1 [1], provides under section 5 (General Plant Design) the requirement concerning the single failure criterion (Requirement 25):

"The single failure criterion shall be applied to each safety group incorporated in the plant design.

5.39. Spurious action shall be considered to be one mode of failure when applying the single failure criterion17 to a safety group or safety system.

5.40. The design shall take due account of the failure of a passive component, unless it has been justified in the single failure analysis with a high level of confidence that a failure of that component is very unlikely and that its function would remain unaffected by the postulated initiating event."

It is, furthermore, explained:

"A single failure is a failure that results in the loss of capability of a system or component to perform its intended safety function(s) and any consequential failure(s) that result from it. The single failure criterion is a criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure."

Generally, based on the SSR-2/1, IAEA requires the application of the single failure criterion (SFC) for all safety systems, which is further covered by the IAEA NS-G guidelines, e.g. NS-G-1.9, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [2] or NS-G-1.10 Design of Reactor Containment System for Nuclear Power Plants [3]. For example, NS-G-1.9 states:

"3.19. In order to achieve a well balanced design, appropriate consideration should be given to the redundancy and diversity of systems and components. For safety systems, this consideration should be based on a deterministic approach such as the application of the single failure criterion, supplemented by a risk informed approach."

Generally, in applicable IAEA NS-G guides it is discussed that the all evaluations performed for design basis accidents should be made using an adequately conservative approach. In a conservative approach, the combination of assumptions, computer codes and methods chosen for evaluating the consequences of a postulated initiating event should provide reasonable confidence that there is sufficient margin bounding all possible conditions. The assumption of a single failure in a safety system should be part of the conservative approach, as indicated in SSR-2/1 (5.26, DBA definition). On the other hand, care should be taken when introducing adequate conservatism, since:

- For the same event, an approach considered conservative for designing one specific system could be non-conservative for another;
- Making assumptions that are too conservative could lead to the imposition of constraints on components that could make them unreliable.

In the past, the IAEA had a document Safety Series 50-P-1, "Application of the Single Failure Criteria" [4]. This document is outdated and there is no new IAEA document specifically superseding it. In its section 2 it deals with the purpose of the SFC with respect to the safety of a nuclear power plant. It also shows where the criterion has its limitations. The third section explains the difference between active and passive types of failure and the consequences of the failure characteristics for the application of the criterion. Specifically referring to the Design Code [5], it states (in its section 7 dealing with exemptions to SFC) that non-compliance with the SFC *"may be justified for:*

- very rare PIEs,
- very improbable consequences of PIEs, or
- withdrawal from service for limited periods of certain components for purposes of maintenance, repair or periodic testing."

It is said that "This statement is in line with the relation between the frequency limit for a plant damage state, the frequency of an initiating event and the reliability of all the systems that are provided for dealing with this initiating event (safety group)".

The "limited periods" from the last bullet are often referred to as allowable outage times (AOTs). This bullet, therefore, relates the exemptions from SCF to the application of AOTs.

However, in the new major document related to the safety aspects of design of the nuclear power plants, SSR-2/1 [1] (which has replaced the Design Core [5]), justifiable exemptions to the SFC are not explicitly addressed anymore.

Screening concerning the application of the SFC was performed, additionally to the IAEA's requirements, of the Western European Nuclear Regulators' Association (WENRA) safety reference levels [6], European Utility Requirements (EUR) [7], and of regulatory requirements in selected countries. The comparative discussion of the requirements and positions is given in [8] and [9]. Matrix in Figure 1 attempts to present a concise comparative summary of positions and requirements.

Requirements or Regulatory Position	SFC applied to:	What systems have to meet SFC?	Is SFC applied during planned maintenance or repair with AOT?	Is SFC applied to passive components?	Is SFC applied in addition to assuming failure of a non-tested component?
IAEA	Safety system	General approach: systems which prevent radioactive releases in environment. Because of different designs, system names and description it can be related to: • Reactor Protection System • Engineering Safety Feature Actuation System • Core Decay Heat Removal System • Emergency Core Cooling System • Containment decay heat removal system • Containment Isolation System • MCR Habitability System • Emergency AC/DC power • Safety System Support System (Component Cooling Water, etc.)	Not explicitly discussed in regulations. The allowable periods of safety systems inoperability and the cumulative effects of these periods should be assessed in order to ensure that any increase in risk is kept to acceptable levels.	General approach is that the fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming Passive Equipment functions properly) nor (2) a single failure of a Passive Equipment (assuming Active Equipment functions properly) results in a loss of capability of the system to perform its Safety Functions. *Exemption for passive components exists if justification of high standard and quality design and maintenance is possible.*	Not explicitly discussed in regulations. See fourth column on the left side. In other words it means that if assessment of potential failure of any single component designed for the function in stand-by (non-tested) system shows the increase in risks above acceptable levels such configuration should not be allowed.
WENRA	Safety system				
EUR	Assembly of equipment (combination of systems and components that perform a specific function)				
U.S. NRC	Safety system				
Finish (STUK)	Safety system		Not explicitly discussed in regulations. The PSA shall be used to determine the surveillance test intervals and allowed outage times of systems and components important to safety. Actually, it is similar with above. YVL B.1 discusses the two failure criteria, (N+1) and (N+2). Some systems need to satisfy criteria (N+1) and some (N+2)		YVL B.1 discusses the two failure criteria mentioned in fourth column on the left side for Finish (STUK).
UK	Safety system		See IAEA, WENRA, EUR, U.S. NRC above.		See IAEA, WENRA, EUR, US NRC above.
Japan	Structures, systems and components (SSCs)				
Korean	Safety system				
Russian	Safety features (safety systems elements)				
China	Safety system				
Canadian	Safety group/Safety system		A request for an exception during testing and maintenance should be supported by a satisfactory reliability argument covering the allowable outage time		Similar position to IAEA, WENRA, EUR, US NRC above even that section 7.6.2 of REG-DOC-2.5.2, [14], refers to the old IAEA, Safety Series No. 50-P-1, [4], which was withdrawn without applicable replacement.

FIG. 1. Overview of SFC Application Requirements.

3. SFC APPLICATION IN THE CONTEXT NEW WATER-COOLED REACTOR DESIGNS

Obviously, SFC as a design principle does not have a definitive alternative. However, the experience with application of SFC which was gained over the period of time stretching over half of a century points out to some weaknesses and/or insufficiency in rigid SFC rule which may need to be reconsidered in the context of new water-cooled (or other) reactor designs. A number of particular considerations may be summarized via the following two main points regarding traditional SFC application, which can be very frequently encountered in the discussions:

— The first point is that traditional application of the SFC has, apparently, sometimes led to redundant system components, which contribute to adequate and acceptable safety margins, but may have only minimal impact on risk, based on conventional risk assessment studies. While maintaining adequate safety margins is a major safety objective, the application of the worst single-failure assumption for all DBAs may, in some cases, result in unnecessary constraints on licensees.

— The second point is that the traditional implementation of the SFC does not consider potentially risk-significant sequences involving multiple (rather than single) failures as part of the DBA analysis. Common-cause failures, some support system failures, multiple independent failures, and multiple failures caused by spatial dependencies and multiple human errors, are phenomena that impact system reliability, which may not be mitigated by redundant system design alone. Some risk-informed alternatives might consider such failures in DBA analyses if they were more likely than postulated single-failure events.

Additionally to those (or, sometimes, coming out from them) there are some other considerations which may become increasingly relevant for the application of SCF to the new water-cooled reactor designs.

One consideration is that, apparently, the SFC has not always been uniformly applied to passive failures in fluid systems, and such passive failures may gain an additional level of importance with many of new water-cooled reactor designs due to increasing reliance on the passive safety features. Therefore, they may need to be particularly addressed in the "new" SFC requirements. This would include resolving the question of which passive failures to include in the treatment. For example, the passive failure of a single check valve, pipe, or tank could have significant implications on the DBA analysis.

Another consideration is that traditional application of the SFC might have not always led to the design of safety systems having the reliability commensurate with the frequency of important safety challenges. Generally, for more frequent challenges (with same or similar consequences), higher system reliability is desirable to enable safety systems to respond in a manner that results in safe plant shutdown. (As an example from practice, a reference can be made to the discussions in [10]. On the basis of generic safety issue studies, rulemaking, and risk considerations, the U.S. NRC had to supplement the SFC with additional regulations or licensing guidance applicable to selected safety systems. These led to plant modifications and licensee programs in the U.S. to either improve system reliability or demonstrate that the system design was otherwise adequate to cope with the postulated initiating events. Relevant examples include the so-called Station Blackout (SBO) Rule, the Anticipated Transient without Scram (ATWS) Rule, and the guidance to increase availability of PWR auxiliary feedwater systems.)

With above in mind, it may be needed to reconsider the traditional, rigid, implementation of the SFC when applying it to new water-cooled (and other) reactor designs. This may include state-of-the-art risk and reliability methods as a basis to adjust the criteria by using quantitative estimates of functional or system reliability as well as quantitative estimates of expected frequency of challenges. As an example, some options for reconsidering the traditional SFC in a way which may be particularly useful for the application to the new water-cooled reactor designs were discussed in the references such as [10] and [11]. Some options, based on considerations in [10], are briefly discussed below.

One option for reconsidering the application of SFC is in risk-informing the selection of single- and multiple-failure accident sequences for DBA analysis based upon their frequencies. This could eliminate some single-failure accident sequences from the design basis, while it also could lead to the addition of some multiple-failure sequences. It would require that the reliability of certain components or structures excluded from the DBA analysis would be additionally monitored to assure their continued high reliability or integrity (e.g., reactor coolant pressure boundary). This one can be referred to as sequence-based approach to SFC reconsideration.

Another option is in risk-informing the application of the SFC to systems in a manner that is commensurate with their safety significance. This one can be referred to as system-based approach. It would rely on the principles of risk-informed safety categorization of SSC which are already well developed (e.g. 10 CFR 50.69 [12]). For example, for non-safety significant systems which were traditionally "safety-classified" the requirements of the current SFC may be relaxed. Additional performance monitoring of safety significant systems would be required.

Yet another option would be to generalize the SFC by varying the redundancy requirement according to the initiating event category, and providing guidance on diversity. This can be supplemented by allocating system or function reliability on the basis of top-level risk guidelines and targets. This is, also, well established practice in some IAEA member states. The licensee would establish targets for lower-level (train-level) reliability satisfying the functional reliability targets upon which the SSC treatment, including performance monitoring, would be established.

The above outlined options include a range of concepts that might be used to pursue risk-informed and performance-based changes to the SFC application to new reactor designs. Other options could be constructed, involving different combinations of the basic concepts.

To put into perspective the need for adjusting the traditional SFC approach when applied to the new reactor designs (or, at least, advisability of it), the following points may be included into a consideration:

— New water cooled reactor designs (at least those into the review of which the authors of this paper were involved) have considerably lower quantitative risk measures (e.g. CDF) and very much different risk profile as compared to the operating plants. This may only amplify the issues discussed above.

— New water-cooled reactor designs have largely increased reliance on passive safety features, which calls for clarification of position toward the treatment of passive failures in fluid systems. Examples may include the following:
  - Some (actually, many of) new designs rely on single large passive components or structures, e.g. large single tanks or large single heat exchangers relying on natural circulation. How should those be treated in the SFC analysis, in the context of CDF for internal initiating events at the order of magnitude of 1E-07/y?
  - Some designs rely on check valves (non-return valves) having to open or close at rather low delta pressure (e.g. dictated by gravity force). How should those be treated in the context of SFC (e.g. active or passive failures)?
  - In the case of some designs Safety Analysis Report claims for physical impossibility of certain initiators, such as "large LOCAs" (opening the questions how "large LOCA" is to be defined; but, further discussion on this matter is omitted, for the sake of the example) due to the limit on the diameter of the pipes connecting to the reactor vessel. This leaves (if the argument is accepted) the residual risk of the reactor vessel failure. In the case of the operating plants the core damage risk contribution of the reactor vessel failure is at the order of 1% or lower. (Typically, it is claimed that CDF contribution of such initiator is below 1E-07/y on account of vessel in-service inspections and similar arguments.) However, with considerably reduced CDF (internal initiators) for the new designs this contribution easily becomes 10% or, even, considerably higher. How is the reactor vessel failure to be treated in the SFC evaluation in such a context?
  - Other examples may include failure modes such as filter or screen plugging or clogging. Those have been seen as considerable contributors to core damage risk in some Safety Analyses Reports for the new designs. How are those to be treated in the SFC analysis?

— With high reliance on passive safety features, the most of the risk coming from active failures is associated with common cause failures. Some of the new designs have been seen to be modified based on the preliminary PSA results due to high contributions of CCF of certain valves. How are CCFs to be treated (or: should they be addressed) in the SFC analysis of a new design with internal initiating evens CDF of 1E-07/y?

— Obviously, with high reliance on passive safety features, the structural reliability of the corresponding SSCs becomes correspondingly important. Examples may include seismic fragility (or extreme wind fragility or aircraft crash fragility, or others). How should those be related to the SFC, putting altogether in the context if internal events CDF of 1E-07?
  - Current SFC philosophy is focused on demonstrating the safety margins (e.g. peak cladding temperature or containment pressure) following the design basis initiator accompanied by the worst single failure. Considering that external events such as earthquakes may become dominating risk contributors for "inherently safe" new designs, should the approaches or concepts such as "seismic margins" or, more generally, structural load margins, be somehow related to the SFC?

— With design-specific internal events CDF considerably lowered, the role of the spatial impacts induced by internal hazards (similarly to external) becomes increasingly important. The same questions apply as above.

— Although most of the new designs with increased role of the passive features are free of the short-term need for human actions, certain human actions may be needed in the longer time frame. Should those be considered in the SFC in the new (low) risk context?

— How the SFC evaluation should be related to the concept of "practical elimination" of sequences with large radioactivity releases, promoted in the IAEA documents [1] and [13]?

These and other points of consideration may make it advisable to consider adjusting the traditional SFC approaches for applications to the new reactor designs, using some of the options discussed above.

4. CONCLUDING REMARKS

One important conclusion which can be made is that nuclear industry and regulation applications either to single failure criterion or defense-in-depth are not very well harmonized in the international practice. Additional effort is advisable to be made in order to establish more strict and harmonized design requirements with regard to either SFC or DiD to improve safety of nuclear installations in future.

Past experience with application of SFC which was gained over the period of time stretching over half of a century points out to some weaknesses in rigid application of traditional SFC rules. Those include failing to establish reliability of the functions important to safety which would be commensurate with frequencies of challenges to them.

It has been many times repeated that traditional design basis analyses (DBA) are conservative because they assume failure of single train. It has been rarely pointed out that the same analyses may be optimistic because they assume that complementary train will succeed for granted.

It seems to be advisable to reconsider and adjust the SFC approach for application to the new water-cooled (or other) reactor designs. The options for that have been discussed and identified in the references.

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ISBN 978–92–0–104618–5
ISSN 0074–1884