International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs

Tuesday, October 18, 2022 - Friday, October 21, 2022

Vienna

Book of Abstracts

Book of abstracts

Contents

3

Issues and needs in applications of deterministic safety analysis for demonstrating safety of nuclear power plants

Author: Jozef Misak[1]

[1] *Nuclear Research Institute Rez (Czech Republic)*

Corresponding Author: mis@ujv.cz

Significant progress has been made in the past in the scope, methods and quality of the deterministic safety analysis, reflecting the experience gained, capabilities of computers and computer codes. The basic rules for performing deterministic safety analysis of all plant states are established in the relevant IAEA safety guide SSG-2 (Rev.1). Nevertheless, there are needs for further enhancements of the current practices in applications of the deterministic safety analysis.
The paper discusses examples of the areas where such enhancements would be appropriate, based on numerous reviews of safety analysis reports for new nuclear projects. The areas for enhancement include harmonization of approaches for analyses of radiological consequences of reactor accidents and their integration with thermal-hydraulic analyses, demonstration of practical elimination of early or large radioactive releases, improvements in consistency between neutronic, thermal-hydraulic, structural and radiological analyses, broader use of quantification of uncertainties, establishment of clear links between analysis of internal and external hazards and safety analysis of plant states, the use of CFD codes in licensing, integration of deterministic and probabilistic safety analysis, and performing safety analysis for novel reactor designs, in particular for small modular reactors. The reasons and main directions for the enhancements will be presented in the paper.

Email address:

Jozef.Misak@ujv.cz

Confirm that the work is original and has not been published anywhere else:

YES

6

ESFR SMART: a SFR project enhancing safety by innovative design

Authors: joel guidez[1]; enrico girardi[2]; konstantin mikityuk[3]; bodi Janos[3]

[1] *CEA*

[2] *EDF*

[3] *PSI*

Corresponding Author: joel.guidez@cea.fr

ESFR SMART is a European project on sodium fast reactors with the aim of improving the safety of these reactors to reach the last safety requirements.
This final design of the reactor is now available. It is based on a search for simplification, passivity and practical elimination of accidents thanks the huge experience feedback from these reactors
This paper presents all the simplifications made in this final design: elimination of the safety vessel, elimination of the polar table, and replacement of the systems inside the primary vessel dedicated to decay heat removal by external systems connected to the heat exchangers, passage to straight secondary pipes, etc…
We also present the innovations of inherent safety features in the reactor core with a zero sodium void effect, an additional system of passive control rods, and evacuation devices to a core catcher located under the core.

Passive systems have been used also for the secondary loops able to operate in natural convection, and with passive thermal pumps. All the general design of the plant uses the natural convection possibilities of sodium.
A review of mitigation improvements is also provided

Email address:

joel.guidez@cea.fr

Confirm that the work is original and has not been published anywhere else:

yes

8

THERMAL HYDRAULIC CODES VALIDATION FOR SPENT FUEL POOL CONDITIONS

Author: Ruslan Sledkov[None]

Co-authors: Oleg Stepanov [1]; Mark Bedretdinov

[1] *JSC GIDROPRESS*

Corresponding Authors: ruslansledkov@yandex.ru, bedretdinov2013@yandex.ru, olegmpei@yandex.ru

The work is devoted to the procedure of validation and cross-verification of the newly developed computational code ROK/B. The main problem solved by the ROK/B code is the calculation proof of the coolant density in the spent fuel pool (SFP) (untight reactor) and the temperature regime of the fuel assemblies when a long-term shutdown of the cooling systems (cessation of the supply of cooling water). In addition, with the help of the computational code ROK/B, it is possible to carry out a calculation for an accident with the discharge of the coolant from the SFP with a simultaneous long-term shutdown of the cooling systems. The computational code ROK/B provides for the ability to carry out calculations for various types of designs of the fuel assemblies and VVER, in particular, VVER-1000, VVER-1200 and VVER-440 power units with single and two-tiered fuel assemblies arrangement, with covered pipes in racks (for compacted assemblies storage) and pipes without covering, with case assemblies and caseless ones.
During refueling, a high level of the coolant is maintained, which makes it possible to do "wet" transportation of the assemblies from the reactor to the SFP. The mathematical model for heat and mass transfer calculation, including the boiling coolant model, implemented in the ROK/B code, includes: the equation of motion, equations for calculating the enthalpy along the height of the fuel part of a fuel assembly during natural circulation of the coolant in the channel with the fuel assembly (lifting section) and the interchannel space (lowering section), ihe equation of the mass balance between the channels of the racks with assemblies and in the inter-assemblies space and the amount of evaporated (and outflowed) water, the heat balance equation for a fuel rod in a vapor medium The system of equations is supplemented by closing relations for calculating the thermophysical properties of water and steam, fuel and shell materials, as well as the coefficients of heat transfer from the wall to the steam, hydraulic resistance and density of the steam-water mixture in the channels, as well as the heat from the steam-zirconium reaction. Validation of the computational code was carried out on the basis of the data of the ALADIN experiment performed by German specialists and the data of OKB Gidropress JSC. Cross-verification of the ROK/B computational code was carried out in comparison with the calculation results using the KORSAR/GP, SOCRAT/B1 codes. Based on the results of the validation, it was concluded that the deviation of the ROK/B results from the experimental data is no more than 2 – 10% (10% for the option with a fuel rod power of 20 W). Based on the results of cross-verification, it was concluded that the deviation of the ROC/B results from the calculation results for the KORSAR/GP and SOCRAT codes is no more than 0.5% (SOCRAT/V1) and to 10% (KORSAR/GP).

Email address:

sledkov@grpress.podolsk.ru

Confirm that the work is original and has not been published anywhere else:

Yes

9

International cooperation: harmonization of safety standards and guidance for innovative reactor technologies and sharing of regulatory reviews

Author: Alexander Bolgarov[None]

Corresponding Author: aibolgarov@rosatom.ru

It seems that the international community has come to a consensus that the requirements for ensuring the safety of nuclear power plants should be the same in all countries (for historical reasons, the requirements were initially different). The report says how the Russian Federation is working to harmonize legislation, codes and regulations with the IAEA safety standards both domestically and at venue of the World Nuclear Association (WNA), raises the question of the rationality of joint consideration of safety aspects of new innovative projects by regulators of different countries with a view to the anticipated reduction in the licensing period for nuclear power plants in the future.

Email address:

aibolgarov@rosatom.ru

Confirm that the work is original and has not been published anywhere else:

YES

10

Application of general design requirements: safety classification, design/protection for internal and external hazards, etc.

Author: Alexander Renev[None]

Corresponding Author: alrenyov@mail.ru

The purpose of the presentation is to present modern approaches to safety of VVER reactors, taking into account the lessons learned from accidents that have occurred in the world, including the accident at the Fukushima Daiichi Nuclear Power Plant in Japan.
This accident forced all the designers and manufacturers of reactor facilities once again to carefully review the design requirements, to revise some approaches to design, taking into account the new challenges to the safety of nuclear power plants. Russian designers even before this accident made many design decisions, which the world started talking about just after the accident.
Nevertheless, after the accident, they also drew conclusions aimed at improving safety. The proposed presentation deals with new design solutions in the field of safety, the Russian approach to the classification of NPP elements in terms of safety and protection from external and internal influences.

Email address:

adrenev@rosatom.ru

Confirm that the work is original and has not been published anywhere else:

YES

12

Design basis and safety justification of nuclear fuel for RITM-200 reactor

Author: Laptev Sergey[None]

Corresponding Author: seglaptev@rosatom.ru

The purpose of the report is demonstration of nuclear fuel design basis and safety justification approaches which are using for RITM-200 project. Nuclear fuel is one of the main physical barriers of a nuclear power plant safety, preventing the spread of radioactive fission products into the environment. The fuel of the RITM-200 reactor is innovative and radically different from the fuel of VVER and PWR reactors. In this regard, the acceptance criteria for safety justification are different from acceptance criteria that are currently used both in Russia and in the world for safety justification of existing reactors. Design features aimed at safety providing of RITM-200 project will be presented in the report.

Email address:

SeGLaptev@rosatom.ru

Confirm that the work is original and has not been published anywhere else:

YES

13

TEST 3.0

Author: Francisco Parada[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

why not

Confirm that the work is original and has not been published anywhere else:

yes

Email address:

f.parada-iturria@iaea.org

14

TEST 4

Author: Francisco Parada[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

Poster with acceptance to oral

Confirm that the work is original and has not been published anywhere else:

no

Email address:

f.parada-iturria@iaea.org

15

TEST 5

Author: Francisco Parada[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

attempt from oral to poster

Confirm that the work is original and has not been published anywhere else:

YES

Email address:

f.parada-iturria@iaea.org

16

test 6

Author: Francisco Parada[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

Rejhect with the possibiliy of a poster

Confirm that the work is original and has not been published anywhere else:

yes

Email address:

f.parada-iturria@iaea.org

17

test 7

Author: Francisco Parada[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

accepted but as poster

Confirm that the work is original and has not been published anywhere else:

yeah baby

Email address:

f.parada-iturria@iaea.org

20

Phenomenological Approach as a Tool for Assessment the Self-protection of New Reactor Types in reactivity accidents

Authors: Maxim Uvakin[None]; Mikhail Bykov [None]

Corresponding Author: uvakin_ma@grpress.podolsk.ru

Methods for solving phenomenological problems with the introduction of the maximum possible positive reactivity for the selected core designs are considered. A method is proposed for assessing the safety and self-protection of the reactor plant based on the results of the solution.

Email address:

arzamas–16@yandex.ru

Confirm that the work is original and has not been published anywhere else:

yes

21

Development of Uncertainty and Sensitivity Analysis Methodology for Severe Accidents at NPP with VVER

Author: Anna Nikolaeva[1]

Co-authors: Valery Astakhov [1]; Alexander Lityshev [1]; Stanislav Pantyushin [1]

[1] *OKB Gidropress JSC*

Corresponding Author: anya.nickolaeva@yandex.ru

According to the IAEA's requirements to NPP design the analysis of beyond the design basis conditions can be based on the best estimate method (footnote 13 to requirement 20 of SSR-2/1) if national requirements do not specify more stringent approaches. The Russian regulatory document of the highest level sets implementation of the best estimate method to SA modeling at the level of requirements (i. 1.2.6 of NP-001-15). The quantitative result of applying the best estimate method is an

estimation of the average value of a parameter important to safety and its uncertainty. Considering that an explicit quantitative assessment of uncertainties in the analysis of SA can be almost impossible due to the complexity of the phenomena and the lack of experimental data, the deterministic analysis of SA in accordance with the IAEA recommendations (SSG-2 (Rev. 1), GSR Part 4 (Rev.1)) should be accompanied by a sensitivity analysis to demonstrate the robustness of the results and the absence of an edge effect.
Despite the fact that methodology for sensitivity and uncertainty (S&U) analysis have already been developing for a long time and relatively successfully (for example, the regulatory document (RB-166-20) has been developed that regulates such analyzes in Russia), there are still a number of unresolved issues. For instance, there are no definite answers to such questions as definition of:
– the goals and objectives of the analysis;
– the necessary and sufficient number of taken into account input uncertainties;
– the list of considered input uncertainties;
– parameters destribution functions for input uncertainties;
– figures of merit (FOMs) list;
– reliable sensitivity analysis method;
– reliable uncertainty analysis method;
– the necessary and sufficient number of calculations to ensure the statistical reliability of the results (when using statistical methods);
– S&U analysis results interpretation;
– etc.
Within the framework of this study, the methodology of S&U analysis developed and implemented for VVER in the OKB Gidropress JSC for the analysis of SA is shown. It is based on the use of the Monte Carlo method with aim to determine average value and its deviation for list of FOMs important for the in-vessel phase of SA and to check the possibility of an edge effect presence. Input uncertainties are set based on data from the design documentation. The convergence of the mean values of all FOMs and their normal deviations is considered as a criterion for the termination of the computational study. To determine the presence of a correlation between FOMs value and input uncertain parameters, rank criteria are used.

Email address:

anya.nickolaeva@yandex.ru

Confirm that the work is original and has not been published anywhere else:

YES

24

U.S.A. Regulatory Efforts for Cybersecurity of Small Modular Reactors/Advanced Reactors

Authors: Ismael Garcia[1]; Juris Jauntirans[2]; Michael Rowland[3]

[1] *United States Nuclear Regulatory Commission*

[2] *U.S. Nuclear Regulatory Commission*

[3] *Sandia National Laboratory*

Corresponding Author: ismael.garcia@nrc.gov

Small Modular Reactors / Advanced Reactors (SMR/ARs) are expected to provide safe, secure, and economical power that have the potential to support initiatives aimed at combating climate change. Current proposed SMR/ARs involve diverse technologies that include next generation modular pressurized water reactors, high temperature gas cooled reactors, molten salt reactors, and liquid metal cooled fast reactors. These diverse technologies each have a unique set of functions (and systems) that support both nuclear safety and security.

To address these challenges, the U.S. Nuclear Regulatory Commission (NRC) is moving toward a risk informed, performance based and technology-neutral regulation and associated regulatory

guides.

The U.S. NRC, supported by cybersecurity experts from DOE national laboratories and U.S. universities, has undertaken efforts to develop a regulatory guide (RG), to provide an advanced reactor licensee with an acceptable approach for meeting the requirements of the proposed cyber security rule for advanced reactors, 10 CFR 73.110, "Technology neutral requirements for protection of digital computer and communication systems and networks." The RG aims to provide a process that accounts for the differing risk levels within advanced reactor technologies while providing reasonable assurance of adequate protection of public health and safety and promoting the common defense and security and protecting the environment. As such, a key RG outcome will be to provide the licensee with a risk-informed approach that would allow for the design and implementation of a cyber security program to meet demands for protection against the unacceptable consequences from a cyber attack.

In order to accommodate the wide range of advanced reactor technologies to be assessed by the NRC, the following is a subset of the key assumptions and trends having relevance to cyber security of SMR/ARs:
1. Advanced Reactor designs are expected to have increased reliance on digital systems, emerging technologies, passive safety features and other novel design features.
2. Novel use cases such as remote monitoring and autonomous operations are planned, which demand reassessing legacy systems isolation paradigms that are common in the existing U.S. commercial power reactor fleet.
3. Ongoing harmonization among international standards and approaches may support more sophisticated security concepts including security design features, customized control catalogs and performance-based objectives.

Email address:

Ismael.Garcia@nrc.gov

Confirm that the work is original and has not been published anywhere else:

Yes

25

Impact of Cyber Security Features on Digital Instrumentation and Control Systems Important to Safety at Nuclear Power Plants – Evaluation Framework

Author: Ismael Garcia[1]

[1] *United States Nuclear Regulatory Commission*

Corresponding Author: ismael.garcia@nrc.gov

Cyber security seeks to prevent unauthorized accesses to information, software, and data in order to ensure that three attributes are met, namely:
• The prevention of disclosures or access of information that could be used to perform malicious or misguided acts which could lead to an accident, an unsafe situation or plant performance degradation (confidentiality);
• The prevention of unauthorized modifications that degrade a safety function (integrity); and,
• The prevention of unauthorized withholding of information, data or resources that could compromise performance of a safety function (availability).

A cyber security feature is a provision, control, or function specifically designed for cyber security purposes. Cyber security features and safety functions are implemented in digital Instrumentation and Control (I&C) systems at nuclear power plants to protect against cyber-attacks and protect the plant from postulated initiating events, respectively, that could compromise safety. Specifically,

cyber security features are implemented for the protection of digital I&C systems against unauthorized access. The safety functions and cyber security features should be designed and implemented to prevent them from compromising one another.

This paper documents an evaluation framework intended to apply to all digital I&C systems important to safety, both hardware and software, and was derived from the ongoing work being performed by the Nuclear Energy Agency (NEA) Committee on Nuclear Regulatory Activities (CNRA) Working Group on Digital Instrumentation and Control (WGDIC). In this context, hardware includes industrial digital devices of limited functionality, for example, while software includes firmware and logic in any form, including supporting data; this includes, but is not limited to application, operational and pre-existing software and software tools, intellectual property cores, field programmable gate arrays, complex programmable logic devices, network equipment, and items intended for non-safety purposes with the potential to interfere with safety systems.

The evaluation framework will address topics such as the following:
• Cyber security requirements and safety requirements;
• Vulnerabilities;
• Assessment of cyber security features;
• Qualification of cyber security features; and
• Maintenance and operational considerations.

Furthermore, the evaluation framework discussed by this paper is not to be construed as a requirement, regulation, or acceptable guidance by either domestic or international regulators. Instead, it is intended to serve as a potential foundation or technical basis to be used for developing clear and sufficient regulatory guidance for ensuring that safety features and cyber security features for digital I&C systems important to safety at nuclear power plants are designed and implemented to prevent them from compromising one another.

Email address:

Ismael.Garcia@nrc.gov

Confirm that the work is original and has not been published anywhere else:

NO

26

Development of the first Russian research molten salt reactor for technology trial of minor actinides burning. Nuclear safety features

Author: Evgeny Vlasenko[None]

Corresponding Author: vlasenko_zhene@mail.ru

The paper reveals designation, tasks and status of development of the research molten salt reactor for technology trial of minor actinides burning at the FSUE "MCC". Minor actinides are the most dangerous long-lived wastes of nuclear power plants. Taking into account potential decrease of period of danger of highly-radioactive wastes from 10 000 years to 300 years, minor actinides burning is an important step towards improvement of ecological ratings of nuclear electricity generation. The main technological parameters are shown. Main scientific and technological challenges are in focus. Nuclear safety features are discussed. Technologies and the results of material science and other types of experiments will be used as a basis for construction of the full-scope industrial minor actinides burning reactor. At the current project milestone necessary justifying R&D works are in progress. These works include studies of physical properties of fuel salt and intermediate circuit salt, development of operations of fuel salt composition modification and reprocessing, development of structural materials for the fuel salt recycling module as well as a reactor equipment, mathematical modelling of neutron-physical and thermal-hydraulic processes. Moreover, nuclear reactor construction documentation is being developed. According to preliminary estimates, the date of physical start-up of the research reactor is 2032.

Email address:

vlasenko_zhene@mail.ru

Confirm that the work is original and has not been published anywhere else:

YES

27

TRACE Confirmatory Analysis Model Development and Validation in NuScale Design

Authors: Peter Lien[None]; Weidong Wang[1]; Jason Thompson[1]; Ron Harrington[1]

[1] *US NRC*

Corresponding Author: peter.lien@nrc.gov

Compared to traditional large reactors, small modular reactors have advantages to attract utilities around the world in recent years. These advantages include shorter construction schedule and lower cost, lower risk of design basis events and more applicable passive systems. NuScale is the first license application among small modular reactors submitted to Nuclear Regulatory Commission for design certification. In the novel design, some unique thermal hydraulic phenomena are involved in the operational and accidental scenarios, e.g., high-pressure condensation, helical coil heat transfer and natural circulation in power operation. These phenomena are critical to the nuclear safety and thus require in-depth evaluation by the applicant. To support regulatory decision, NRC TRACE confirmatory analyses becomes indispensable. In this paper, the NRC TRACE confirmatory model development and validation for NuScale design are presented. Experimental separate effect tests and integral effect tests were used to benchmark the TRACE models. The goal was to prepare a plant model to support regulatory decision making. Model development strategy, assumptions, and high-level assessment results are reported to the maximum extent that complies with the requirement to withhold proprietary information.

Email address:

Peter.Lien@nrc.gov

Confirm that the work is original and has not been published anywhere else:

YES

28

The OECD/NEA Working Group on the Analysis and Management of Accidents (WGAMA): Advances in codes and analyses to support safety demonstration of nuclear technology innovations

Authors: Ahmed BENTAIB[1]; Hideo Nakamura [2]; Martina Adorni[3]; Luis Enrique Herranz [4]; Pierre Ruyer [1]; Didier Jacquemain[3]; Fulvio Mascari [5]

[1] *IRSN*

[2] *JAEA*

[3] *OECD*

[4] *CIEMAT*

[5] *ENEA*

Corresponding Authors: martina.adorni@oecd-nea.org, ahmed.bentaib@irsn.fr, nakamura.hideo@jaea.go.jp

The OECD Nuclear energy Agency (NEA) Working Group on the Analysis and Management of Accidents (WGAMA) is responsible for activities related to potential accidental situations in nuclear power plants focusing mainly on current light water reactors, and also have application for some advanced reactor designs. The objective is to assess and, where necessary, strengthen the technical basis needed for the prevention, mitigation and management of potential accidents in nuclear power plants, and to facilitate international convergence on safety issues and accident management analyses and strategies.
For this purpose, the WGAMA activities address topics related to reactor coolant local and system thermal-hydraulic behavior, in-vessel and ex-vessel behavior of degraded cores, containment behavior and protection, and fission product (FP) release, transport, deposition and retention, for both current and advanced reactor designs. As a result, the WGAMA's achievements have been outstanding in preparing technical reports, becoming reference materials, and in organizing workshops and conferences to discuss innovative methods, materials and technologies in the various fields covered by the group.
The paper aims to summarize the recent activities and the substantial outcomes produced in the recent years such as in thermal-hydraulics, computational fluid dynamics (CFD) and severe accidents (SAs), with focus on new reactors applications. Particular emphasis will be placed on the current progress in each of the fields: the consolidation and extension of the validation database in thermal-hydraulics; the extension of CFD use in their application to nuclear reactor safety; and, in SAs, the better understanding of the boundary conditions prevailing during key accident phases. As such, ongoing WGAMA activities are numerous and several of them are planned to be launched into/carried out in the near future, which are shortly mentioned too in this paper. Main outcomes and recommendations from such activities that should contribute for continuous safety improvements will conclude the paper.

Acknowledgements
The significant contributions of those individuals who had a key role in conduct and success of the activities of the Working Group on the Analysis and Management of Accidents, such as the bureau members, task leaders, contributors and members, are kindly acknowledged.

Email address:

ahmed.bentaib@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

29

ADVANCED MICRO REACTOR (AMR)

Author: Wayne Boyes[1]

Co-authors: Johan Slabber [2]; Jat DU TOIT [3]

[1] *STL Nuclear*

[2] *University of Pretoria*

[3] *North-West University*

Corresponding Author: wayne.boyes@thorium100.com

ADVANCED MICRO REACTOR (AMR)

W.A. Boyes, J.F.M. Slabber and F.A. van Heerden
STL Nuclear (Pty) Ltd. Prof C.G. DU TOIT
* Corresponding author: wayne.boyes@thorium100.com

Abstract - South Africa requires safe affordable distributed base load energy, one way to achieve this is to use nuclear power integrated with renewable energy sources on a decentralized basis. This suggests the development of its own micro modular nuclear reactor, to supply energy to towns, small communities, mines and processing plants. Large Light Water Reactors (LWRs) are expensive and require a large infrastructure development. A High Temperature Reactor (HTR) called the Advanced Micro Reactor (AMR) is in the process of being developed and the design philosophy is to design for inherent safety, maximally using technology that has been developed and validated in previous HTR programs albeit in a completely different and unique configuration. The concept is based on existing knowhow and experience/expertise in South Africa during the time of the Pebble Bed Modular reactor (PBMR) project. These AMR reactors are to be factory built to obtain good quality control and rolled out to various sites. Once the reactor has reached its end of life, it would be returned to a licensed organisation for refuelling. The AMR produces 10MW of thermal power. The reactor configuration uses hexagonal graphite blocks for structural and moderator material, which are arranged to form a cylindrical core layout. The fuel assemblies are silicon carbide tubes that house coated particle fuel, immersed in a lead-bismuth eutectic alloy (LBE). Each fuel assembly is contained in a boring within the graphite moderator that allows an annulus for cooling. There are 420 fuel assemblies in the core. Low enriched fuel in the form of UO2 or UCO is used. Helium gas is used as coolant. The coolant enters the core at 450˚C and exits at 750˚C. The mechanical, neutronic and thermal-hydraulic design of the AMR, is being evaluated with assistance from STL Nuclear (Pty) Ltd., the University of Pretoria (UP), the North-West University and the South African Nuclear Energy Corporation (NECSA). The OSCAR-5 code package, together with the Serpent neutronic code were used to perform the basic neutronic studies while the Flownex package was used to determine the thermal-hydraulic and safety evaluation for the Design Base Accident (DBA) specifically the Depressurized Loss of Forced Cooling (DLOFC) event.

Key Words: Advanced Micro Reactor (AMR), High Temperature Reactor (HTR), Multi-group Time Dependent Neutronics and Temperatures (MGT), Overall System for the Calculation of Reactors (OSCAR5), Nuclear Energy Corporation of South Africa (NECSA), Very Superior Old Program (VSOP).

Email address:

wayne.boyes@thorium100.com

Confirm that the work is original and has not been published anywhere else:

YES. Was shown at HTR2021 but not published

30

A COMPREHENSIVE THERMO-HYDRAULIC NEUTRONIC AND SAFETY ANALYSIS OF A 100MWth PEBBLE BED REACTOR CORE

Author: Wayne Boyes[1]

Co-author: Johan Slabber [2]

[1] *STL Nuclear*

[2] *University of Pretoria*

Corresponding Author: wayne.aboyes@gmail.com

This study investigates the neutronic and thermal-hydraulic modelling of a 100MWth PBR called the HTMR100 which has a core diameter of 2.6 m, a height of 5.226 m, and a core volume of 27.746 m3 producing a core power density of 3.6 MW/m3 with an outer Reactor Pressure Vessel (RPV) diameter of 5.25 . The 100MWth limit ensures that the core power density is kept low.

This reactor uses uranium dioxide fuel (UO2) at 10wt% enrichment (U-235 content) and 10 g of heavy metal in a Fuel Sphere (FS). In this reactor the fuel spheres pass through the core only once, this is known as a Once-Through-Then-Out (OTTO) fuel cycle, this cycle was also chosen for the simplicity of the fuel handling system.

The neutronic and thermal hydraulic design of the HTMR100 reactor was assessed. A transient safety analysis was also performed to determine if the reactor design allows for the fuel temperatures to remain below the set value of 1600˚C for the oxide-based UO2 fuel during design-based accident events e.g. a Loss Of Coolant Accident (LOCA).

The study assessed the core operating at normal conditions. The reactor has been operating continuously at 100% power which is 100MWth with the control rods situated at the nominal position called the equilibrium core position.

The equilibrium core was evaluated by the Very Superior Old Programs (VSOP99) suite of codes to obtain equilibrium burnup results. These results are then utilized in a second code called Multi-group Time Dependent Neutronics and Temperatures (MGT). MGT assesses the dynamic transient results of the accident scenarios such as the Depressurized Loss Of Forced Cooling (DLOFC), Pressurized Loss Of Forced Cooling (PLOFC), a Load Following (LF), Control Rod Withdrawal (CRW) and a Control Rod Ejection (CRE) for the HTMR100 reactor core.

The HTMR100 reactor utilizing the specified uranium dioxide fuel for the enrichment and heavy metal loading specified did indeed produce the targeted 80 000 MWD/THM burnup for the OTTO fuel cycle. The study also proves that the VSOP99 and the MGT codes do in fact yield similar results for the HTMR100 with regards to fuel centerline temperatures, outer sphere surface temperatures as well as moderator temperatures for the postulated accident scenarios that were analyzed. The results also indicate that the design-based transient safety analysis simulations prove that the fuel temperatures to remain below the set value of 1600˚C for the oxide-based fuel.

The beyond-design base events which will rarely ever occur do however exceed the set value of 1600˚C but the fuel only heats up for short periods of time while the transient takes place. The probability that this will lead to fuel damage is estimated to be low since the time at temperature of the fuel is short. A temperature-volume analysis of these beyond-design base events shows that only a portion of the fuel would exceed these values. VSOP provides values of 1% to 2% of the entire fuel amount in the reactor would experience these temperatures.

Email address:

wayne.boyes@thorium100.com

Confirm that the work is original and has not been published anywhere else:

Yes. Was shown at HTR2021 but not published (will be modified)

31

Efficient and Effective Advanced Reactor Design and Licensing Basis Development Requires Codifying a Rationalist Approach

Authors: Brandon Chisholm[1]; Amir Afzali[1]

[1] *Southern Company*

Corresponding Author: bmchisho@southernco.com

For nuclear energy generating systems to remain an option for energy generation, a variety of designs/technologies are needed, and they must be commercially viable, socially acceptable, and reg-

ulatory licensable. The regulatory body and its regulatory framework are an important factor that impacts all three of these attributes; therefore, it is critical for the nuclear industry to work collaboratively with the regulator to ensure an agile, predictable, and resilient regulatory framework is available to facilitate efficient and effective licensing that is not a barrier to innovation by imposing unnecessary burden.

In the United States, the current licensing pathways for nuclear reactors are largely based upon a structuralist approach, which places emphasis on specific outcome requirements that have been prescribed. An example of this philosophy in the US Nuclear Regulatory Commission framework is the General Design Criteria, which prescribe functions and programmatic controls that are identified to be important to safety based upon industry experience with large light water reactors. This structuralist construct was developed over many years to provide licensing predictability while providing adequate technical basis for the regulator to discharge its regulatory obligation.

By contrast, a rationalist approach places emphasis on performance requirements and systematic processes to derive solutions that satisfy these performance requirements. Such an approach enables a regulatory framework to be agile, predictable, and resilient for a variety of technologies, not just technologies that have produced extensive operating experience. Such a framework also facilitates achieving the following goals:

1. Cohesive requirements and regulatory oversight programs to be set for all stages of a nuclear power plant life cycle.
2. Coherent and consistent requirements and regulatory oversight programs for different designs based on design specific safety margins, thereby minimizing unbalanced regulatory requirements for different designs resulting in unfair competitive advantage.
3. Development and deployment of owner-controlled programs to improve efficiency and effectiveness of programmatic requirements and regulatory oversight programs while reducing staffing level.

This paper will discuss the importance of a rationalist approach to the success of advanced reactors and provide examples of processes that support a technology-inclusive, holistic, and integrated framework, including those that produce a safety case that is risk-informed, performance-based, and supported by insights from Probabilistic Risk Assessment.

Email address:

bmchisho@southernco.com

Confirm that the work is original and has not been published anywhere else:

YES

33

Computer Aided Design and Simulation of Professional Hybrid Electrical Energy Backup System for Nuclear Facilities

Authors: Ashraf Aboshosha[1]; E. Solyman Ahmed[2]

[1] *Associate Prof. Atomic Energy Authority*

[2] *Dr., Atomic Energy Authority*

Corresponding Author: ashraf.shosha@gmail.com

The following article discusses how to use Computer-Aided Design (CAD) and Simulation to create a Hybrid Electrical Energy Backup System (HEEBS) for Nuclear Facilities (NF). This study paradigm struggles to improve performance across multiple criteria in technical, economic, and environmental arenas. The primary goal is to increase the safety of the nuclear facility and supports electrical energy backup system. A heterogeneous modular power backup system is presented to guarantee the electrical power of the facility in any circumstances. This modular backup system, HEEBS, includes; Superconducting Magnetic Energy Storage (SCMES), Magnetic Energy Generator (MEG), Battery bank, Diesel Generator (DG), Solar Photovoltaic Panels (PV), Public Grid (PG), and smart metering system. The proposed micro-grid system provides high-quality electrical energy while also ensuring the safety of the facility and the environment as well. The software programs utilized to create these CAD projects are Sketchup, Skelion, Matlab/Simulink, Proteus Design Suite, Lumion, and PVSyst were. Instead of employing expensive storage for this huge power, this concept uses MEGs to cover the power generation at emergencies. This study is given to promote renewable energy resources for nuclear facilities electrical energy sources and to introduce a new arena for Smart Grids (SG). A sample microgrid design for the accelerator building at the EAEA is presented in this paper.

Email address:

ashraf.shosha@gmail.com

Confirm that the work is original and has not been published anywhere else:

Yes

34

IAEA/SANIS Severe Accident Experiments, Codes and Training Simulators Database Part II: Overview of SANIS Severe Accident Codes

Authors: Ahmed BENTAIB[1]; Sanjeev Gupta[2]; Haseeb ur Rehman3[3]; Tatjana Jevremovic[3]

[1] *IRSN*

[2] *Becker Technologies*

[3] *IAEA*

Corresponding Authors: ahmed.bentaib@irsn.fr, h.u.rehman@iaea.org, t.jevremovic@iaea.org, gupta@becker-technologies.com

The IAEA Simulation and Experimental Analyses Network Information System (SANIS) database aims to collect worldwide available information from research and industry on severe accident codes that contribute to new knowledge on severe accidents in water cooled reactors. In this framework, the SANIS database is designed to provide complementary information to those issued from other relevant and related international activities.
The SANIS database provides collected information on severe accident simulation codes that are used widely around the world as well as on the codes under development. Information on the codes developed by organizations in Europe, USA, and China had been collected. The information has been consolidated according to codes' validation domain and their field of applications, i.e. thermohydraulics, hydrogen behavior, corium behavior, fission products, and accident management. In addition, the SANIS database provides an overview of the capacity of these codes to handle the reactors of existing technologies and those under development, in particular the small modular reactors (SMRs).
The present paper provides an overview of the consolidated information on severe accident codes and on their validation status, namely on the experimental facilities addressed in the SANIS database. The use of collected information for knowledge/expertise preservation as well as for establishing new international co-operations will be highlighted with some examples.

Email address:

ahmed.bentaib@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

35

IAEA Simulation and Experimental Analyses Network Information System (SANIS) Database: Severe Accidents Experimental Facilities, Codes and Education and Training Tools

Authors: Sanjeev Gupta[1]; Ahmed BENTAIB[2]; Haseeb ur Rehman[3]; Tatjana Jevremovic[4]

[1] *Becker Technologies GmbH*

[2] *IRSN*

[3] *International Atomic Energy Agency, Vienna, Austria*

[4] *IAEA*

Corresponding Author: h.u.rehman@iaea.org

The IAEA Simulation and Experimental Analyses Network Information System (SANIS) database aims to systematically collect worldwide available information from research and industry, laboratories on experimental facilities, and infrastructure relevant to the experiments that contribute to new knowledge on severe accidents in water cooled reactors. The SANIS database provides complementary information with regard to similar online databases prepared as part of other international organisation activities.

The SANIS database provides information on more than 60 experimental facilities belonging to organizations in Europe, India, Japan, Republic of Korea and China. The information has been consolidated based on their field of operation, i.e. thermal hydraulics, hydrogen behaviour, corium behaviour, fission products, and accident management relevant system investigation (e.g. spray, hydrogen recombiners, filtered containment venting system, in-vessel melt retention, pressure suppression pool). In addition to existing technologies, SANIS database also provides information on facilities suitable for investigations towards new technologies, such as light water small modular reactors and accident tolerant fuels.

The present paper provides an overview of the consolidated information on severe accident experimental infrastructure and their relevance to the severe accident analyses codes (e.g. for validation purpose) and learning (simulation) tools in the framework of SANIS. The use of collected information for knowledge/expertise preservation as well as for establishing new international co-operations will be highlighted with relevant examples.

Email address:

gupta@becker-technologies.com

Confirm that the work is original and has not been published anywhere else:

YES

36

Application of Probabilistic method in the EPZ determination of SMR

Authors: Ji XING[1]; Nan WU[1]; Na XUE[1]; Lin QIU[1]; Qun CAO[1]

[1] *China Nuclear Power Engineering Co., Ltd*

Corresponding Author: xuejoyjoy@sina.com

SMRs have lower fission product inventories, larger water inventory relative to core thermal power, and enhanced cooling capability through passive features. These features can eliminate some postulated initiating events and delay the fuel uncovering and concomitant fuel damage, resulting in a significantly reduced radiological release into the environment. In this case, a relative smaller EPZ is consistent with SMR design features, rather than the over-conservative large EPZ.
The plume EPZ size of current large PWRs is in the range of 7˜10km with an inner zone of 3˜5km as required in Chinese national standards. Moreover deterministic method based on does results are usually used in the existing EPZ sizing, which can not show the advantage safety feature for SMR. SMRs will be probably located nearby cities, the sizes of EPZ are critical prerequisites for the successful deployment of SMRs, so a reasonable conservative EPZ that commensurate with SMR designs shall be point out.
In this paper, the probabilistic method involving accident frequency is considered to be applied in the determination of EPZ. Take ACP100 SMR type as an example that, for probabilistic approach, level 2 PSA results are utilized. 8 release categories and their frequency of the level 2 PSA results for internal events of ACP100 are considered in this research.
In this probabilistic EPZ sizing method, complementary cumulative distributions (CCDF) are calculated, presenting the probability of exceeding specific doses at different distances. For the total spectrum of PSA, 2d effective dose, 7d effective dose and thyroid dose at different distances of different accidents are calculated, and the CCDFs are figured out by using the GILs as the specific dose limits. For more severe accidents of CFE, BP and CI, the 2d projected absorbed doses of different organs in different accidents are calculated.
The results show that based on the advanced design of integrated layout, multiple passive safety systems, and larger aerosol removal capability, the EPZ size for ACP100 can be significantly reduced by contrast with the current large PWRs. The recommended EPZ radius can be 200m in terms of stochastic effects, and may extend to 2.5km considering the deterministic effect of more severe accidents, which can be concluded to support the Emergency Simplification.

Email address:

xuejoyjoy@sina.com

Confirm that the work is original and has not been published anywhere else:

YES

37

Towards innovative reactors licensing - ALFRED approach

Author: MIRELA NITOI[1]

Co-authors: Michele Frignani [2]; Marco Caramello [2]; Giacomo Grasso [3]; Mariano Tarantino [3]

[1] *RATEN ICN*

[2] *Ansaldo Nucleare*

[3] *ENEA*

Corresponding Author: mirela.nitoi@nuclear.ro

Before having an operational reactor with an innovative design, it is mandatory to demonstrate to safety authorities the license condition compliance with the regulations and proper management of the safety. Even if the process itself seems clear, in case of innovative reactors, significant challenges may arise.
As the demonstrator of the Lead-cooled Fast Reactors (LFR) technology, ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator) aims to represent the solid cornerstone in LFR deployment strategy. Specific design solutions (e.g., no intermediate circuit, simplified and robust component design, passive heat removal systems), combined with an efficient thermal cycle, are used, aiming to provide an extensive proof of the safety performances envisaged for the LFR fleet.
Romania expressed in 2014 the availability to host and implement ALFRED, choosing as reference

site the Mioveni nuclear platform. Until the decision to implement ALFRED in Romania, the existing Romanian regulation framework was applied to obtain the license for construction and operation only for already validated reactor designs. There are no regulatory standards or policy statements specifically developed for GenIV reactors, and particularly for ALFRED design, and this leaves room for many uncertainties during the licensing process for the innovative reactors. Even from the beginning of the interactions, both parties, Romanian National Commission for Nuclear Activities Control (CNCAN) and FALCON (Fostering ALFRED Construction) Consortium, agreed that adding a preparation phase will be beneficial to the whole licensing process. Going through a pre-licensing stage should help in developing and using strong safety, security, and radiation protection arguments in the licensing application. Unfortunately, there is no clear guidance on how to proceed on this stage to reach the desired objective.
Wishing to shed somehow clarity in this new proposed phase of the licensing process, elaborating and agreeing on a plan for the proposed involvement of all actors of the licensing process was considered as necessary. The paper will present the shaping element for this phase and its specifics for ALFRED, specifying the initial planning for the process, the main milestones and expected outcomes. The on-going and future efforts to prepare the ALFRED licensing process and the staged approach proposed to be followed for the ALFRED licensing process will be highlighted.

Email address:

mirela.nitoi@nuclear.ro

Confirm that the work is original and has not been published anywhere else:

YES

38

IAEA/SANIS Severe Accident Experiments, Codes and Training Simulators Database Part I: Education and Training Opportunities

Author: Haseeb Ur REHMAN[1]

Co-authors: Tatjana Jevremovic [2]; Sanjeev Gupta [3]; Ahmed BENTAIB [4]; Waqar Riaz [5]

[1] *International Atomic Energy Agency*

[2] *IAEA*

[3] *Becker Technologies GmbH*

[4] *IRSN*

[5] *Pakistan Institute of Engineering and Applied Sciences*

Corresponding Author: h.u.rehman@iaea.org

The IAEA Simulation and Experimental Analyses Network Information System (SANIS) database assembles information on Member States programmes and activities related to study and analysis of severe accidents in nuclear reactors including the reference data for code development and assessment with supporting information and documentation, detailed information about relevant experimental facilities with references, and collection of severe accidents learning tools, such as but not limited to part task simulators and training course materials.
The input received as part of SANIS database development supports research, development and capacity building not only for existing water cooled reactor designs but also for advanced new technologies, including small modular rectors and microreactors of various designs. The education and training opportunities based on the SANIS database content are manifold, considering the wealth of information collected and consolidated on experimental research and analytical tools employed for severe accident analyses.
The present paper aims to give an illustration of education and training opportunities which could be developed or further expanded based on existing examples of hydrogen safety related training courses, part task simulators, and relevant publications.

ses, part task simulators, and relevant publications.

Email address:

h.u.rehman@iaea.org

Confirm that the work is original and has not been published anywhere else:

YES

39

Status of the H2020 McSAFER Project-Experimental and analytical investigations for the safety evaluation of water-cooled SMRs

Authors: Victor Sanchez Espinoza[1]; Heikki Suikkanen[2]; Ville Valtavirta[3]; Marek Bencik[4]; Sören Kliem[5]; Cesar Queral[6]; Anthime Farda[7]; Paul Smith[8]; Paul Van Uffelen[9]; Marcus Seidl[10]; Christophe Schneidesch[11]; Dmitry Grishchenko[12]; Hector Lestani[13]

[1] *Karlsruhe Institute of Technology (KIT)*

[2] *LUT*

[3] *VTT*

[4] *UJV*

[5] *HZDR*

[6] *UPM*

[7] *CEA*

[8] *Jacobs*

[9] *JRC Karlsruhe*

[10] *PresusenElektra*

[11] *TRACTEBEL*

[12] *KTH*

[13] *CNEA*

Corresponding Author: victor.sanchez@kit.edu

This paper presents the main technical goals of the H2020 project entitled "High-performance advanced methods and experimental investigations for the safety evaluation of generic Small Modular Reactors (McSAFER)". The focus is on both numerical tools based on multiphysics and multiscale methods for SMR-safety investigations and on the experimental program at three European facilities, namely the COSMOS-H at KIT, HWAT at KTH, and MOTEL at LUT, where safety-relevant thermal hydraulic experiments for the core and helical heat exchanger are performed. The different safety analysis methodologies are applied to four water-cooled SMR-designs (CAREM, SMART, F-SMR, and NuScale), specifically to evaluate the core, reactor pressure vessel and plant behaviour under selected transient conditions (REA, Boron dilution, ATWS, and MSLB). The paper will describe the current status of the numerical and experimental investigations and will discuss selected results. The dissemination and education and training activities of the project will also be mentioned and an outlook provided.

Email address:

victor.sanchez@kit.edu

Confirm that the work is original and has not been published anywhere else:

yes

40

U-Battery Enhancing safety through innovative design

Author: John Eldridge[1]

[1] *U-Battery Development Ltd*

Corresponding Author: john.eldridge@urenco.com

The U-Battery is a 10MWth High Temperature Gas-cooled Reactor (HTGR) being developed by U-Battery Developments Limited. The U-Battery will be deployed in a novel factory build configuration, based on well-established technologies first deployed in the Dragon reactor at Winfrith in the UK and confirmed operationally by larger HTGRs.

The purpose of U-Battery is to provide a flexible source of low carbon heat, electrical power or hydrogen to serve energy intensive industrial applications or remote communities over long periods of reliable autonomous operations. The U-Battery's small size and modular construction will enable rapid build and deployment and its high level of inherent safety independent of complex engineered systems, will ensure negligible radioactive release under the most severe accident conditions. The paper will provide examples of the U-Battery project's approach to evolutionary and innovative reactor design including;

• The deployment of the fuel handling and maintenance systems across multiple U-Battery HTGRs.
• Remote monitoring and inspection of critical primary systems
• Designing for the planned exchange of life limited plant and equipment.

In adopting these technologies with others, U-Battery will deliver a product that presents a low risk of encountering insurmountable issues during the development of the design, safety case and subsequent licensing.

Email address:

john.eldridge@urenco.com

Confirm that the work is original and has not been published anywhere else:

YES

41

Evolution in Level 3 Probabilistic Safety Assessment Methodology for the UK-EPR

Authors: Neil Harman[1]; Liz Grindon[1]; Matthew Sugden[2]; Chris Hills[2]

[1] *Jacobs*

[2] *EDF Energy*

Corresponding Author: neil.harman@jacobs.com

The evolution of the EPR design is briefly described discussing in particular design features implemented to mitigate severe accidents.
A brief history of the Hinkley Point C project (HPC) in the UK is given discussing in particular the licensing steps and the timeline so far and for completion of the project. How lessons learned and experienced gained might be carried forward to other projects is also discussed. Links with other international EPR projects are also mentioned.
The regulatory framework in the UK is briefly described and how this was addressed in the regulatory submissions for HPC; this includes the assessment of individual and societal risk.
The Level 1 and 2 PSA performed for HPC are briefly described. How some of the design features

mentioned above were modelled in determining the source terms is discussed along with the Level 2/Level 3 interface that was developed for the HPC Level 3 PSA.
Level 3 PSA endpoints or metrics are discussed in general including some of the issues that need to be addressed in the analysis such as dealing with countermeasures, selecting a representative person for whom the individual risk or dose calculations are performed, determining an appropriate level of conservatism in the many steps of the analysis (including for example applying the Linear No Threshold (LNT) approach), and selecting an appropriate geographical and temporal scope.
The Level 3 PSA methodology developed for HPC is described including how the issues mentioned above were addressed. Some example results are presented showing the assessment against the individual and societal risk numerical targets.
The initial licensing began in 2010 with the Level 3 PSA described above completed in 2017. Since then, there have developments in the Level 3 PSA software available including, for example: more advanced atmospheric dispersion models, more flexibility in modelling extended duration releases, the capability to model releases from multiple locations, and the use of Geographical Information Systems. Some of the results of the original Level 3 PSA are reassessed using the latest software – and developments in the methodology allowed by this software – and the results compared. Applying these new methodologies and the potential impact on the results to future projects is also discussed.
Finally, the benefits of performing a Level 3 PSA for any nuclear installation – including, for example, multi-unit facilities, SMRs, Gen IV concepts and other innovative designs ¬– are discussed and how Level 3 PSA might overcome some of the difficulties in performing only Level 1 or 2 PSA for such facilities (for example how to determine a core damage frequency for a multi-unit site or for some innovative fuel concepts).

Email address:

neil.harman@jacobs.com

Confirm that the work is original and has not been published anywhere else:

YES

42

Regulatory Experiences in Licensing of Advanced Reactor Technologies

Author: Nasir Mughal[1]

Co-authors: Siraj Ahmed [1]; Muhammad Arif Javed [1]

[1] *Regulator*

Corresponding Author: nasir.mughal@pnra.org

Rapid technological advancements in the world brought forward several types of NPPs having innovative and evolutionary design features. Nuclear regulators worldwide are increasingly facing different challenges in licensing the NPPs with novel technical features. Owing to this, regulators have to modify existing regulatory processes in order to be commensurate with these advancements and bring their processes at par with new needs. Recently, PNRA has completed review and licensing of K-2 and K-3 NPPs which are HPR1000 (Gen-III+) design reactors. K-2 and K-3 are first of their kind reactors built and operated in Pakistan, with the introduction of various passive safety features along with active systems and design improvements to cope with design extension conditions based on feedback of Fukushima Accident. Licensing of these NPPs was taken-up as a challenge by PNRA due to the fact that design was improved from conventional PWRs. During licensing, PNRA devised a novel methodology for review of new features and dimensions associated with these NPPs focusing on three challenges: review of passive safety features; review of technical specifications and licensing examination of control room operating personnel.
This paper comprehensively elaborates PNRA approach for licensing of these NPPs, challenges faced and antidote to cope such challenges.

Email address:

nasir.mughal@pnra.org

Confirm that the work is original and has not been published anywhere else:

YES

44

Neutron data library for lead-cooled fast reactor precision calculations

Authors: Andrei Moiseev[1]; Andrey Zhirnov[2]; Ivan Rozhdestvenskiy[3]; Kseniya Kalugina[3]; Varvara Yufereva[None]; Vladimir Davydov[3]

[1] *JSC NIKIET*

[2] *NIKIET*

[3] *Federal State Unitary Enterprise "N.A. Dollezhal Research and Development Institute of Power Engineering" (NIKIET)*

Corresponding Author: wenddy@ya.ru

In Russia, the construction of the world's first fast power reactor with lead coolant and nitride fuel continues. New core materials not used in power reactors before as well as tight restrictions on the reactivity margin (<βeff) during the refueling interval have stepped up the requirements for the choice of neutron data in the design.
MCU-BR precision software package based on the Monte Carlo method is used to carry out neutronics calculations of a lead-cooled reactor. The absence of approximations in solving the neutron transport equation, a detailed description of the core design, an accurate determination of thermophysical parameters and a careful approach to the selection of the constant base make it possible to create a digital twin of the nuclear reactor. One of the ways to reduce the constant error is to create problem-oriented libraries based on evaluated neutron data files.
The MCU-BR SP constant base is an MDBBR50 databank created on the basis of the actual ROSFOND-2010, ENDF/B-VII.1, JENDL-4.0 evaluated neutron data files and supports modeling of neutron interactions with matter in the energy range from 0 to 20 MeV.
This work covers computational modeling of more than 50 critical experiments carried out at research reactors and critical experimental facilities with fast neutron spectra (SNEAK, JOYO, BFS-1, BFS-2, ZPPR, ZPR). Critical experiments were selected based on the closeness of neutron spectrum characteristics to the facility being designed. Plutonium and uranium-plutonium fuel was used in 43 critical experiments. Lead coolant was simulated in 16 experiments. Nitride uranium-plutonium fuel was simulated in 10 experiments. And only one experiment (BFS-113-B) combined all the above features. The computational models of the experiments have been developed with the maximum detail of the experimental systems. When modeling cores, the level of detail was individual pellets and cladding in the BFS experimental facilities and a plate in ZPPR/ZPR.
When carrying out computational modeling, along with the MDBBR50 library, we used neutron cross sections from all modern evaluated neutron data files: ROSFOND-2010, ENDF/B VII.1, ENDF/B-VIII.0, JEFF-3.2, JEFF-3.3, JENDL-4.0, CENDL-3.1.
For all calculated critical experiments the minimum calculation uncertainty was obtained using MDBBR50 and ENDF/B-VII.1, RMSD is 0.25% ΔK/K. Based on a set of calculated critical experiments with plutonium and uranium-plutonium fuel, the minimum calculation uncertainty was obtained using MDBBR50, RMSD is 0.17% ΔK/K. For a set of experiments in which lead coolant was simulated, the minimum calculation uncertainty was also obtained using MDBBR50, RMSD is 0.14% ΔK/K. For a set of experiments with nitride fuel, the minimum uncertainty was obtained using JENDL-4.0 and MDBBR50, RMSD is 0.08% ΔK/K and 0.10% ΔK/K, respectively.
In 2022, a full-scale modeling of the core of the lead-cooled fast reactor is scheduled at the BFS-2 experimental facility.

Email address:

wenddy@ya.ru

Confirm that the work is original and has not been published anywhere else:

YES

45

System of design codes for the computational modeling of the lead-cooled fast reactors

Author: Evgenii Soldatov[None]

Co-authors: Alexei Balovnev [1]; Vladimir Davidov [1]; Andrei Zhirnov [1]; Anastasia Ivanyuta [1]; Andrei Moiseev [1]; Dmitrii Rodnev [1]; Ivan Rozhdestvenskiy [1]; Varvara Yufereva [1]

[1] *JSC NIKIET*

Corresponding Author: eo-soldatov@mail.ru

A project is currently under way to build the BREST-OD-300 fast neutron lead cooled reactor as the pilot and demonstration prototype of base commercial reactor facilities for the future nuclear power with a closed nuclear fuel cycle. BR-1200 is a competitive commercial unit of the electric power 1200 MW, is under design for the large-scale evolution of nuclear power meeting the current requirements to reactors of a new generation. Solving complex problems in searching for the best reactor core configurations of BREST-OD-300, BR-1200 requires developing a system of design codes to perform physical design and justification of safety.
To carry out computational studies of the BREST reactor core, a design code system is developed, which includes the diffusion software package, the software package based on the Monte-Carlo method, and the thermophysical module. The system of codes comprises the following software packages, which interact through the input and output data management module:
– FACT-BR diffusion neutronic SP
– CONSYST neutron cross-section preparation system
– MCU-BR precision neutronic SP
– IVIS-BR thermophysical module
– Advanced burn-up module
– BREST platform
Primarily, there is a need for a tool offering high rate and multivariance of calculations. A diffusion neutronic software package FACT-BR allows us to carry out rapid estimation of the reactor condition. Extending the capabilities of a diffusion software package in terms of increasing its accuracy and restoration of fuel element-wise power density, taking into account the interaction photons with the reactor materials, a certified precision software package
MCU-BR is required. Current requirements to the development of the reactor facility design demands coupled consistent neutronic and thermophysical calculations. A thermophysical module IVIS-BR has been developed to estimate the thermophysical properties of the reactor core. The developed system of codes allows us to conduct multivariate calculations with high rate, while maintaining the accuracy of precision calculations. The presence of a functional interface in the system of codes makes it possible to perform consistent calculations in the diffusion and precision approximations.
The paper presents a scheme for the implementation of the system of codes and a demonstration of the capabilities of the system of codes for the modelling the reactor campaign with refueling of fuel assemblies, including in the closed nuclear fuel cycle mode. Using the system of design codes, the BREST-OD-300, BR-1200 reactors campaign were simulated for the entire service life.
In the course of computational modeling, the ranges of changes in fuel density and enrichment, reactivity margin, core reproduction rate and isotopic composition of plutonium were determined. Fuel and absorber burn-up, power distribution in the core, reactivity effects were calculated. The assessment of the fuel element-wise power density, maximum fuel cladding temperature and residual energy release of the fuel assemblies the in-reactor storage were carried out. CPS working member shutdown systems are sufficiently worth to ensure reactor transition into a subcritical state and to keep it subcritical in accordance with the regulatory requirements.

Email address:

eo-soldatov@mail.ru

Confirm that the work is original and has not been published anywhere else:

YES

46

The Development and Validation of the Accident Prevention and Mitigation Strategy for the ACP100 Modular Small Reactor

Authors: dang gaojian[1]; DanRong Song[None]

[1] *Nuclear Power Institute Of China*

Corresponding Author: danggaojian@163.com

ACP100 is a multi-purpose modular small PWR developed by China's national nuclear corporation. It is an innovative reactor based on existing PWR technology, adapting the integrated reactor design technology and the passive safetysystem. ACP100 is currently under construction in Changjiang, Hainan Province, China. This paper mainly introduces the process of the development and validationof the ACP100 accident prevention and mitigation strategy. These accident treatment approaches have been researched from the aspects of residual heat removal, safety injection, reactivity control, radioactive releases limit and severe accidents mitigation. Based on these researches, a series of passive safety systems have been designed to realize the accident prevention and mitigation functions. To verify the propose safety systems and validate the prospective functions, ascaled multi-system coupling experimental facility was constructed. Some typical accidents such as LOCA and SBO were performed in the facility. An overall safety evaluationwhich combines the deterministic analysis and probabilistic safety assessment wasconducted to confirm the safety of the ACP100. The experimental and analysis results indicate that the passive safety features are effective to the accident treatment and keep the ACP100 at high safety level.

Email address:

danggaojian@163.com

Confirm that the work is original and has not been published anywhere else:

YES

47

Enhancing a common understanding on the importance of nuclear 3S (safety, security and safeguards) in developing and deploying advanced reactor technologies

Author: Jorshan Choi[1]

[1] *Lawrence Livermore National Laboratory (retired)*

Corresponding Author: jorshan@yahoo.com

To meet an increasing demand of electricity without producing additional greenhouse gases, a new generation of advanced reactor technologies are proposed and relevant stakeholders (e.g., designers, investors, regulators, operators, and sovereign nations) are now actively involved in developing and deploying these new technologies. To ensure that these advanced reactors are safe, secure,

and for purely peaceful purposes, the nuclear 3S (safety, security, and safeguards) are important considerations in the successful development and deployment of such technologies.
When 3S is considered holistically, as illustrated in Figure 1, the interfaces (or focuses) between each of the 2S pairs are: facility (safety–security), materials (security–safeguards), and technology (safeguards–safety). These destinations of interfaces are not always so rigid, for example, safety, beside its focuses on facility and technology may also have a "materials" focus when criticality safety is considered in a material-processing facility. Also, the IAEA Safeguards, beside its materials and technology focuses, may also include "other facilities," for instance in its request for complementary access during a routine inspection.

FIG.1 A holistic illustration of nuclear 3S (safety, security, safeguards).

The interfaces depicted in Figure 1 lay the groundwork for a holistic approach to integrate 3S in the evaluation of advanced reactor technologies during their development and deployment phases. Due to the respective operating principles of 3S, there are synergies and challenges in the 3S interfaces. This paper will address these synergies and challenges for the advanced reactor technologies, and show that a holistic 3S approach is the common ground for international transparency and confidence in deploying advanced reactors and for the sustainable use of nuclear energy.

Email address:

jorshan@yahoo.com

Confirm that the work is original and has not been published anywhere else:

yes

48

Current approaches to the analysis of design for new Nuclear Power Plants

Author: Oleksandr Balashevskyi[1]

[1] *SE "Scientific and Technical Center" of SE NNEGC "Energoatom"*

Corresponding Author: balaas@ua.fm

Over the past several years, a number of technology developers have expressed interest in the possible construction and deployment of evolutionary and innovative reactor designs.

Innovative reactor are designed with enhanced safety features based on lessons learnt from past experience of plant operation. Reliance on natural circulation and addition of passive safety systems made them inherently safe and simple in design.

However, compliance with the safety objective may significantly vary depending on the safety options of the proposed design and the targeted applications and thus it needs to be further investigated. The focus should be given to support licensing process, including of safety assessment (deterministic and probabilistic safety analysis, possible risks, human factors, regulatory requirements and etc.).
To demonstrate achievement of the safety objectives, a comprehensive hazard analysis,a deterministic safety analysis, and a probabilistic safety assessment are traditionally carried out.

The key areas of Deterministic Safety Analysis (DSA) and Probabilistic Risk Assessment (PRA) use is in the development of methodologies and tools that will be used to predict the safety, security, safeguards, performance, and deployment viability of evolutionary and innovative reactor systems starting in the design process through the operation phase. Development and implementation of safety assessment methods may require new analytic methods or adaptation of traditional methods to the advanced design and operational features of evolutionary and innovative reactor design. The development of specific safety models for margin determination will provide a safety case that describes potential accidents, design options (including postulated controls), and supports licensing activities by providing a technical basis for the safety envelope.

Availability best estimate system codes are playing significant role in assessment of safety as well as in design, certification and evaluation of these innovative types of reactors.
Consequently, it is required to study assessment of safety during postulated transients prior to their deployment on commercial scale.

Email address:

o.balashevsky@ntc.atom.gov.ua

Confirm that the work is original and has not been published anywhere else:

YES

50

NUGENIA/TA2 Recent achievements in Severe Accidents Research

Authors: Fabrizio Gabrielli[1]; IRSN-Ahmed BENTAIB[None]; Luis Enrique HERRANZ[2]; Federico ROCCHI[3]; Ivo Kljenak[4]; Pascal Piluso[5]; Sanjeev Gupta[6]; sandro PACI[7]

[1] *KIT*

[2] *CIEMAT*

[3] *ENEA*

[4] *IJS*

[5] *CEA*

[6] *Becker Technologies*

[7] *UNIPI*

Corresponding Authors: sandro.paci@unipi.it, luisen.herranz@ciemat.es, pascal.piluso@cea.fr, federico.rocchi@enea.it, gupta@becker-technologies.com, ivo.kljenak@ijs.si, ahmed.bentaib@irsn.fr, fabrizio.gabrielli@kit.edu

Severe accident research is the only way to achieve the best possible management in case such unlikely events occur. Early this century, the Severe Accident Research NETwork of Excellence (SARNET) was born as an EC project and a decade later it became the Technical Area 2 (TA2) of NUGENIA, the SNETP (Sustainable Nuclear Energy Technology Platform) pillar devoted to research on Gen. II and Gen. III Light Water Reactors (LWRs). During these years, NUGENIA TA2 has produced meaningful advances in methodologies used for managing severe accidents, from enabling severe accident analytical tools to deepening the performance of mitigation devices, passing through extension of some databases and exploration of new methodologies, especially thanks to the different EURATOM Framework programs. At least part of this progress is experiencing its application to upcoming nuclear systems.
This paper describes the most recent achievements from NUGENIA/TA2 research related to the main phenomena occurring during a severe accident (as the coolability of in-vessel and ex-vessel corium/debris, the different in-containment phenomena and the estimation of the source term); besides, the progress made and underway on severe accident modelling is outlined. Finally, the NUGENIA/TA2 commitment to knowledge dissemination through courses and conferences is highlighted.

Email address:

ahmed.bentaib@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

51

Recent Improvements of ATHLET Models for Passive Safety Features of LW-SMR

Author: Sebastian Buchholz[1]

Co-authors: David Sonntag [2]; Julia Krieger [3]; Sinem Cevikalp Usta [4]; Nelson-Felipe Rincón-Soto [4]

[1] *Gesellschaft für Anlagen und Reaktorsicherheit (GRS) gGmbH*

[2] *Ruhr Universität Bochum PSS*

[3] *Ruhr Universität Buchum PSS*

[4] *Universität Stuttgart IKE*

Corresponding Author: sebastian.buchholz@grs.de

The paper will give an overview about the current running joint German project VASiL (Enhancement and Validation of AC2 for the Simulation of innovative LW-SMR) enhancing the system code ATHLET, part of AC2, to better simulate passive safety features of LW-SMR designs. Work in this project is performed by GRS, Ruhr-Universität Bochum PSS and University of Stuttgart IKE.
In the paper two aspects explained in more detail. First, ATHLET models for compact heat exchanger geometries (of plate type, helical coil, bayonet type and loop heat pipes) have been tested against relevant experiments and improved correlations for heat transfer and pressure drop for such systems have been implemented and validated. Second, we have investigated heat transfer during free convection at high containment wall (e.g. up to 20 m in the NuScale case) to an external water pool. As current correlations in the core and literature are validated only for Rayleigh numbers up to 1012 (or about 2 m height) and no dedicated experiments are available, we performed CFD-calculations and derived local and integral Nusselt numbers. These are then compared to ATHLET models and potential code and model improvements are investigated.

Email address:

sebastian.buchholz@grs.de

Confirm that the work is original and has not been published anywhere else:

YES

52

Potential Gaps in Safety Demonstration Methods for LW-SMR Identified in the ELSMOR Project

Author: Sebastian Buchholz[1]

Co-author: Andreas Wielenberg [2]

[1] *Gesellschaft für Anlagen und Reaktorsicherheit (GRS) gGmbH*

[2] *GRS gGmbH*

Corresponding Author: sebastian.buchholz@grs.de

Upcoming SMR designs possibly deployed in the EU provide special safety features, which can pose challenges to established safety demonstration methodologies. The EU-funded ELSMOR (towards European Licensing of Small MOdular Reactors) project joined 15 partners with the aim to create methods and tools for the European stakeholders to assess and verify the safety of light water small modular reactors (LW-SMR) that would be deployed in Europe.
In ELSMOR, potential gaps in established safety demonstration methodologies introduced by new safety features and specific detail of new LW-SMR designs were identified for relevant aspects of a

facility safety case. In this paper, we present selected examples of particular interest related to the topics reactivity control, decay heat removal, containment integrity, decommissioning, refuelling, spent fuel management, transport and disposal, multi-unit site and sharing of systems issues, severe accident management and emergency planning as well as operation and human factors.

Email address:

sebastian.buchholz@grs.de

Confirm that the work is original and has not been published anywhere else:

YES

53

Results of a quantitative comparison of different nuclear fuel cycles by their safety

Authors: Albert Shapovalov[1]; Angelica Khaperskaya[2]; Anton Kuryndin[1]; Anton Ponizov[1]; Elena Suvorova[2]; Konstantin Ivanov[2]; Lev Belinsky[2]; Maksim Felicin[1]; Rashet Sharafutdinov[1]; Roman Poliakov[1]

[1] *Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS)*

[2] *State Atomic Energy Corporation Rosatom (ROSATOM)*

Corresponding Author: ashapovalov@secnrs.ru

The Russian Federation is implementing the closure of nuclear fuel cycle (NFC), which provides for the parallel operation of thermal and fast neutron reactors. The use of new BN-1200 reactors is also envisaged in these activities. A transitional stage to two-component NFC can be considered the stage with the closure of the NFC in the power system, consisting of thermal reactors, due to the use of mixed uranium-plutonium fuel (REMIX fuel).

The energy strategy of the Russian Federation provides for an increase in electricity production at nuclear power plants in the Russian Federation, and, as a result, the increase in the number of operating power units.

The increase in the generation of radionuclides in nuclear fuel, inevitable in such conditions of the development, will lead to the increase of exposure of personnel, the population and the environment.

There are many researches devoted to the economic aspects of the development of the NFC with the transition to a closed NFC, where the pros and cons of choosing an open or closed NFC are considered. However, due attention is not paid to safety issues, when choosing between the development of one or another NFC option. To fill this gap, the authors developed a methodology for a quantitative comparison of possible NFC options by their safety (see figure) with account of recommendations of the ICRP and the NEA OECD.

The paper presents the results of a comparison of various NFC options by their safety, possible in the Russian Federation, and the open NFC, which has become widespread in the world. This comparison has been made, using the developed methodology.

Indico rendering error

Could not include image: Problem downloading image (https://b.radikal.ru/b02/2201/4b/2fa80

Email address:

shapovalov@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

No

54

Evaluation of the innovative steam generators passive heat removal system efficiency for VVER-1200 power units in beyond design basis accidents conditions

Authors: Aleksei Samokhin[1]; Nadezda Kozlova[1]; Sergei Katkovskii[1]

[1] *SEC NRS*

Corresponding Author: asamokhin@secnrs.ru

This paper presents the results of the independent assessment of the steam generators passive heat removal system (PHRS) efficiency for the NPP unit with VVER-1200 reactor (V-491) during beyond design basis accidents.
In SEC NRS the calculation full-scale model of the water-type passive heat removal system through a steam generator was developed and was implemented into VVER -1200 reactor (V-491) calculation model. All models were developed using the best-estimated code ATHLET 3.1A (GRS, Germany).
Since PHRS is an innovative safety system, the independent confirmation of its effectiveness in a wide range of emergency modes is especially important.
PHRS is able to remove the heat from the reactor core due to the natural circulation of the coolant for a long period of time during beyond design basis accidents including the case of NPP unit total blackout.
The report contains a brief description of the PHRS and its calculation model, as well as the results of independent calculation analyzes of the beyond design basis accidents, confirming of the PHRS efficiency.
This paper demonstrates the possibility of technical and scientific support organization of the Russian nuclear regulation authority (Rostechnadzor) to carry out its own independent scientific and technical assessments during licensing process of innovative reactor designs and this complies with the provisions of the IAEA safety standards GSR part 4 and GSG-13.

Email address:

asamokhin@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

YES

55

Piloting a Generic Process for Risk-Informed Evaluations

Authors: Andrea Maioli[1]; Carroll Trull[1]; Hartz James[1]; Nathaniel Hall[2]; Raymond Schneider[1]; Reed Labarge[1]; Roy Linthicum[2]; Sean Kinnas[1]

[1] *Westinghouse Electric Company, LLC*

[2] *PWROG*

Corresponding Author: maiolia@westinghouse.com

The PWR Owners Group (PWROG) has been developing an approach that uses the definition and characterization of Licensing Basis Events (LBE) on the frequency/consequence space developed in NEI 18-04 for advanced reactors to explicitly quantify the conservatism associated with an operating plant Design Basis Event (DBE) with a Main Steam Line Break (MSLB) event with coincident Loss of Offsite Power (LOOP) and a postulated single failure of mitigation system that could result in a calculated dose exceeding the regulatory thresholds due to the need for a prolonged cooldown strategy. With this approach, that marries the risk-informed decision-making principles based on Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) metrics with the NEI 18-04 LBE definitions, this DBE is showed to be of extremely low risk significance and to be a Beyond Design Basis Event (BDBE) per the NEI 18-04 definitions. The PWROG is using this approach to develop and pilot a generic version of the NRC Risk Informed Process for Evaluations (RIPE) process that allow plants to use realistic assumptions for low-risk significance scenarios in showing compliance with regulations.

Email address:

maiolia@westinghouse.com

Confirm that the work is original and has not been published anywhere else:

YES

56

The Russian approach to the regulatory review of computer programs used for multi-physics modelling and safety analysis of innovative nuclear installations

Authors: Sergey Bogdan[1]; Sergey Shevchenko[1]

[1] *SEC NRS*

Corresponding Author: s.a.shevchenko@gmail.com

Safety analysis of innovative nuclear installations using modern computational computer programs, including those based on the multiphysics simulation and artificial intelligence (such as, artificial neural networks), requires not only the use of special approaches to verification, validation and uncertainty analysis of such computer programs, but also the established approaches to the review of such computer programs by the regulatory authorities.
The paper contains a description of the Russian procedure for the review of computer programs used, for licensing calculations of innovative nuclear installations, contains information on current Russian requirements in the field of safety calculations, verification and validation of simulation models. It is also describes various methods for analyzing the uncertainties which are recommended for the design and beyond-design accidents simulation of nuclear power plants.

Email address:

sshevchenko@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

YES

57

The Effect of Intermittent Passive Heat Removal on HTGR Conduction Cooldown Performance

Authors: Brian Mays[1]; Lewis Lommers[2]; Farshid Shahrokhi[2]; Stacy Yoder[2]

[1] *former Framatome*

[2] *Framatome Inc.*

Corresponding Author: finis.southworth@gmail.com

This paper evaluates the performance of Framatome's Steam Cycle–High Temperature Gas-Cooled Reactor (SC-HTGR) during a depressurized loss of forced circulation (DLOFC) event with the passive heat removal system temporarily out of service.
Like other modular HTGRs, the SC-HTGR is optimized to conduct heat out of the reactor to a reactor cavity cooling system (RCCS). The Framatome RCCS is a completely passive, fully redundant natural circulation heat removal system. Its performance is continuously monitored during normal plant operation. Therefore, RCCS failure is not credible for any accident scenario. Nonetheless, evaluation of a DLOFC event with a hypothetical RCCS failure is still interesting to confirm that no cliff edge effects occur for very low probability events (i.e. beyond design bases) and to support accident management, emergency planning, and investment risk requirement development.
Results are presented for several different scenarios involving RCCS failures at the beginning, middle, and end of the DLOFC event. The results confirm that there are no safety consequences even for such a non-credible event. Fuel integrity is not challenged. Some structural impact is possible, potentially effecting near-term restart of the plant. But the investment risk is still negligible given the extremely low probability of such a scenario.

Email address:

lewis.lommers@framatome.com

Confirm that the work is original and has not been published anywhere else:

YES

58

Integrated RAMI for Advanced Nuclear Power Plants

Authors: Stephen M Hess[1]; Paul Amico[None]; Jeffrey A Julius[1]

[1] *Jensen Hughes*

Corresponding Author: shess@jensenhughes.com

Advanced nuclear power plants are being developed with the objectives of providing highly safe, reliable, and economical electricity. To provide a substantial contribution to the transition to carbon free energy production, these plants will need to meet each of these objectives in a manner that is at least as economical as alternative energy sources. Because these plants represent first of a kind (FOAK) deployment of these technologies, application of an integrated reliability, availability, inspectability, and testability (RAMI) process will be needed to be incorporated from the design stage through the construction, commissioning, and operational phases of plant life.

In this paper, we describe an integrated RAMI process to support the development, licensing, and operation of advanced technology nuclear power plants. This program develops specific performance objectives related to plant safety, production, and economics against which the design and operation of plant structures, systems, and components (SSCs) will be measured. During the design and construction phases, RAMI assessments are performed to ensure plant SSCs are capable of achieving the identified performance objectives. At this stage the RAMI program provides input to support decisions that can impact the performance, reliability, and availability of plant SSCs that impact safety and production. Additionally, at this stage the RAMI analyses develop initial integrated life-cycle management plans to ensure safe and cost effective long-term plant operation. During the commissioning and operational phases, integrated RAMI executes and monitors plant testing and

maintenance programs that ensure plant safety and operational objectives are met throughout the plant operating lifetime.

Email address:

pamico@jensenhughes.com

Confirm that the work is original and has not been published anywhere else:

Yes

59

New test plan about safety features of HTGR by using HTTR in JAEA

Author: Taiju Shibata[None]

Corresponding Author: shibata.taiju@jaea.go.jp

Japan Atomic Energy Agency (JAEA) has High Temperature Engineering Test Reactor (HTTR) which is 30MW test reactor of HTGR. Although the HTTR was not operated after the Fukushima Daiichi accident in 2011, it restarted operation in July 2021. The HTTR can be used to carry out various tests about safety features of HTGR. In 2010, safety demonstration test about a loss of forced cooling (LOFC) condition was carried out under a framework of OECD/NEA. Two LOFC tests with different conditions will be done in early 2022. The LOFC project is an international joint project. After the LOFC, JAEA is planning to launch a new test project. Test theme concerning Xe build-up/decay behavior after reactor scram is one of the attractive new test themes, but not limited to this. This paper presents possible test themes by using the HTTR. JAEA has room to discuss possible new themes with expecting foreign partners to launch an international joint project.

Email address:

shibata.taiju@jaea.go.jp

Confirm that the work is original and has not been published anywhere else:

YES

60

Safety design approaches for future SFRs in Japan

Author: Shigenobu Kubo[1]

[1] *Deputy Director Advanced Fast Reactor Cycle System R&D Center, JAEA*

Corresponding Author: kubo.shigenobu@jaea.go.jp

Sodium-cooled fast reactors (SFRs) are expected as nuclear energy systems that can provide sustainable, efficient use of uranium resources and reduction in the amount of radioactive waste and the potential waste toxicity, so research and development of SFRs are ongoing around the world. In Japan, research institutes and companies are striving to demonstrate SFR technology, based on deep insight and sophisticated technology obtained through the design, construction, and operation of the experimental fast reactor Joyo and the prototype reactor Monju. SFRs are designed to have safety features such as passive reactor shutdown and decay heat removal functions. Meanwhile, global safety criteria for SFRs have been established by the Generation IV International Forum (GIF), in

which SFR developing countries participate to formulate the Safety Design Criteria (SDC) and the Safety Design Guidelines (SDGs). This paper describes safety design approaches for future SFRs in Japan.
The SFR uses the primary cooling system, through which liquid sodium removes heat from the core. The core uses fission chain reactions of fast neutrons, it is designed to have inherent reactivity features. The primary cooling system is operated at low pressure and high temperature, requiring that the system design ensure structural integrity with materials resistant to thermal and seismic loads throughout the plant lifetime. SFR design is based on the defence-in-depth philosophy involving layers of protection such as proven active safety systems and passive mechanisms. The results of probabilistic risk assessment (PRA) as well as pervious licensing practice are reflected in the selection of potential initiating events and event sequences. Deterministic safety analysis is conducted to confirm the validity of safety related systems, structures, and components (SSCs) against design basis accidents. For design extension conditions, PRA serves to analyze complicated event sequences and identify complementary SSCs.
SFR designers have been evaluating core damage sequences resulting from anticipated transients without scram, while examining measures to achieve in-vessel retention of degraded core materials. In addition, they have been studying inherent reactivity feedback, passive reactor shutdown mechanisms, and thermal inertia of the coolant system to prevent core damage even after active safety systems fail. Considering that an SFR uses liquid sodium coolant, it should have static components such as guard vessels for maintaining the coolant level and maintain natural circulation of the coolant, so that decay heat can be removed even under complicated event sequences involving multiple failures. Accident management measures should also be taken in the design, considering the time margin until fuel damage. To establish the architecture of safety design, SFR designers should concern not only internal events but also external events such as an earthquake. The reactor structure is designed as a thin-walled structure, the cooling systems should have sufficient seismic margin against a design basis earthquake. SFR designers, therefore, perform seismic assessment using PRA methods to confirm that the margin is sufficient. Interface with security is implemented by arranging decay heat removal systems separately from each other, installing multiple systems in different locations, and providing compartments for SSCs containing sodium.

Email address:

kubo.shigenobu@jaea.go.jp

Confirm that the work is original and has not been published anywhere else:

YES

61

Uncertainty and Sensitivity Analysis of the QUENCH-06 Experiment by Means of Severe Accident Codes in the Framework of the IAEA CRP I31033 on Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors

Authors: FABRIZIO Gabrielli[1]; A. Stakhanova[2]; V.H. Sanchez-Espinoza[2]; N. Elsalamouny[3]; T. Kaliatka[3]; N.I. Ryzhov[4]; V.N. Semenov[4]; R.V. Chalyy[4]; F. Mascari[5]; A. Bersano[5]; P. Maccari[5]; T. Jevremovic[6]; D. Luxat[7]; L.I. Albright[7]; H.U. Rheman[6]; K.I. Ahn[8]; J. Ortiz[9]; S.M. Petoukhov[10]; J.M. Garcia[11]

[1] *Karlsruhe Institute of Technology*

[2] *KIT*

[3] *LEI*

[4] *IBRAE RAN*

[5] *ENEA*

[6] *IAEA*

[7] *SNL*

[8] *Kaeri*

[9] *ININ*

[10] *CNL*

[11] *CNEA*

Corresponding Author: fabrizio.gabrielli@kit.edu

The International Atomic Energy Agency (IAEA) launched in 2019 the Cooperative Research Project (CRP) I31033 entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors" [1]. The main goal is to advance the understanding and characterization of sources of uncertainty and to investigate their effects on the key Figure-Of-Merits (FOMs) of the Severe Accident codes (SA) predictions in Water Cooled Reactors (WCRs). Having this in mind, within the five years duration of the CRP, 22 organizations representing 18 Member States have been developing and accessing calculation platforms based on the currently-available severe accident codes and relevant Uncertainty Tools (UT) for Uncertainty and Sensitivity (U&S) analyses of severe accident scenarios in different WCRs, i.e. PWR, BWR, CANDU, VVER, and SMR.
In the framework of the CRP I31033, the U&S analysis of the QUENCH-06 experiment [2], performed on 13 December 2000, at the Karlsruhe Institute of Technology (KIT), by means of SA codes and UT platforms has also been performed to give relevant insights. The goal of such work is testing relevant calculation platforms in view of their applications to plant analyses supporting other CRP tasks. Four institutions participated in the QUENCH-06 test exercise, being the SA code/UT platform shown in parenthesis: ENEA (ASTEC/RAVEN), KIT (ASTEC/URANIE and ASTEC/In-house FSTC tool), IBRAE (SOCRAT/ELENA), and LEI (RELAP-SCDAPSIM/SUSA) [3].
The QUENCH-06 experiment aimed at investigating the coolability and at determining the hydrogen source term in a PWR pre-oxidized rod bundle quenched with water [2].
To perform the U&S test exercise, 23 input parameters required by the SA codes to model the QUENCH-06 experiment have been selected and the probability distribution functions of the corresponding uncertainties have been assessed. The selected parameters refer to the experimental boundary conditions, the geometrical modelling, the heat transfer phenomena, and the clad integrity criteria. Such uncertainties are propagated by means of the uncertainty tools to evaluate their effect of the following FOMs: hydrogen accumulated mass and generation rate, temperature of the central rod at 950 mm height, and axial profile of the oxide scale of the heated and corner rods.
In this paper, the results of the U&S analyses performed by the ENEA, KIT, IBRAE and LEI are described. The results showed that the employed severe accident codes can predict the experimental behaviour of the FOMs distinguished by CRP partners. Furthermore, the work activity revealed that the severe accident codes / uncertainty tool calculation platforms assessed by each institution are well characterized by high performance in terms of wall-clock time and accuracy. In particular, the results of the sensitivity analyses showed that the uncertainties of the boundary conditions, i.e. power and coolants flow rates, mainly affect the FOMs. The results of the QUENCH-06 test exercise represent therefore a solid basis of understanding in view of the application of the U&S approaches for SA analyses in WCRs.
[1] https://www.iaea.org/projects/crp/i31033.
[2] Sepold, L., et al., 2004, Report FZKA-6664, Forschungszentrum Karlsruhe.
[3] Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, IAEA-TECDOC-1872, Vienna, 2019.

Email address:

fabrizio.gabrielli@kit.edu

Confirm that the work is original and has not been published anywhere else:

YES

62

Integrated design analyses of beyond-design-basis accident at VVER-1200, including all stages of fuel severe damage.

Author: Daria Buzykina[None]

Co-authors: Nadezhda Kozlova [1]; Roman Fuks [1]

[1] *SEC NRS*

Corresponding Author: buzikina@secnrs.ru

A brief description and results of a pilot application of a VVER-1200 simulator intended for end-to-end integral simulation of emergency processes in VVER-1200 reactors from the initial event to the processes of nuclear fuel severe damage, including simulation the internal and external stages of the accident, as well as for simulation the behavior of the environment in the containment, are encapsulated in the report.
The NPP simulator provides simulation of the main systems of a VVER-1200 NPP power unit, including innovative passive systems. The calculation models of the simulator are based on simulator technologies. It makes possible to carry out calculation analysis of beyond-design-basis accidents in a reactor facility, both in real time and in an accelerated mode (5 - 7 times faster than real time). This makes it possible to use this simulator both for assessing the effectiveness of actions provided in the accident management procedures, and for predicting the development of an emergency in the information and analytical center of Rostekhnadzor, as a tool for performing quick assessment and forecast calculations.
Thus, a universal calculated tool has been developed that allows performing independent expert evaluation calculations as part of justification review of safety and predictive calculations in conditions of emergency training.

Email address:

buzikina@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

YES

63

Probabilistic AI for Prediction of Material Properties in Nuclear Reactors

Authors: Adolphus Lye[1]; Edoardo Patelli[2]; Prinja Nawal[3]

[1] *Institute for Risk and Uncertainty, University of Liverpool*

[2] *Centre for Intelligent Infrastructures, University of Strathclyde*

[3] *Prinja and Partners*

Corresponding Author: adolphus.lye@liverpool.ac.uk

This research work presents a probabilistic artificial intelligence framework aimed towards the prediction of material properties within nuclear power plants. Currently, 2 Artificial Neural Networks (ANNs) have been constructed aimed at predicting chemical properties as well as material behaviours (i.e. fatigue and fracture) due to environmental factors. However, these existing ANNS are trained with deterministic data and would require a large data-set in order to account for the inherent variability of the material properties.In the nuclear industry, data is sparse which presents a challenge in achieving the above task.

To address the above problem, Uncertainty Quantification (UQ) tools would be introduced in the form of Bayesian model updating to produce synthetic data which, in turn, will be used to train the ANNs and perform the necessary uncertainty analysis.In addition to this, the ANNs will also be validated to ensure that the results are consistent with the underlying physics, as well as its applicability in the nuclear industry.

Email address:

adolphus.lye@liverpool.ac.uk

Confirm that the work is original and has not been published anywhere else:

YES

64

Main lessons learned from the Clearinghouse topical operational experience report (TOER) on external hazards and their possible use for the design of new reactors

Authors: Abdallah AMRI[1]; Andrey RODIONOV[1]; Ján STULLER[2]; Marc FOLDENAUER[3]; Michael MAQUA[3]; Miguel PEINADOR[4]; Miroslav HREHOR[2]; Sadek BHAR[1]

[1] *Institut de Radioprotection et de Sûreté Nucléaire (IRSN)*

[2] *Statni Ustav Radiacni Ochrany v.v.i. (SÚRO)*

[3] *Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH*

[4] *Joint Research Centre (JRC) of the European Commission*

Corresponding Author: abdallah.amri@irsn.fr

The European Network on Operating Experience Feedback (OEF) for Nuclear Power Plants, or 'European Clearinghouse' (CH), was established by European nuclear safety regulators to promote the regional sharing of operating experience, the dissemination of lessons learned from nuclear power plants operation, and the understanding of the role of OEF systems in the safe and reliable operation of NPPs.
In the frame of the CH, a consortium constituted of GRS, IRSN and SÚRO performed an in-depth analysis of events related to external hazards that occurred or were reported during the period between 2017 and 2020 for external flooding events, and the period between 2010 and 2020 for the other external hazards events. Those events were selected either from the IRS database or from the national databases VERA, SAPIDE and DBEvents respectively; they were divided in nine groups:
- Group 1: extreme weather conditions;
- Group 2: external flooding;
- Group 3: earthquakes;
- Group 4: external fires;
- Group 5: lightning strikes;
- Group 6: fouling events with fouling of water intake entrance and biofouling caused by external environment;
- Group 7: chemical events comprising corrosion caused by external environment and chemical fouling caused by external environment;
- Group 8: man-induced events with effects of nearby industries (river/sea traffic hazards, air traffic hazards, road traffic hazards, fire and explosion as well as working in NPP vicinity);
- Group 9: other, including solar magnetic disturbances.

Indicative statistical analyses were performed with respect to the number of events of each group, the relative distribution by mode of event detection, the relative distribution of the safety relevance, and the relative distribution of the systems affected. However, the most important part of the in-depth analysis was the derivation of high-level lessons learned, which were derived from the most important events in the respective groups. To support or justify those high-level lessons learned, the relevant events were described in detail to illustrate the recommended actions including their purposes. In addition, the actual (observed) consequences motivating the recommended action were provided based on event specific causes or lessons learned. For each event group, the related recommendations have been assigned to 4 groups: (1) prediction and monitoring; (2) design and equipment related features; (3) procedures and training; and (4) review and management.
The paper aims at summarizing the main lessons learned from the Clearinghouse TOER on external hazards, with examples of lessons learned/recommendations that can be used for the design of new reactors.

Email address:

abdallah.amri@irsn.fr

Confirm that the work is original and has not been published anywhere else:

Yes

66

Probabilistic Safety Assessment of Multi Unit Site of Nuclear Power Plant

Author: Abdul Qavi[1]

Co-authors: Noman Shah [2]; Shah Waqar Hammad [2]

[1] *Pakistan Nuclear Regulatory Authority*

[2] *PNRA*

Corresponding Author: a.qavi@pnra.org

It is a common practice around the world to install multiple units of nuclear power plants on a single site taking advantage of common intake cooling structure, shared components, etc. The Fukushima Daiichi accident highlighted the risks from multiple nuclear reactor unit accidents at a site urging the nuclear community to analyze the multi-unit risks to avoid re-occurrence of Fukushima like accidents. A multi-unit site PSA model has to deal with the dependencies existing between the units on that site. Common cause models used in traditional single unit PSAs were originally developed for arrays of redundant components within a single nuclear power plant. If there are identical, redundant components and systems that are replicated on multiple reactor unit, the Common cause failure (CCF) that may impact components on different units needs to be considered.
This study is conducted to identify main risk contributors that delineate multi-unit risk from single-unit risks. A case study is performed based on full power level-I PSA model to identify plant vulnerabilities considering risk from shared components, common-cause failure of multi structures and components to calculate Multi Unit Core Damage Frequency (MUCDF) based on Multi Unit Initiating Events (MU-IE).

Email address:

a.qavi@pnra.org

Confirm that the work is original and has not been published anywhere else:

YES

67

Practical application of sensitivity and uncertainty analyzes in the analysis of radiological consequences for BDBA at modern NPPs with VVER

Authors: Leonid Petrov[1]; Nikita Shutov[1]; Yury Shmelkov[1]; Yury Zvonarev[2]

[1] *NRC Kurchatov Institute*

[2] *National Research Centre "Kurchatov Institute"*

Corresponding Author: shmelkov_yb@nrcki.ru

The analysis of the radiological consequences of accidents at NPPs is one of the most important part of the NPP safety justification. At the same time, due to the significant uncertainty of the initial data and models used in modeling the fission products behavior in beyond design basis accidents, uncertainty analysis plays an important role in interpreting the results of the accident radiological consequences analysis. Also worth noting that, in accordance with the regulatory documentation in force in Russia, the NPPs safety analysis must be accompanied by an uncertainty and sensitivity analysis.
This paper presents the results of uncertainty and sensitivity analysis for a hypothetical severe accident "Large Break with Station Blackout" for NPPs with VVER-1000 and VVER-1200. Uncertainty analysis was performed according to the methodology based on the ASME V&V 20 standard [1]. The number of variable parameters ranged from 10 to 20; the influence of the initial data uncertainty on the following results was analyzed:
1. fission products release from the fuel;
2. mass of fission products in the containment atmosphere;
3. release of fission products isotopes into the environment.
Calculations of the fission products release and transfer processes were performed using the MAVR-TA code, which has a module for uncertainty and sensitivity analysis [2, 3]. Totally 200 calculations were performed in the course of the work. The influence of the calculations number on the uncertainty results convergence was analyzed. Spearman and Kendall's rank correlation coefficients were used as criteria for the sensitivity of the models to the uncertainty of the initial data.

1. ASME, "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer - V&V 20", ASME, (2009).
2. Shmelkov Yu.B., Zvonarev Yu.A., Shutov N.V., Petrov L.V. Development and validation of the MAVR-TA code for analyzing the release and transport of fission products during a severe accident at a VVER NPP. Part 2 – Modelling of the fission products transport in the primary circuit and inside the containment // Nuclear Engineering and Design. 2021. V. 382, 111377.
3. Shmelkov Yu.B, Zvonarev Yu.A. Shutov N.V., Petrov L.V. Development and validation of the MAVR-TA code for analyzing the release and transport of fission products during a severe accident at a VVER NPP. Part 1 – Modelling of the release of fission products from the fuel // Nuclear Engineering and Design. 2021. V. 385.

Email address:

Shmelkov_YB@nrcki.ru

Confirm that the work is original and has not been published anywhere else:

YES

69

Using calibrated water data for preliminary validation of the SRT code for advanced reactors

Authors: Dong Hoon Kam[1]; Dave Grabaskas[1]; Tyler Starkus[1]; Matthew Bucknor[1]; Kyle Becker[2]; Mark Anderson[2]

[1] *Argonne National Laboratory*

[2] *University of Wisconsin-Madison*

Corresponding Author: dkam@anl.gov

Recent interest and corresponding progress worldwide regarding advanced nuclear reactors has renewed focus on their performance and related safety assessments. Specifically, the U.S. Nuclear Regulatory Commission has emphasized the importance of mechanistic approaches to source term

analysis for advanced reactor licensing applications, which attempt to realistically account radionuclide transport and retention phenomena. Further model development is required due to the numerous and complex physical and chemical phenomena associated with mechanistic source term analyses. Reflecting the need for modeling advancement, Argonne National Laboratory developed a mechanistic source term analysis tool for sodium fast reactors. The Simplified Radionuclide Transport (SRT) code describes fuel pin failure (for simulating the initial condition at the point of fuel pin breach), bubble scrubbing, deposition, leakage and following environmental impact. In the current work, a validation study of the SRT bubble scrubbing model is performed using a water-loop experiment performed at the University of Wisconsin-Madison. Through the analysis, the approach and fundamental bubble scrubbing models in SRT, which examine the removal of aerosols within the bubble as it is transported through a pool, have been widely evaluated. The results of the assessment demonstrate a high level of agreement in the regions of greater aerosol size. In the parameter range of minimal aerosol removal, the simulation slightly underpredicts the experiment results; however, considering the scale of plots and huge uncertainties inherently included, the deviation can be judged to be minor and would produce a conservative result. In addition, uncertainty analysis has been further refined to reflect the experimental distribution of parameters including aerosol sizes, which induces a span of performance for each representative aerosol size. Based upon the initial validation results along with uncertainty effects, SRT is expected to provide meaningful insights for the analysis of bubble scrubbing. Future sodium-loop tests will provide further validation basis.

Confirm that the work is original and has not been published anywhere else:

YES

Email address:

dgrabaskas@anl.gov

70

Considering International Safeguards During the Design of Advanced Reactors and Interfaces with Safety and Security

Authors: DONALD KOVACIC[1]; Guido Renda[2]

[1] *ORNL*

[2] *EC JRC*

Corresponding Author: kovacicdn@ornl.gov

Advanced reactors promise to bring new approaches to deployment of nuclear power to meet the increasing clean energy demands in the world. This includes some reactors that are smaller, more affordable, and compatible within smaller grids. Some that offer inherent safety features that greatly simplify licensing and siting. Some that use novel closed fuel cycles that burn waste or use thorium. Some that take advantage of non-fixed fuel in liquid or pebble form. Some that operate at high temperatures that can be used to provide process heat for industrial applications besides electricity. Some that can operate with little or no staff and can be placed in remote locations. Some that can be fabricated in factories and shipped to a location. And some that are movable and can be deployed at multiple locations during their lifecycles. And most advanced reactor designs, if not all, employ some combination of these, and other, attributes.

All of these designs must meet the appropriate levels of safety and security, as well as facilitate the efficient and effective application of international safeguards. This paper will review some of these classes of advanced reactors to suggest what safeguards considerations may exist and to highlight the interfaces with safety and security. It is a truism that the best time to consider design features is as early in the design process as possible and the further along the design process gets, the more difficult and costly it becomes to make changes. However, it is challenging for designers to understand how to consider safeguards early in the design process and how to develop and manage interfaces with safety and security. As such, there exists an opportunity to apply a holistic approach to this new generation of reactors to support their successful deployment that will not only address the major energy and climate change mitigation concerns, but do so safely, securely, and peacefully.

Email address:

kovacicdn@ornl.gov

Confirm that the work is original and has not been published anywhere else:

YES

71

The Consideration of the Independence of DID and its Application in HPR1000 Design

Authors: Yuxiang Wu[1]; chengcheng Wang[1]

[1] *China Nuclear Power Engineering Co Ltd*

Corresponding Author: wangcc0928@outlook.com

As stated in the IAEA SSR-2/1 (Rev.1) and IAEA TECDOC-1791, the design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable. When applying the independence requirement of defence in depth in the reactor design, some limitations of this requirement need to be considered, including physically impossible, practically unreasonable and complexification.
In this paper, the consideration of the independence between levels of defence in depth with regard to the IAEA latest requirements is discussed. Base on the understanding of the defence in depth concept, the application of the independence requirement is considered in the new design PWR in China. The analysis of defence in depth design features of the advanced nuclear power plant HPR1000 and new enhanced version of HPR1000 are also show the increased understanding and practices of the independence requirement of defence in depth in new reactor safety design.

Email address:

wangcc0928@outlook.com

Confirm that the work is original and has not been published anywhere else:

YES

72

Estimation of Reactor Pressure Vessel Breach Time for various Emergency Injection Modes in Case of Large Break LOCA

Author: Shahid Naveed[1]

[1] *Pakistan Nuclear Regulatory authority*

Corresponding Author: shahid.noveed@pnra.org

Pakistan Nuclear Regulatory Authority (PNRA) is developing Regulator's PSA Level-II Model to verify the compliance of regulatory acceptance criteria of Large Early Release Frequency (LERF), gain confidence in design and develop capability for review and assessment. This specific analysis is performed for estimation of Reactor Pressure Vessel (RPV) breach time for various emergency

injection modes i.e accumulator injection & High Pressure Safety Injection (HPSI) in case of double ended Large Break Loss of Coolant Accident (LOCA) for two loop Pressurized Water Reactor (PWR) Nuclear Power Plant (NPP). The objective of study is to determine the delay in RPV breach time in different cases viz. (a) Scenario-I: Accumulators injection & HPSI are unavailable, (b) Scenario-II: Accumulators' injection is available but HPSI is unavailable and (c) Scenario-III: Accumulators' injection & HPSI are available. In Scenario-III, HPSI is assumed available at onset of core damage and supposed to be terminated automatically at RWST depletion. The analysis results show that RPV breach is not delayed substantially in Scenario-I whereas it is delayed by marginal time (˜1 hour) in scenario-II. Nevertheless, a reasonable time delay in RPV breach was found in Scenario-III (˜5-6 hours) which provides a considerable time window for necessary operator action(s) to ensure In-Vessel Retention (IVR). The outcome of these analyses are used to optimize the function headers related to Emergency Core Cooling System (ECCS) in Level-I and Level-II PSA interface event trees (bridge event trees) to finalize Plant Damage States (PDS) for further analysis in containment event trees of Regulator's Level-II PSA Model.

Email address:

shahid.noveed@pnra.org

Confirm that the work is original and has not been published anywhere else:

YES

73

Advances in knowledge, modelling and methods to support safety demonstration of conventional and advanced nuclear fuels in the OECD/NEA Working Group on Fuel Safety

Authors: Didier Jacquemain[1]; Michelle Bales[1]; Zoltan Hozer[2]

[1] *OECD NEA*

[2] *Centre for Energy Research*

Corresponding Author: zoltan.hozer@energia.mta.hu

The OECD Nuclear Energy Agency (NEA) Working Group on Fuel Safety (WGFS) was created two decades ago and is tasked with advancing the current understanding of nuclear fuel safety issues by assessing the technical basis for current fuel safety criteria and their applicability to high burn-up and to advanced fuel designs and cladding materials. The group aims at facilitating international convergence in this area, including as regards experimental approaches and interpretation and the use of experimental data in fuel safety evaluation. The working group is closely following the activities of OECD fuel projects (e.g. Halden Reactor Project, CABRI, SCIP, FIDES).
In recent years, WGFS's achievements have been outstanding in developing state of the art technical reports on Loss of Coolant accidents (LOCA) and Reactivity Initiated Accidents (RIA), which are reference publications, in organizing a workshop on advanced modelling of fuel behaviour to support fuel safety and performance, co-ordinating RIA benchmark calculations and providing a technical opinion paper on applicability of nuclear fuel safety criteria to Accident-Tolerant Fuel (ATF) designs. Main lessons and recommendations from these activities are provided in the paper as well as perspectives including for instance works on applicability of nuclear fuel safety criteria to high burn-up fuels, on guidance for Design Extension Conditions (DEC-A) analyses and on defining features of a data management tool for preserving key RIA data sets for fuel safety applications.

Acknowledgements
The significant contributions of those individuals who had a key role in conduct and success of the activities of the Working Group on Fuel Safety, such as task leaders, contributors and members, are kindly acknowledged.

Email address:

hozer.zoltan@ek-cer.hu

Confirm that the work is original and has not been published anywhere else:

YES

75

Experimental Investigation of Liquid Tin Surface Property Safety Features for Potential Application as a Coolant in Direct Contact Liquid Metal Fast Reactors (DCLMFR) Heat Removal

Authors: Plamen Petkov[1]; Dimitar Lyutov[2]; Boyan Todorov[2]; Hristo Iliev[1]

[1] *Sofia University "St. Kliment Ohridski"*

[2] *Sofia University "St. Kliment Ohridski'*

Corresponding Author: pvpetkov@phys.uni-sofia.bg

Liquid tin (Sn) has several advantages as a coolant in Liquid Metal Fast Reactors (LMFR). The investigations for its potential applications were initiated during 1950s [R. E. Lyon, "LIQUID-METALS HANDBOOK," Atomic Energy Commission, Washington D. C., 1952]. One of the earliest proposals for liquid tin application as a coolant, was published by Weeks [Weeks, "Lead , Bismuth , Tin And Their Alloys as Nuclear Coolants," Nucl. Eng. Des., vol. 15, pp. 363-372, 1971], where he considered not only pure tin, but also some eutectic mixtures. Later, Khorasanov et. al. [G. L. Khorasanov, A. P. Ivanov and A. L. Shimkevich, Possibilities of using the liquid tin as a coolant in electronuclear plants, Russian Federation, 1999] mentioned several advantages of tin application in nuclear industry as a coolant. However, it aggressively interacts with the structural materials which complicates the usage.

The concept of Direct Contact Liquid Metal Fast Reactors (DCLMFR) was originally proposed in [J. Buongiorno, N. E. Todreas and M. S. Kazimi., "Thermal Design of a Lead-Bismuth Cooled Fast Reactor with In-Vessel Direct-Contact Steam Generation," in ICONE-9, Nice, April 8-12, 2001]. It is advantageous, because the design does not consider specific heat exchanging equipment: Liquid water is injected directly into the Pb-Bi coolant in reactor core and the generated steam is directed further to the turbine, assuring this way the natural circulation of the coolant. The concept that we propose is very similar: We intend to use liquid tin as a primary core coolant and for secondary direct heat removal coolant we consider the usage of non-condensing gases (or water/steam). However, we plan the secondary coolant injection to take place separately from the reactor core. It appears again to be the driven force for the primary coolant circulation. The liquid tin is suitable to become a coolant of this nuclear reactor design because non-condensable gases nitrogen, argon, helium carbon monoxide and carbon dioxide are practically insoluble.

The current paper represents our pre-concept and pre-design studies that involve gas/tin surface interactions. We designed a simple device, based on Wilhelmy plate surface tension measuring approach, and studied the kinetics of formation of stannic and hydrated oxides while liquid tin was exposed to atmospheric air as well as liquid tin surface properties. We considered temperatures from the tin melting point up to below 600. We pay attention also to liquid tin dissolution properties regarding construction materials, some of which could be possibly used in the future design. The oxidation kinetics measuring plates were considered made from: copper, nickel, brass and various steels. In addition, we collected the accumulated at the liquid surface oxides and measured their composition to identify reliably the metal dissolution rates of the measuring plate. The results of these studies directly affect the selection of materials that could be possibly used in the future design. We also extended our study not only to pure liquid tin but also to its industrial grade alloys implemented as solders in electronic industry: Sn100, Sn60 and Sn40.

Email address:

pvpetkov@phys.uni-sofia.bg

Confirm that the work is original and has not been published anywhere else:

YES

76

NUWARD Safety Approach, Implementing SMR Specifics and Preparing International Deployment

Authors: Debbie Francis[1]; Emmanuel Vieilletoile[None]; Stephane Beils[None]

[1] *EDF*

Corresponding Author: debbie.francis@edf.fr

Drawing on the deep experience and understanding of the principles of nuclear safety as well as many years of nuclear power plant design and operation, the EDF led NUWARDTM Project is at the point of delivery of the concept design for a GEN III+ Small Modular Reactor (SMR) of 340 MWe, that will supplement the offering of high-output nuclear reactors, especially in response to specific needs such as replacement of fossil-fuelled power plants. NUWARDTM is a nuclear power plant composed of two integrated Pressurised Water Reactors with a mix of proven and innovative design features that will make it more commercially competitive, while integrating safety features that comply with the highest international standards. Following the principles of redundancy and diversity and rigorous application of Defence in Depth (DID), with an international view on nuclear safety licensing, the Project also incorporates new safety approaches into its design development.
This paper will:
• Summarise the foundation principles and technological background which underpin the design;
• Contextualise the key design features with regard to the international safety regulatory framework with particular emphasis on innovative passive safety aspects;
• Illustrate the Project activities in preparation for licensing, including forward looking technical exchanges with the ASN and the IRSN in France, and also a wider international view of design licensing viability;
• Articulate the collaborative approach to design development from involvement with the Project partners (CEA, Naval Group, TechnicAtome and Framatome) to the establishment of the International NUWARD Advisory Board (INAB), to gain greater international insight and advice;
• Conclude with the focus on next steps into detailed design development, standardisation of the design and its simplification to enhance its commercial competitiveness in a context of further harmonisation of the nuclear safety and licensing requirements and aspirations.
Keywords: SMR, passive safety, innovation in design

Email address:

debbie.francis@edf.fr

Confirm that the work is original and has not been published anywhere else:

YES

77

On the approaches to licensing of activities during realization of innovative projects in the field of atomic energy use including legal and regulatory framework improvement

Author: Andrey KIRKIN[1]

[1] *Scientific and Engineering Centre on Nuclear and Radiation Safety*

Corresponding Author: kirkin@secnrs.ru

In the Russian Federation the development and realization of innovative projects in the field of atomic energy use is envisaged within the framework of several governmental programs. In particular, it is planned to realize nuclear installations with small modular reactors for energy production, molten-salt reactors for minor actinides burnup and high-temperature gas-cooled reactors for hydrogen production. Creation of innovative fusion installations is also planned in the future.
In order to enable an effective safety regulation and licensing of activities during realization of such projects there is a large amount of work carried out in the Russian Federation focused on the improvement of legal and regulatory framework. The regulatory requirements improvement is necessary for accounting features of new technologies and main design solutions.
This report is dedicated to the activities of SEC NRS on interaction with developers of innovative technologies, including the analysis of design prior to the licensing procedure. Conducting such an analysis helps identify safety deficiencies in design and allows developers to make necessary adjustments to the design or develop necessary safety cases at an early stage of the project realization. In addition, the information obtained can be used to develop in advance proposals for the regulatory body on improving regulatory documents, including procedure for licensing of activities in relation to future innovative installations.
Described activity of SEC NRS helps to provide technical and scientific support to Rostechnadzor on the regulatory framework improvement and development of approaches to licensing of activities during the realization of innovative projects in the field of atomic energy use.

Email address:

kirkin@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

YES

78

External Hazard challenges for Evolutionary and Innovative Reactor Designs

Author: Karel Deknopper[None]

Corresponding Author: karel.deknopper@tractebel.engie.com

External Hazard (EH) are generally not considered as novel for Evolutionary and Innovative Reactor Designs (EIRD). However they are important in several areas which are considered novel for EIRD :
- Standardized design : characterization of EH for the standard plant parameter envelope, aiming at sufficient robustness to be deployable internationally, but still within the envelope of being reasonably achievable;
- PIE Identification : PIE (postulated initiating events) can originate from EH, but there is a lack of experience on PIE identification for EIRD;
- Passive safety features : specific attention for conditions resulting from EH to confirm successful operation of passive systems;
- Multi-reactor-modules SMRs : demonstration that reactor modules closer proximity and sharing of safety features is not challenged by EH;
- Emergency Planning Zones : reduced EPZ allow to show reduced risk from EIRD to the public. EH are important as they could be at the origin of the scenarios to be assessed for EPZ determination.

EH represent some challenges :
- DBEH characterization (generally at a frequency of 10-4/yr) being feasible, this is difficult for BDBEH (e.g. at a frequency of 10-6/yr), challenging integrated safety/risk decision making in general, and assessment of DEC and demonstration of practical elimination in particular;

- Even if EH safety objectives are harmonized at international level,
their application and interpretation at national level could
necessitate significant design changes.

The paper will elaborate on the importance and challenges of EH for EIRD. Examples of country-specific approaches will be discussed. Most important needs for harmonization will be identified.

Email address:

karel.deknopper@tractebel.engie.com

Confirm that the work is original and has not been published anywhere else:

YES

80

DERIVATION OF RADIONUCLIDE DISCHARGE LIMITS FOR BELARUSIAN NPP WITH WWR-1200 REACTOR FOR RADIATION PROTECTION OF THE PUBLIC

Author: Viktoryia Kliaus[1]

[1] *Republican Scientific Practical Centre of Hygiene*

Corresponding Author: vkliaus@gmail.com

According to the Decree of the President of the Republic of Belarus dated November 2, 2013 No. 499 «On the construction of the Belarusian nuclear power plant», the NPP consisting of two power units with WWER type reactors with a total capacity of up to 2400 (2×1200) MW at the Ostrovets site was constructed. The first power unit was put into commercial operation on the 10th June 2021, the physical launch of the second unit was in December 2021.
In order to get a license for the operation in accordance with the IAEA requirements and recommendations and national regulation authorized limits for radioactive discharges must be set. The WWR-1200 reactor is an innovative project and by the 2019-2020 there were no enough data about the type and amount of airborne releases and liquid discharges of radioactive substances to the environment from this reactor type. That is why the data from the WWR-1200 project documentation were taken in order to perform an assessment of impact to public and the environment from airborne releases and liquid discharges of radioactive substances in case of normal operation of the Belarusian NPP.
The guides of Russian Federation were used for the derivation of discharge limit values for radioactive substances. These guides use the procedure for the calculation of activity concentrations and dose assessment outlined the IAEA SRS-19 Publication together with the ranking procedure (radionuclides that contribute up to 99 % to dose are considered) for defining a list of radionuclides that must be controlled.
The processes involved in setting discharge limits and authorization conditions included the following steps: characterization of the source term and defining exposure pathways, dose assessment and comparison of the resulting public dose values with the established dose constraint (100 μSv/y in Belarus).
For public dose assessment the «representative person» concept was used in combination with site-specific radioecological parameters and habit data.
According to the WWR-1200 project characterization the list of radionuclides discharged during normal operation of Belarusian NPP includes: 3H, 131I–135I, 89Sr, 90Sr, 95Zr, 95Nb, 134Cs, 137Cs, 141Ce, 51Cr, 54Mn, 58Co, 59Fe, 60Co with total amount of 1,10×1013 Bq/y from one NPP unit. The derived list of radionuclides in the liquid discharges into Viliya river for which a discharge limit values were set includes: 137Cs, 134Cs, 3H, 60Co, 131I. These radionuclides will form the annual public dose of about 12 μSv/y.
The list of radionuclides released from the main stack with total amount of 4,30×1013 Bq/y and from the turbine building – 5,20×1011 Bq/y includes 3H, 14C, 83mKr-88Kr, 131mXe-138Xe, 131I-135I, 51Cr, 54Mn, 60Co, 89Sr, 90Sr, 134Cs, 137Cs. The airborne discharge limit values were set for: 14C, 3H, 88Kr,

137Cs, 134Cs, 131I,87Kr, 133Xe, 133I (main stack) and 88Kr, 137Cs, 87Kr, 134Cs, 131I, 138Xe, 135Xe, 133I, 133Xe, 3H, 85mKr (turbine building). Additionally, 60Co and 41Ar are included into the list in accordance with national regulatory requirements. The resulting dose to the representative person will be one order of magnitude less than 10 μSv/y.

Email address:

vkliaus@gmail.com

Confirm that the work is original and has not been published anywhere else:

YES

81

Safety, Security and Safeguards working together in the modernised Generic Design Assessment

Authors: Duncan Barley[1]; Sarah Halhead[2]

[1] *CSyP MISMI FCMI FSyI CSMP*

[2] *NA*

Corresponding Author: duncan.barley@onr.gov.uk

The UK Office for Nuclear Regulation (ONR), based on learning from experience, has started to examine the benefits and challenges of taking a more holistic approach when assessing proposed nuclear facilities. Consistent with emerging IAEA expectations and initiatives, we are already engaged in related strands of work that seek greater interaction between security, safety and safeguards functions (commonly called the 'three Ss'). Furthermore, in taking this holistic approach and encouraging innovation through our 'goal setting' regulatory regime, we seek a better appreciation of the potential cross-specialism opportunities and risks offered by more novel nuclear power plant designs.
The UK's ONR and the Environment Agency are targeting completion of the Generic Design Assessment (GDA) of the UK HPR1000 reactor design early in 2022. For the security assessment of that design, we have applied our new regulatory approach based on 'Security Assessment Principles' so to better align security with safety. Although this initiative is only the beginning of seeking closer integration between the 'three Ss', it sets the conditions for a more integrated approach to assessing risks inherent in a design and how these might be addressed adequately by a vendor or 'Requesting Party'. Moreover, our expectation is that, within any GDA, the security case draws from and informs a safety case. Likewise, the expectation is that a Requesting Party should adopt a 'secure by design' approach that itself demands close cooperation between safety, security and safeguards specialisms. Therefore, building on our successful experience of safety and security working together, our future GDAs will seek to integrate safeguards as an assessment discipline.
As part of this paper, we will present our developing ideas on how ONR will now incorporate 'safeguards by design' into our learning on safety and security collaboration. We will also offer our thinking on where interaction between the 'three Ss' would be mutually beneficial while acknowledging any limitations on this approach. Such limitations suggest our approach would need to be evolutionary and carefully considered. In this way we will be able to confidently guide and assess future GDA Requesting Parties with the expectation that their final designs reflect the potential benefits that could be offered through the further integration of the 'three Ss'.

Email address:

Sarah.Halhead@onr.gov.uk

Confirm that the work is original and has not been published anywhere else:

Yes

82

Early Assessment of Innovative and Advanced Modular Reactor (AMR) Designs - Regulatory Process and Insights from Application to Eight Designs in the UK

Author: Diego Fernández Lisbona[1]

Co-author: Ana Gomez Cobo [2]

[1] *CEng MIChemE*

[2] *NA*

Corresponding Author: diego.lisbona@onr.gov.uk

In 2019, as part of the Advanced Modular Reactors (AMR) Feasibility and Development (F&D) project sponsored by the UK Department for Business Energy and Industrial Strategy (BEIS), the UK Office for Nuclear Regulation and the Environment Agency developed and applied a process for early regulatory assessment of innovative (generation IV) reactor designs. To deliver a measure of the potential for regulatory alignment with UK regulatory expectations to BEIS, the process was applied to 7 innovative fission reactors - three high temperature gas reactors, one sodium-cooled fast reactor, two lead-cooled fast reactors, a molten salt reactor - and a fusion reactor. The process comprised assessment on 18 technical discipline areas across nuclear safety, site safety, transport, security, safeguards and environmental protection.

As part of this paper, we will present the process and guidance developed to enable a targeted regulatory submission from vendors, and the stages, timeline and assessment strategies applied through the process. We will also share the regulatory themes, evaluation points and indicators of regulatory confidence developed to guide proportionate regulatory judgements, and the key findings and lessons learnt from applying the process to the above innovative designs.

Email address:

Diego.Lisbona@onr.gov.uk

Confirm that the work is original and has not been published anywhere else:

Yes

83

Regulatory perspective on the application of the defence-in-depth concept to innovative reactor technologies

Author: David Plummer[1]

[1] *Institute of Physics*

Corresponding Author: david.plummer@onr.gov.uk

In 2021, the Advanced Nuclear Technologies (ANT) team at the UK Office for Nuclear Regulation (ONR) undertook a review of the potential challenges to the well-established nuclear safety principle of defence in depth (DiD) posed by some of the safety arguments commonly being proposed for 'innovative' reactor technologies. This review was informed by experience gained through literature reviews, training courses and regulatory review of vendor submissions to the Advanced Modular Reactors (AMR) Feasibility and Development (F&D) project sponsored by the UK Department for Business Energy and Industrial Strategy (BEIS). Our review was also informed by engagement at international working groups, ONR's regulatory guidance, and the output of dedicated workshops with members of ONR's ANT project team. The review focussed on four reactor technologies i.e. high temperature gas-cooled reactors, sodium-cooled fast reactors, lead-cooled fast reactors and molten salt reactors. The workshops included participants from eight technical disciplines and the

output was captured in an internal technical guidance note, written to assist ONR's inspectors in reaching consistent regulatory judgements in future assessments.
As part of this paper, we will present a selection of safety arguments being proposed for each reactor technology that could challenge common approaches to implementation of DiD, and summarise the output of the internal workshops held. We will also share examples of assessment considerations and regulatory expectations of particular relevance to judgements on the adequacy of DiD provisions for an innovative reactor design, also developed as part of the workshops. Finally, we will present the key conclusions and recommendations from the review.

Email address:

David.Plummer@onr.gov.uk

Confirm that the work is original and has not been published anywhere else:

Yes

84

Technology-Inclusive Human-System Considerations for Advanced Reactors

Author: Jesse Seymour[1]

[1] *U.S. Nuclear Regulatory Commission*

Corresponding Author: jesse.seymour@nrc.gov

It is anticipated that developers of advanced reactor technologies will propose novel approaches to reactor operations. Existing U.S Nuclear Regulatory Commission (NRC) requirements for human factors engineering (HFE), staffing, and personnel qualifications were developed for large light water reactors and may not be appropriate for other nuclear technologies. Therefore, the NRC is developing a new risk-informed, performance-based, and technology-inclusive regulatory framework for licensing new technologies.

The new framework takes an integrated approach to human-system interactions and achieves being technology-inclusive by linking requirements for HFE, staffing, and operator qualifications to design-specific plant safety functions and their fulfillment. In developing this framework, the NRC strategy was influenced by a mandate to establish performance-based requirements while also satisfying other statutory requirements. Within this framework, HFE is required for settings where HFE is needed to support plant safety, versus limiting its application to the main control room. Operator staffing requirements are based on analyses that demonstrate minimum staffing is adequate to support safety function fulfilment, versus a prescribed number of operators. Finally, the fundamental role of reactor operators and senior operators is centered around the management and fulfilment of safety functions, with personnel qualification requirements being performance-based and applying an examination process that is tailored to the facility.

This paper will discuss the method the NRC is using to develop technology-inclusive, performance-based, and risk-informed proposed requirements for HFE, staffing, and plant personnel qualifications.

Email address:

jesse.seymour@nrc.gov

Confirm that the work is original and has not been published anywhere else:

YES

85

Application of the Level-1 PSA Model in Various Regulatory Processes and Challenges Faced during their Execution

Authors: Ghulam Farid[1]; Khalil Ur Rahman[1]; Sayed Muhammad Waseem Bukhari[1]; Muhammad Majid Khan[1]

[1] *Pakistan Nuclear Regulatory Authority*

Corresponding Author: waseem.bukhari@pnra.org

The Technical Support Organization (TSO) of Pakistan Nuclear Regulatory Authority (PNRA), has developed independent regulatory Level-1 PSA Models for 300MWe and 1100MWe PWR Nuclear Power Plants (NPP) to support the regulatory oversight process. PSA model development was carried out by the regulatory body as a tool for ensuring the safety of the nuclear power plants in relation to potential initiating events caused by random component failure and human errors. The PSA model for 300MWe NPP was developed under IAEA TC Project where IAEA experts provided support through periodic reviews and the PSA model for 1100MWe NPP was developed independently by PNRA. The PSA models provided many useful insights into plants' systems and components including independent assessment of the probabilities of occurrence of core damage states resulting from various initiating events and adequacy of plant emergency procedures. The Level-1 PSA models are being utilized at PNRA for various regulatory applications e.g. review and assessment of SARs (PSAR, FSAR), review of design modifications and technical specifications modifications, risk-informed inspection plans, PSA-based analysis of operational events, etc. In this paper, the applications of the regulator's Level-1 PSA models have been discussed. The challenges faced during the development of regulatory PSA models and implementation of PSA applications are described and potential future applications at PNRA are also highlighted.

Email address:

waseem.bukhari@pnra.org

Confirm that the work is original and has not been published anywhere else:

YES

86

The Finite Element Neutronics Code FENNECS for the Safety Assessment of SMRs, Micro Reactors and other Innovative Concepts

Authors: Armin Seubert[1]; Jérémy Bousquet[1]; Romain Henry[1]

[1] *Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH*

Corresponding Author: armin.seubert@grs.de

Worldwide, there is increasing interest in Small Modular Reactors (SMRs) and Micro Modular Reactors (MMRs). Their specific purposes range from energy delivery at remote locations to space applications. MMRs typically have electrical outputs of not more than 10 MW and are typically not Light Water Reactor (LWR)-based technologies. Their geometries are characterized by, e.g., rotating control drums and heat pipes, and so are deviating from regular assembly lattice arrangements as known from, e.g., LWRs. Established neutron kinetics codes for, e.g., LWRs however assume regular core lattices and cannot be applied to MMRs. Though Monte Carlo methods with their geometric flexibility are becoming more and more standard for steady state simulations, they are not yet mature enough for transient applications and not practicable due to their huge computational demands. To perform future coupled transient safety assessments of SMRs, MMRs and other innovative concepts, the Finite ElemeNt NEutroniCS (FENNECS) code is being developed at GRS. FENNECS is accompanied by the spatial meshing tool named PEMTY (Python External Meshing Tool with Yaml input) which is also developed at GRS and has capabilities to provide the meshing of, e.g., control drums, circular core boundaries etc. which are characteristic for MMRs, in addition to embedded hexagonal and Cartesian pin cell lattices which allow, e.g., pin-cell resolved simulations of hexagonal assembly arrangements of Generation IV and other innovative reactors. This paper describes the major features of FENNECS, which is also coupled with the thermal hydraulics code ATHLET, and PEMTY. In the second part, applications to the Heat Pipe Micro Reactor (HPMR) and the China Experimental Fast Reactor (CEFR) as well as an approach to pin-by-pin subchannel-like coupled simulations of a CEFR minicore are shown.

Email address:

armin.seubert@grs.de

Confirm that the work is original and has not been published anywhere else:

YES

87

FRENCH DIGITAL NUCLEAR REACTOR INITIATIVE: A COMMON PLATFORM FOR ADVANCED CODE COUPLING FOR THE FRENCH NUCLEAR INDUSTRY.

Authors: Anthime FARDA[1]; Chaï KOREN[2]; Clément LAFAURIE[3]; Cyril PATRICOT[1]; Eric FAYOLLE[2]; Pascale NOYRET[2]; Roland LENAIN[1]; Yohann PIPEAU[2]

[1] *Université Paris-Saclay, CEA, Service d'Études des Réacteurs et de Mathématiques Appliquées*

[2] *EDF R&D*

[3] *FRAMATOME*

Corresponding Author: anthime.farda@cea.fr

Code coupling has a long history in reactor physics. However, coupling calculations are often seen has individual exercises serving the purpose to perform a specific calculation with already defined codes in each physic. With the Digital Reactor project, the French nuclear industry aims at developing a common coupling platform for every advanced code currently in use. The interchangeability is obtained by a clear definition of an integration process allowing the coupling script to be written on an abstract object representing the code itself. Following this procedure, new calculation codes can be added in the platform as long as they comply with a limited number of prerequisites, the most important of which being the ability to be driven in Python through an ICoCo like interface [ref]. This paper presents the current status of the project with its key components and already available codes. An interchangeability demonstration is proposed through a Rod Ejection Accident (REA) calculation of typical PWR.

Email address:

anthime.farda@cea.fr

Confirm that the work is original and has not been published anywhere else:

YES

88

U.S. and Canada Cooperation on Advanced Reactor Technologies - Progress and Challenges

Author: Donna Williams[1]

[1] *US Nuclear Regulatory Commission*

Corresponding Author: donna.williams@nrc.gov

The U.S. Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission have embarked on a first of a kind effort to collaboratively perform safety reviews of advanced reactor and small modular reactor designs that are expected to be constructed in both countries. The collaboration was formalized in a Memorandum of Cooperation (MOC) in 2019. The goal in forming this MOC was to gain efficiencies by making joint observations on advanced technologies or identifying where differing regulations may result in different regulatory decisions. Under the terms of this MOC, CNSC and U.S. NRC may take each others' experiences, regulatory information and results into account when conducting technology assessments for the purposes of feedback to vendors or making recommendations for regulatory decisions.

The scope of the collaboration includes comparison of licensing approaches and the development of shared approaches to conducting technical reviews, as well as collaboration on pre-licensing activities and licensing reviews. The projects chosen for joint licensing reviews generally involve specific aspects of a design with unique or novel technical considerations. Current active projects include joint reviews of white papers and topical reports submitted to the NRC or CNSC as part of pre-application engagement by advanced reactor vendors, and comparisons of the strategies for licensing advanced reactors in the U.S. and Canada.

This effort has produced initial results and products and has provided insights into the challenges of performing joint reviews by multiple regulators. In August 2021, the NRC and CNSC completed the first joint products under the MOC. These reports discussed NRC and CNSC positions on a white paper submitted by X Energy, LLC, on choosing a construction code for the reactor pressure vessel and provided a comparison of the U.S. Licensing Modernization Project with Canada's practice for review of advanced reactors.

In response to lessons learned from early implementation of the MOC, the NRC and CNSC have established processes and protocols to ensure that the collaborative projects benefit both agencies and do not result in longer review times and increased resources expended. The NRC and CNSC have identified criteria to strategically select licensing projects that are at similar phases of submittal in each agency and have actively engaged with vendors to ensure that the requests to both regulators are similar enough to allow for a joint review. We anticipated that differences in licensing frameworks and processes would be a challenge to developing joint regulatory positions. To address this, the working groups held training sessions and compared the regulatory frameworks to enable an understanding of the regulatory decisions issued and how they may be applied to other frameworks. Other challenges encountered include differences in the priority of licensing projects and resources available by each regulator, as well as logistical challenges in sharing sensitive information and efficiently working together to develop products. These have been addressed through the use of IT tools, staff exchanges, and alignment on the processes and protocols to be used during the reviews.

Email address:

donna.williams@nrc.gov

Confirm that the work is original and has not been published anywhere else:

YES

89

A review of the requirements of the licensing procedure for the HTR-PM

Author: Rozbeh Vadi[1]

[1] *Iran Nuclear Regulatory Authority*

Corresponding Author: roozbehvadi@yahoo.com

To ensure sustainability of green energy production and to face the challenges like shortage of fresh water recourses and global warming, selecting and acquiring a suitable generation IV SMR technology have been pursued by many countries. The fact that the HTR-PM demonstration plant as the first of its kind has recently become fully operational in the Shidao Bay China has boosted the interest toward this type of SMRs. Hence, a research performed, from a regulator point of view, on various aspects of the associated regulatory activities and subject of this article is to present a concise review of the results. Accordingly, in this article, first, a discussion on the procedure for HTR-PM licensing is presented. Then, some crucial challenges and concerns recognized in the process are delineated. As prime examples, very hot and humid environment would make the current design of the RCCS (Reactor Cavity Cooling System) impractical and high frequency and intensity of earthquake occurrence could increase the probability of the critical components failure (such as the coaxial hot-gas duct break which results in the air-ingress as a sever beyond design-basis accident) to a level that necessitates establishing the "emergency response plan", in contrast to the original design. The article also includes the proposed solutions to the mentioned concerns. For example, using the modern type of heat-pipe aka TPCT (Two-phase closed Thermosyphon) for cooling the reactor cavity and mandating to present detailed seismic fragility curves for all critical components to the safety.

Email address:

roozbehvadi@yahoo.com

Confirm that the work is original and has not been published anywhere else:

YES

90

3S interface considerations for novel advanced reactors

Authors: Minna Tuomainen[None]; Paula Karhu[1]; Tapani Honkamaa[2]

[1] *Radiation and Nuclear Safety Authority (STUK)*

[2] *STUK*

Corresponding Author: paula.karhu@stuk.fi

A security-safeguards-safety (3S) interface is any decision point where nuclear safeguards, safety and security should be considered. Interface management is a systematic way to recognise the decision points, to take advantage of the synergies and to resolve the conflicts, to achieve the joint fundamental objective to protect people and the environment from harmful effects of ionizing radiation

These principles are the same for all nuclear reactors. Some characteristics of novel advanced reactors (NAR) may require extra consideration.

New types of owners/users are potentially envisaged for NAR: cities, communities, non-nuclear industry operators… To achieve the expected benefits of NAR while ensuring the same high standard of 3S as for traditional reactors we may wish to consider where the profound expertise for 3S and nuclear engineering and operations should reside, how much expertise the owners should be required to have, what are they allowed to outsource, who has the competence to purchase, commission, operate and decommission a (fleet of) NARs, and to oversee 3S in the supply chain.

Leadership, management, and organizational culture as overarching concepts to ensure that nuclear safety, security, and safeguards have due priority are already rooted deep. It has been a process. The

same must be applied to any NAR, notwithstanding the characteristics and numbers of new owners, operators, designers, and suppliers.

Information security is in the crux of 3S. For effective operational monitoring as well as response to anomalies availability and integrity of information on facility status and secure and efficient information transmission are crucial, in balance with confidentiality. The interdependence between information security, physical security, safety, and safeguards is strong due to digital, programmable systems and general increased connectivity. This applies to any modern facilities, including NAR and their potential remote monitoring, maintenance, and/or operations.

While small (fuel elements, cores, modules) may be beautiful from the point of view of practicability or safety, for security it may present a new kind of "theft target". New designs need to be assessed for potential consequences to specify 3S requirements in accordance with the risk-informed, graded approach. Could technological solutions help reduce safeguards inspection effort? Could systems for monitoring, surveillance and detection be implemented in a more synergistic way? It would seem attractive to attempt to apply 3S by design to find the effective, balanced, risk-informed solutions.

Response is a test for interface management: demonstration of effectiveness of the design in delay, mitigation and minimization of harmful consequences, coordination of accident response, security response and NMAC measures, on-site and off-site response, and all entities involved in off-site response while considering potential exotic siting—urban, remote, marine, mobile.

Has the window of opportunity for 3S by design been and can it still be taken advantage of for NAR?

Email address:

paula.karhu@stuk.fi

Confirm that the work is original and has not been published anywhere else:

YES

91

Long Term Effects and Numerical Simulation of Radiolytic Gas, Non-Condensable Gas and Boron Transport For Small Modular Light Water Reactors

Author: Shanlai Lu[1]

[1] *U.S Nuclear Regulatory Commission*

Corresponding Author: lushanlai@hotmail.com

In the past decade, nuclear industries and governments world wide have developed interests to design and deploy small modular reactors as viable energy source options to reduce carbon dioxide emissions and help resolving climate change issues. According to International Atomic Energy Agency (IAEA), there are approximately 70 small modular reactor (SMR) designs under investigation in 17 countries. These new SMR designs are at different stages of research, development, licensing and commercialization. The collaborations coordinated by IAEA and driven by major industrial countries have put several SMR designs as the front runners in the phase of technology commercialization. These collaborations envisage 2030 as the target in-operation date. The construction of small modular nuclear reactors by 2030 will be a solid declaration and an element of world wide energy transformation.

Some SMR designs under development have evolved from large light water reactor (LWR) designs and use light water as a coolant and neutron moderator with passive gravity driven systems for normal operation and accident mitigation. The design goals of these passive safety systems are to maintain reactor core cooling for at least 72 hours without any power supply or operator intervention for a broad range of hypothetical accident scenarios including loss-of-coolant accidents (LOCA) and station blackout. These SMR designs normally have significantly higher coolant inventory/reactor power ratio than that of conventional large LWRs and are expected to keep the reactor core covered under two-phase water level, or, at a minimum, preclude the fuel from experiencing prolonged heat-up for at least 72 hours. The reliance on gravity driven buoyancy flow for their long term Emergency Core Cooling System (ECCS) operation makes it possible to eliminate the need for higher cost active pumping systems, simplify the containment designs, and, reduce the initial capital investment. The gravity driven ECCS designs can be reliable because of their reliance on inherent features and natural phenomena. However, the gravity driven buoyancy flow and natural circulation change the system characteristics during the long term cooling period after the initial transients. Depending on the design, the systems may become sensitive to some physical phenomena, such as (1) radiolytic gas generation and migration, (2) non-condensable gas effects and (3) boric acid transport if used in the primary circuit to control the reactivity. These phenomena were not considered to be significant concerns for most current LWR designs utilizing active ECCS, so that they were not explicitly modeled in detail to the degree necessary for a design that relies on passive ECCS features as part of the design basis analyses. Proper evaluation of these accumulated effects may become necessary for passive LWR SMRs designs.

In this paper, the authors reviewed the information on these three phenomena and identified their potential safety implications for passive LWR SMRs. The state of art computer simulation tools commonly used by both the industry and regulatory agencies are summarized for their applications to evaluate the accumulated effects of these phenomena, including the challenges, limitations of these computer codes and the future development needs.

Email address:

lushanlai@hotmail.com

Confirm that the work is original and has not been published anywhere else:

YES

92

BWR Severe Accident Uncertainty and Sensitivity Analysis in the Framework of the IAEA CRP I31033

Author: J. Ortiz[1]

Co-authors: H. Sanchez [1]; D. Luxat [2]; L. I. Albright [2]; R. Bocanegra [3]; L. Herranz [3]; E. K. Boafo [4]; C. Múgica [5]; R. López [5]; T. Jevremovic [6]; H. Rehman [6]; F. Mascari [7]; A. Bersano [7]; F. Gabrielli [8]; J. M. García [9]; K. I. Ahn [10]; N.I. Ryzhov [11]; S. M. Petoukhov [12]

[1] *ININ, Mexico*

[2] *SNL, USA*

[3] *CIEMAT, Spain*

[4] *GAEC, Ghana*

[5] *CNSNS, Mexico*

[6] *IAEA, Austria*

[7] *ENEA, Italy*

[8] *KIT, Germany*

[9] *CNEA, Argentina*

[10] *KAERI, Republic of Korea*

[11] *IBRAE RAN, Russian Federation*

[12] *CNL, Canada*

Corresponding Authors: heribertosanchez7@hotmail.com, javier.ortiz@inin.gob.mx

The International Atomic Energy Agency (IAEA) launched in 2019 the Coordinated Research Project (CRP) I31033 entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors" [1], with the main goal to advance the understanding and characterization of sources of uncertainty and investigate their effects on the key figure-of-merits (FOMs) of the severe accident (SA) codes predictions in water-cooled reactors (WCRs). Having this in mind, within the 5 years duration of the CRP, 22 organizations in 18 Member States have been developing and assessing calculation platforms based on SA codes and relevant uncertainty tools (UT) for uncertainty and sensitivity (U&S) analyses of SA scenarios in different WCRs, i.e. PWR, BWR, CANDU, VVER, and SMR. In the CRP framework, the U&S analysis of the QUENCH-06 experiment [2] has also been performed to give relevant insights.
The BWR group in this CRP is formed by the following five institutions: 1) SNL (USA), 2) CIEMAT (Spain), 3) GAEC (Ghana), 4) CNSNS (Mexico), and 5) ININ (Mexico). The first three institutions address severe accident scenarios in a BWR/3 with Mark I primary containment, while the last two use models of a BWR/5 with Mark II containment. Except for CNSNS, the participants have chosen a short-term (unmitigated) Station Blackout scenario for analysis, and the scope of the analysis focused on in-vessel phenomena. In the case of CNSNS, depressurization and RCIC injection actions are considered as mitigating measures and the analysis considers both in- and ex-vessel phenomena. The MELCOR code is employed for the severe accident simulations by the first four participants and the MAAP5 code is used by ININ. DAKOTA has been chosen as a tool for the sensitivity and uncertainty calculations by CIEMAT, GAEC, and CNSNS, while in-house developed tools are used by SNL and ININ. Currently, all these institutions have determined the figures of merit and the set of uncertain code input variables with their associated probability density functions.
In this paper, a description of the methodologies for performing uncertainty and sensitivity analysis developed by each institution is presented. Furthermore, a summary of results for the base case scenario and relevant sensitivity and uncertainty results as well as the insights obtained from these results will be discussed.

[1] https://www.iaea.org/projects/crp/i31033
[2] Sepold, L., et al., 2004, Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45), Report FZKA-6664, Forschungszentrum Karlsruhe.

Email address:

heribertosanchez7@hotmail.com

Confirm that the work is original and has not been published anywhere else:

YES

93

GIF Framework for Risk-Informed Approach for Design and Licensing of Novel Advanced Reactors

Author: Tanju Sofu[1]

[1] *Argonne National Laboratory*

Corresponding Author: tsofu@anl.gov

Novel aspects of numerous advanced reactors require a systematic and technology-neutral approach for identification and categorization of event sequences to support their design and licensing. The risk-informed approach offers an iterative process, complementary to the traditional deterministic approach, for alternative searches of event sequences by examining both their probability and consequences to understand the risks and identifying additional requirements or regulatory actions as needed. The approach can also support safety classification of plant equipment and defense-in-depth

(DiD) assessment as an integral part of the process to ensure compliance with the safety design criteria and establish links between required safety functions and design requirements.

This paper will summarize the ongoing work on joint development of a risk-informed approach between GIF Risk and Safety Working Group (RSWG) and OECD/NEA's Working Group on Safety of Advanced Reactors (WGSAR). The paper will introduce foundational concepts and main elements of a risk-informed design process to establish the event sequence categories considered in design, integrate the deterministic input and risk insights to identify and classify the event sequences in each category,outline a process to classify the plant equipment based on their risk-significance and the role in plant safety, and assess the alignment of event sequence categories considered in design with the DiD levels.

Confirm that the work is original and has not been published anywhere else:

YES

Email address:

tsofu@anl.gov

94

Insights for Risk-Informed Approaches to Sizing Emergency Planning Zones

Author: Shilp Vasavada[1]

Co-authors: Marie Pohida [1]; Todd Smith [1]; Stacey Rosenberg [1]

[1] *United States Nuclear Regulatory Commission*

Corresponding Author: shilp.vasavada@nrc.gov

Emergency preparedness provides the final layer of defense-in-depth for protection of public health and safety. Requirements for emergency plans will include an emergency planning zone (EPZ) when planning for prompt protective actions is warranted. The size of an EPZ can be scaled to reflect the risk profile of a particular nuclear facility as determined by a risk-informed analysis of the consequences from a spectrum of accidents with considerations given to accident frequency and other factors. The NRC and designers of innovative and evolutionary reactors are seeking to expand the use of risk-informed approaches to determine the appropriate EPZ size for a particular design and site.

A risk-informed approach can be supported by contemporary probabilistic risk assessments (PRAs). Advantages and challenges of using PRAs to support EPZ sizing will be discussed. This paper will identify strategies to address prominent challenges, such as the treatment of uncertainty, justifying screening thresholds for accident sequences, consideration of cliff-edge effects, and performance monitoring. These strategies will be shown to be consistent with the principles of risk-informed regulation.

Email address:

shilp.vasavada@nrc.gov

Confirm that the work is original and has not been published anywhere else:

YES

95

Uncertainty and Sensitivity Analysis of the PWR and SMR by Means of Severe Accident Codes in the Framework of the IAEA CRP I31033

Author: Kwang-Il Ahn[1]

Co-authors: J.M. García [2]; S. Ali [3]; B. Foad [4]; X. Cao [5]; Y.J. Cho [6]; B. Khuwaileh [7]; R. Pericas [8]; T. Jevremovic [9]; H. Rheman [9]; F. Mascari [10]; A. Bersano [10]; F. Gabrielli [11]; D. Luxat [12]; L.I. Albright [12]; J. Ortiz [13]; N.I. Ryzhov [14]; S.M. Petoukhov [15]

[1] *KAERI, Republic of Korea*

[2] *CNEA, Argentina*

[3] *DNPER/PAEC, Pakistan*

[4] *ENRRA, Egypt*

[5] *SJTU, China*

[6] *KINS, Republic of Korea*

[7] *Univ. of Sharjah, UAE*

[8] *ENSO, Spain*

[9] *IAEA, Austria*

[10] *ENEA, Italy*

[11] *KIT, Germany*

[12] *SNL, USA*

[13] *ININ, Mexico*

[14] *IBRAE RAN, Russian Federation*

[15] *CNL, Canada*

Corresponding Author: kiahn@kaeri.re.kr

The International Atomic Energy Agency (IAEA) launched in 2019 the Cooperative Research Project (CRP) I31033 entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors" [1]. The main goal is to advance the understanding and characterization of sources of uncertainty and investigate their effects on the key figure-of-merits (FOMs) of the severe accident codes predictions in water cooled reactors (WCRs). Having this in mind, within the five-year duration of the CRP, 22 organizations representing 18 Member States have been developing and evaluating calculation platforms based on the currently-available severe accident codes and relevant uncertainty tools for uncertainty and sensitivity (U&S) analyses of severe accident scenarios in different WCRs, i.e. PWR, BWR, CANDU, VVER, and SMR. In the CRP framework, the U&S analyses of the QUENCH-06 experiment [2] has also been performed to give relevant insights.

With respect to the Pressurized Water Reactor (PWR) and Small Modular Reactor (SMR) group, the following eight institutions in seven Member States participated in the project to share their own methods for the U&S analyses and relevant results: 1) DNPER/PAEC, 2) ENRRA, 3) KAERI, 4) KINS, 5) SJTU, 6) Univ. of Sharjah, 7) CNEA, and 8) ENSO. All the group participants employed their own scope and calculation framework for the U&S analyses, covering six types of PWRs (ACP1000, KWU-PWR1300, OPR1000, APR1400, CPR600, and SMR), five severe accident analysis codes (MELCOR, MAAP5, ATHLET, RELAP/SCDAPSIM3.5, and RELAP5/NESTLE-based 3KEYMASTER), especially with somewhat different versions in the case of MELCOR, four representative accident scenarios (LBLOCA, SBLOCA, SBO, and STSBO), and two accident progression phases (in-vessel only and in-/ex-vessel both). The relevant FOMs were defined and the underlying code input parameters for the U&S analyses, which might affect the defined FOMs, were proposed after a careful screening-out process by each participant. The corresponding probability distribution functions (PDFs) were assigned based on the relevant code manuals, literature survey, engineering judgment, and parametric analysis wherever necessary. Per each partner, a suit of sensitivity/importance measures was also employed to obtain more credible insights into the influence of each input on the defined FOMs. Then, each partner carried out separately the planed tasks such as: 1) plant modeling and nodalization, 2) simulation of the reference cases, and 3) assessment of the U&S through a coupling of the relevant SA codes with the corresponding uncertainty and sensitivity quantification tools), and general conclusions were drawn based on the analysis of the results. This paper presents the results of the U&S analyses performed by the PWR/SMR group and relevant insights in view of the best-practice application of the U&S approaches for severe accident analyses in WCRs.

[1] https://www.iaea.org/projects/crp/i31033
[2] Sepold, L., et al., 2004, Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45), Report FZKA-6664, Forschungszentrum Karlsruhe.

Email address:

kiahn@kaeri.re.kr

Confirm that the work is original and has not been published anywhere else:

YES

96

CAPSULE-TYPE AND LOOP-TYPE EXPERIMENTAL RIGS FOR IRRADIATION TESTING OF MOLTEN SALT REACTOR MATERIALS

Authors: Alexander Tuzov[1]; Alexey Izhutov[2]; Alexey Petelin[3]; Andrey Burukin[3]; Natalia Kalinina[3]; Pavel Palachev[4]

[1] *Research Institute of Nuclear Reactors, State Scientific Centre, Joint-Stock Company (RIAR JSC)*

[2] *JSC "SSC RIAR"*

[3] *Research Institute of Nuclear Reactors, State Scientific Centre, Joint-Stock Company*

[4] *Research Institute of Nuclear Reactors, State Scientific Centre, Joint-Stock Company*

Corresponding Author: niiar@niiar.ru

Nowadays, such countries as Russia, China, USA and others perform calculations and experiments focused on the molten salt reactor (MSR) concept.
This project requires, first of all, the justification of structural materials to have the appropriate corrosion and radiation resistance under the MSR operating conditions. The MSR materials should undergo both out- and in-reactor testing under the relevant condition.
Capsule-type irradiation rigs (IR) have been developed to test candidate MSR materials in the SM-3 reactor in the environment LiF-BeF2 and LiF-NaF-KF.
The IR allows testing flat and cylindrical (small dog-bone) samples in molten salts. The samples are placed into the capsules filled with molten salt and sealed. Each capsule is made of the same material as the sample. Up to thirty capsules are inserted into the IR that also has tellurium-containing capsules and samples to be tested in the helium environment. In total, one IR may contain up to fifty samples to be tested in the molten salt, up to fifty samples to be tested in the tellurium vapors and up to hundred samples to be tested in the helium environment.
The IR design provides for maintaining the test temperature from 500℃ to 750℃ depending on the requirements to the material and type of molten salt.
A loop design has been developed for the SM-3 reactor with a natural molten salt circulation to test structural materials under the simulated operating conditions of both MSR vessel and internals.

Email address:

adm@niiar.ru

Confirm that the work is original and has not been published anywhere else:

YES

97

SYNERGIZING DETERMINISTIC AND PROBABILISTIC SAFETY ANALYSES - A HOLISTIC APPROACH TO SAFETY AT LEIBSTADT NPP, SWITZERLAND

Authors: Janaki Devi Kompella[None]; Olivier Nusbaumer[1]; Kai Fischer[1]

Co-authors: Pavol Zvoncek [1]; Mathias Trautmann [1]; Venkata Saikumar Bulusu [2]; Vignesh Anandan [2]; Karthik Ravichandran [2]

[1] *Kernkraftwerk Leibstadt, Switzerland*

[2] *RELSAFE PRA Consulting Private Limited, Mumbai, India*

Corresponding Authors: devi.kompella@relsafe.co.in, olivier.nusbaumer@kkl.ch

The operating Swiss NPPs maintain Deterministic and Probabilistic Safety Assessments (DSA & PSA) as 'living' studies with periodic updates, given the prominent role of both analyses in the Integrated Regulatory Safety Oversight process promoted by Swiss Federal Nuclear Regulatory Authority (ENSI). ENSI recently issued augmented requirements and acts in different technical areas, to ascertain the adequacy of safety measures for long term operation (LTO) of existing NPPs.

Kernkraftwerk Leibstadt (KKL) undertook major initiatives to fulfil new ENSI requirements and to stay up-to-date with international developments, launching several sophisticated projects considering recent operating experience, plant modifications and technical advancements in methods, standards and data. In particular, the projects include Deterministic and Probabilistic Assessment of Internal Fire events, Internal Flooding events, Seismic events (considering the new hazard assumptions), and Seismically Induced Internal Fire and Flooding (SIFF) events. The latest evolutions in international state-of-the-art (USNRC/EPRI/NEI/ ASME/IAEA guidance) were applied in these projects. For example, the assessment of internal fire events involved generic fire ignition frequencies taken from NUREG-2169, fire simulations performed using CFAST & FDS in accordance with NUREG-1934, NUREG-1805, NUREG-2178, NUREG/CR-7010, transient fire modelling techniques based on NUREG-2232, NUREG-2233 & EPRI 3002005303, analysis of multiple spurious operations (MSOs) as per NEI 00-01, USNRC RG 1.189 and NUREG/CR-7150 and many more.

One major highlight in these projects was the synergy established between deterministic and probabilistic studies. Methodologies were formulated with due consideration to the individual goals of both studies and harmonizing the technical basis wherever deemed effective. KKL adopted a holistic approach to plant safety by synergizing DSA and PSA studies, enhancing the level of consistence and confidence in risk informed decision making.

This paper summarises the interesting new aspects, observations and insights from these recent deterministic and probabilistic studies at KKL.

Email address:

devi.kompella@relsafe.co.in & olivier.nusbaumer@kkl.ch

Confirm that the work is original and has not been published anywhere else:

YES

98

VERIFICATION OF SSC SAFETY CLASSIFICATION ACCORDING TO IAEA SSG-30 FUNCTIONAL APPROACH: BENEFITS OF DSA AND PSA INTEGRATION

Authors: Janaki Devi Kompella[None]; Kai Fischer[1]; Olivier Nusbaumer[1]

Co-authors: Mathias Trautmann [1]; Pavol Zvoncek [1]; Karthik Ravichandran [2]; Rahul Agarwal [2]

[1] *Kernkraftwerk Leibstadt, Switzerland*

[2] *RELSAFE PRA Consulting Private Limited, Mumbai, India*

Corresponding Authors: devi.kompella@relsafe.co.in, olivier.nusbaumer@kkl.ch, karthik.ravichandran@relsafe.co.in

Recognizing the need for a more structured approach to safety classification of Structures, Systems and Components (SSCs) in Nuclear Power Plants (NPP), IAEA established requirements in Safety Standard IAEA SSR-2/1 and detailed methodological guidance in Safety Guide SSG-30 and Technical Document TECDOC-1787. The guidance specifies that safety classification of SSCs should be determined primarily by deterministic methods based on a functional approach and be complemented by Probabilistic Safety Assessment (PSA). The guidance also reaffirms that safety classification is an iterative process, adequacy of which should be verified throughout the plant lifetime.

In Switzerland, guideline ENSI-G01 governs the safety classification of SSCs in NPPs complemented by recently updated guidance like ENSI-G02 and ENSI-A01, which are aligned with the IAEA SSG-30 functional approach. Swiss Atomic Energy ordinance KEV promulgates use of PSA for supporting the safety classification process; guideline ENSI-A06 provides criteria for the evaluation of the risk significance of SSCs. Consistency between the deterministic and probabilistic approaches is therefore of paramount importance, ensuring appropriate safety classification.

In view of the new ENSI and IAEA guidance, Leibstadt Nuclear Power Plant (KKL) proposed to carry out a detailed verification of the safety classification of KKL SSCs. The study was planned to be carried out in two phases. Phase 1 was a pilot study which included application of the functional approach with seismic event as the Postulated Initiating Event (PIE). Phase 2 will consider all other PIEs individually. In addition, risk significance of SSCs will also be determined using KKL's integrated multistate, all events PSA model based on ENSI-A06 and NEI-00-04 guidance. The insights from PSA will be used to support the deterministic safety classification process thus aligning with ENSI and international requirements.

This paper discusses the process followed and the insights gained from the Phase 1 study.

Email address:

devi.kompella@relsafe.co.in & olivier.nusbaumer@kkl.ch

Confirm that the work is original and has not been published anywhere else:

YES

99

Toward developing a novel combined Licensing & Safety approach for advanced nuclear reactors based on the international maritime and nuclear safety framework. Case study of the CMSR Power Barge.

Authors: Bernat Cirera[None]; Sorouche Mirmiran[1]; Federico Fuente Espel[2]; Peder Norborg[2]; Ivanna Rodriguez Ruocco[2]

[1] *Seaborg Technologies*

[2] *Seaborg Technologies ApS*

Corresponding Authors: smi01@seaborg.com, bcr01@seaborg.com

The Compact Molten Salt Reactor (CMSR), being developed by Seaborg Technologies, is a SMR and as per its name is characterized by a core in which the fuel is in a molten state dissolved in fluoride salts. The CMSR is integrated into a floating non-self-propelled barge (covering the milestone phase

to decommissioning lifecycle). Molten salt reactors such as the CMSR cannot melt down and offer significant advances and advantages compared to solid fuel LWRs. Most of the technology neutral IAEA safety guidelines were developed based on LWRs. Furthermore, the SMR characteristics of the CMSR combined with its transportability in maritime environment represent new licensing opportunities based on the International Maritime Organization (IMO) framework. In the absence of a consistent and harmonized framework to enable international licensing and safety considerations for SMRs and specifically for the CMSR combining SMR, maritime, innovative nuclear design and existing nuclear safety approaches, Seaborg is taking a proactive step toward proposing a novel combined licensing & safety approach. The Seaborg approach is based on the issuance of an International Licensing & Safety Case covering both the Seaborg safety approach and the Seaborg licensing approach. The Seaborg safety approach integrates in a systematic and traceable manner international and relevant recognized national guidelines. Requirements are redefined when relevant to the CMSR to provide a basis combining deterministic and probabilistic approach through a deliberative process justifying a risk informed performance-based approach to safety. The Seaborg licensing approach with the involvement of international stakeholders and recognized IMO framework addresses an important number of IAEA recommended functions and processes traditionally associated with the licensee and host state authorities. This approach covers the development of IAEA milestones and up to and including decommissioning aspects. The added benefits of the Seaborg licensing approach are to lessen the burden on licensee and host state authorities, and providing a predictable licensing path for all stakeholders. Furthermore, this is facilitated by the reduced site-specific considerations of the CMSR and its transportability, SMR serial and standardized production aspects increasing efficiency, safety and lessons learned, shipyard-fuelled improving safeguards, harmonized integration of operating experiences, experienced supply chain, etc. While lessening the burden on the licensee and host state authorities, by means of digital transparency and traceability and the Seaborg licensing process itself, licensee and host state authorities retain an overview and required capabilities to understand and intervene for all matters related to safety throughout the whole lifecycle of the CMSR when necessary and relevant. The development of the combined Seaborg Licensing and Safety approach is an opportunity for the development of SMRs. It provides a systematic safety & licensing roadmap applicable to other SMRs particularly micro and mobile reactors. Additionaly, it provides a basis for discussion and recommendations for the improvement of the existing IAEA and IMO frameworks to facilitate future SMRs development including consideration by the international community of non-safety relevant provisions or absence thereof, strengthen safety, and increase international cooperation for the peaceful use of nuclear energy tailored to SMRs and associated non-traditional use and actors.

Email address:

bcr01@seaborg.com

Confirm that the work is original and has not been published anywhere else:

YES

100

Safety Approaches for the CMSR Power Barge

Authors: Bernat Cirera[None]; Federico Fuente Espel[1]; Ivanna Rodriguez Ruocco[1]; Peder Norborg[1]; Sorouche Mirmiran[2]

[1] *Seaborg Technologies ApS*

[2] *Seaborg Technologies*

Corresponding Authors: smi01@seaborg.com, bcr01@seaborg.com

The Compact Molten Salt Reactor (CMSR), being developed by Seaborg Technologies, is a Small Modular Reactor (SMR) and as per its name is characterized by a core in which the fuel is in a molten state dissolved in fluoride salts. The CMSR is integrated into a floating non-self-propelled barge (covering the milestone phase to decommissioning lifecycle). Molten salt reactors such as the CMSR cannot melt down and offer significant advances and advantages compared to solid fuel light water reactors

(LWR). Most of the technology neutral International Atomic Energy Agency (IAEA) safety guidelines were developed based on LWRs.Furthermore, the SMR characteristics of the CMSR combined with its transportability in maritime environment represent new challenges for applying traditional safety approaches such as the "dentification, categorization and grouping of postulated initiating events and accident scenarios" (SSG-2) and the "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants" (SSG-30). This paper discusses the established methodologies to identify postulated initiating events for the CMSR. It also presents the methodology developed to address Safety Classification of structures, systems and components in harmony with IAEA.
To identify, categorize and group postulated initiating events and accident scenarios, a set of tools have been applied to the CMSR Power Barge. Such as a "Master Logic Diagram (MLD)" and a "Functional Failure Modes and Effects Analysis (FFMEA)". The MLD is a "top-down" qualitative safety analysis whose purpose has been to identify the possible initiating events of the CMSR Power Barge, through a deductive and structured approach (logic based on Fault Tree Analysis). Whereas the FFMEA is a "bottom-up" qualitative safety analysis used to identify functions and structures, systems and components, and sets the foundations for the first quantitative safety analysis on the frequency of accident scenarios. Using both tools, a series of postulated initiating events and accident scenarios have been identified and categorised as anticipated operational occurrences, design basis accidents and design extension conditions.
To classify the structures, systems and components, a multi-step process following the IAEA guidance has been applied to the CMSR Power Barge. The process has involved a group of suitably qualified and experienced individuals and has identified and categorised the safety functions of the CMSR Power Barge, classified its structures, systems and components and established engineering rules for mechanical, electrical, control and instrumentation and naval engineering disciplines.

Email address:

bcr01@seaborg.com

Confirm that the work is original and has not been published anywhere else:

YES

101

ADVANCED PROGRAM FOR IRRADIATION TESTING OF FUEL AND MATERIALS TO JUSTIFY SAFETY OF EVOLUTIONAL AND INNOVATIVE DESIGNS FOR NUCLEAR REACTORS

Authors: Alexey Izhutov[1]; Alexey Petelin[2]; Andrey Burukin[2]; Igor Zhemkov[2]; Vladimir Ovchinnikov[2]

[1] *JSC "SSC RIAR"*

[2] *Research Institute of Nuclear Reactors, State Scientific Centre, Joint-Stock Company*

Corresponding Author: izhutov@niiar.ru

The improvement of performance and economic features and enhancement of safety criteria for operated water-cooled reactors (VVER, PWR, BWR) require the state of reactor materials to be analyzed to justify and extend their lifetime, to introduce more efficient fuel cycles, and to create technologies for the multiple reuse of regenerated fuel as well as fuel more tolerant to design-basis and beyond-the-design basis accidents (accident-tolerant fuel).
The further development of two-component nuclear power industry with a closed nuclear fuel cycle, controlled fusion, advanced energy systems, construction of reference NPP units, including low-power nuclear power plants, is to be based on the results obtained using innovative next-generation reactors (fast reactors with sodium and heavy metal coolants, supercritical water reactors, molten salt reactors, high-temperature gas-cooled reactors, etc.).
The creation of safe and competitive Gen-IV nuclear facilities requires new materials, the operating conditions of which are more extreme than today. Structural materials must ensure reliable operation of core components, at least until damage doses of 170 dpa, and even higher in future. Fuel safety must be experimentally justified in normal operation modes, under transient and cyclic conditions and design-basis accident conditions.
RIAR JSC, being one of the world's largest nuclear research centers, implements the full cycle of

irradiation tests, post-irradiation examinations of fuel, structural and absorbing materials of the cores of various nuclear reactors. RIAR operates five high-flux nuclear research reactors (SM-3, MIR.M1, BOR-60, RBT-10/2 and RBT-6) with unique experimental characteristics and the world's largest chains of hot cells equipped with modern equipment for post-irradiation examinations of materials and core components, experiments on the processing of irradiated nuclear fuel and disposal of radioactive waste.
The paper presents the key areas of activity and advanced program for irradiation testing of fuel and materials at RIAR's research reactors to justify the safety of evolutionary and innovative designs for nuclear reactors.

Email address:

adm@niiar.ru

Confirm that the work is original and has not been published anywhere else:

YES

103

Application of Lines of Defense (LoD) methodology for defensein depth implementation in the design of Gen4 reactors

Authors: Etienne Courtin[None]; Nathalie Playez[None]

Corresponding Author: etienne.courtin@framatome.com

Defense in depth is widely acknowledged as a cornerstone of nuclear safety. However practical implementation of this concept may be a challenge in the early stage of a new reactor design, in particular when it comes to the assessment of the sufficiency of the features implemented.
In France, the Line of Defense concept (LoD) has been successfully used for a long time in the design and licensing of research reactors and advanced reactors like sodium fast reactors (SFR). It is now used in the design of new reactors, like ASTRID SFR or molten salt fast reactor that is investigated in the frame of the European SAMOSAFER R&D project. The implementation of this concept at an early stage of the design guarantees that it meets the safety objectives related to defense in depth.
Basically the principle is that unacceptable consequences of an accident situation, like large or early releases, shall always be prevented by at least three LoDs that are independent, diversified and reliable enough. This concept shall include, as appropriate, specific regulatory requirements related to the prevention and mitigation of Severe Plant Conditions, like fuel melting if applicable. It is also used to achieve the demonstration of practical elimination of severe plant conditions that are not reasonably manageable. This methodology is applicable to any kind of technology and provides a robust deterministic assessment of the plant safety.
This paper describes the nature of the LoDs, the general requirements associated to their design and provides example of application in current advanced reactor projects.

Email address:

etienne.courtin@framatome.com

Confirm that the work is original and has not been published anywhere else:

yes

104

Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors

Authors: Christophe Herer[1]; Giacomo Grasso[2]; Timothy Drzewiecki[3]; Walter Klein-Heßling[None]; David Plummer[4]; Nikolay Khrennikov[5]; Valerii Pivovarov[5]

[1] *Institut de Radioprotection et de Sûreté Nucléaire (IRSN)*

[2] *Italian National Agency for New Technology, Energy and Sustainable Economic Development (ENEA)*

[3] *US Nuclear Regulatory Commission (US NRC)*

[4] *Institute of Physics*

[5] *Scientific and Engineering Centre for Nuclear Radiation Safety (SEC NRS)*

Corresponding Author: walter.klein-hessling@grs.de

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. Regulatory agencies establish requirements and/or expectations on the nuclear power plant designer or licensee for the development and use of analytical codes and methods in order to ensure the quality, credibility, and confidence in the analyses produced by the analytical codes and methods. In addition, regulatory agencies have used analytical codes and methods to perform confirmatory analyses as part of due diligence during a regulatory review.

Based on the responses to a survey from Canada, France, Germany, Italy, Russia, UK and USA, the Task Group on Analytical Codes and Methods (TGACM) of the OECD-NEA Working Group on the Safety of Advanced Reactors (WGSAR) has performed a review to (1) identify and clarify the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants, and (2) identify best practices for the use of confirmatory analyses by regulatory agencies.

The paper summarizes the report prepared by TGACM, which includes the identification of phenomena relevant to each of the six reactor concepts regarded by Generation-IV International Forum (GIF). Modelling requirements on key topics (such as reactor physics, core thermal hydraulics and so on) are described briefly. These insights could help guide the assessment and qualification of simulation platforms inherent to development and deployment of the innovative designs.

Email address:

walter.klein-hessling@grs.de

Confirm that the work is original and has not been published anywhere else:

YES

105

Determination of Temperature Condition of absorbing Elements for Design Basic Conditions and Design Extension Conditions of Nuclear Power Plants

Author: Michail Polevoy[1]

Co-authors: Denis Gordeev [2]; Vladimir Ydakhin [2]; Pavel Yagov [2]

[1] *OKB "Gidropress"*

[2] *JSC "Gidropress"*

Corresponding Author: mpolevoy89264267726@mail.ru

Absorbing elements is one of the part of control rod of control and protection system in nuclear power plants with VVER. It is used for fast stopping of nuclear reaction in the reactor core, maintaining power at a particular level, transferring power from one level to another, for peak power flattering in the reactor core and for suppression of xenon pulsations. Also for justification of operational of absorbing elements, it is important to know its working conditions.

Previously in order to justify the operability of absorbing elements for design basic conditions and design extension conditions of nuclear power plants, the temperature conditions of the fuel elements were taken as initial dates. But as the absorbing element is located in a guiding channel the cooling condition for absorbing element is different from fuel elements.
It creates the need for determination of a temperature condition of an absorbing element and performing the analysis that allows demining the more realistic changes for temperature of absorbing elements in abnormal conditions of nuclear power plants. This analysis also allows us to check that there is no melting in absorbing elements and its structural materials.
The analysis of the absorbing element is provided using a thermohydraulic code
KORSAR\GP. This work was performed separately for modes with leaking from the primary circuit and without leaking from the primary circuit. For modes without leaking from the first circuit, we considered two groups with action of an emergency protection system and without action of an emergency protection system. For modes with leaking from primary circuit, we considered the mode "Accident with loss of coolant through a large leak with failure of the active part of the emergency cooling system of low pressure zone".
This work shows the realistic temperature condition of the absorbing element and its structural materials. The results show that there is no melting in absorbing elements and control rods cladding for anticipated operational occurrences of nuclear power plants. The results of this work were used for the projects of nuclear power plants with VVER-1200.

Email address:

mpolevoy89264267726@mail.ru

Confirm that the work is original and has not been published anywhere else:

Yes

106

Harmonizing mechanical codes & standards for innovative reactors

Author: Ronan Tanguy[1]

Co-authors: Byung-Chan Na [2]; Nawal Prinja [3]; Frédéric Beaud [4]

[1] *World Nuclear Association*

[2] *WNA*

[3] *Jacobs*

[4] *EDF*

Corresponding Author: ronan.tanguy@world-nuclear.org

The Cooperation in Reactor Design Evaluation and Licensing Working Group (CORDEL) of the World Nuclear Association (WNA) promotes the development of a worldwide nuclear environment where internationally accepted standardized reactor designs can be deployed without major design changes. As reactor designs are standardized at a high level through IAEA SSR-2/1, CORDEL worked to identify the most influential requirements within this safety standard and align their interpretations by national regulators. Engineering design rules (Requirement 18) are a key requirement and their application heavily influences reactor designs via applicable codes and standards (C&S) in a given country. The CORDEL Mechanical Codes and Standards Task Force (MCSTF) was set up in 2011 to promote the harmonization of these C&S.
The MCSTF proposed harmonized approaches for several topics within international C&S. These approaches have their foundations in industrial best practice and are informed by experts who have contributed to MCSTF reports. A standardized methodology has been employed across the topics which initially compares the requirements and scopes of C&S to determine where areas of equivalence already exist, where discrepancies lie and gaps where scopes do not overlap. Following the comparison, benchmarking exercises are undertaken to determine the differences in outcomes for the application of the various codes on standard class 1 nuclear components. The results of the benchmarking exercises were used to propose recommendations for harmonized practices. To date,

the MCSTF has completed work on several key topics:
- The certification of non-destructive examination personnel. The MCSTF compared existing requirements across nuclear and non-nuclear codes and proposed a recommendation for convergence in certification requirements for fabrication.
- A comparison of welding qualification and welding quality assurance across six international C&S.
- Proposed harmonized practices for non-linear analysis design rules, based upon code comparisons and benchmarking exercises.
- A comparison of requirements in C&S for fatigue life analysis design rules.
- Highlighting the challenges faced by the nuclear industry in introducing advanced manufacturing techniques with regards to regulation and C&S.

This paper presents an overview of the MCSTF's work and methodologies with a focus on its recent publications; the Non-Linear Analysis Design Rules Series and the Advanced Manufacturing paper. The first of these is a three part study on existing mechanical design C&S with the aim of proposing identifying a more harmonized approach in using non-linear analysis methods. An initial comparison showed that many different approaches are currently used within the industry, this gives rise to discrepancies. The impact of choices made by analysts in setting up simulations and assessing their outcomes was a notable source of divergence highlighted in the report. The MCSTF therefore proposed best practices and recommended that standards developing organizations should seek to include them in their C&S.
The paper on nuclear advanced manufacturing techniques presents current initiatives in the domain by WNA members and pathways to overcome current regulatory challenges.
Regulatory acceptance of these techniques and design approaches will be key in enabling the licensing and deployment of innovative reactors over the next decade.

Email address:

ronan.tanguy@world-nuclear.org

Confirm that the work is original and has not been published anywhere else:

YES

107

GIF LFR Safety Design Criteria

Authors: Alessandro Alemberti[None]; Kamil Tucek[1]; Mariano tarantino[2]; Minoru Takahashi[3]; Toru Obara[3]; Masatoshi Kondo[3]; Andrei Moiseev[4]; Lev Tocheny[5]; Iuliia Kuzina[6]; IL SOON HWANG[7]; Craig Smith[8]; Anton Moisseytsev[9]; Yican Wu[10]; Ming Jin[11]

Co-author: Laura VIDELA [12]

[1] *European Commission, Joint Research Centre*

[2] *ENEA*

[3] *Tokyo Institute of Technology*

[4] *JSC NIKIET*

[5] *JSC "NIKIET"*

[6] *IPPE JSC*

[7] *Seoul National University*

[8] *Naval Postgraduate School, Moterey, US*

[9] *Argonne National Laboratory*

[10] *Institute of Nuclear Energy Safety Technology (INEST),Chinese Academy of Sciences*

[11] *International Academy of Neutron Science (IANS), Qingdao, People's Republic of China People's Republic of China*

[12] *IAEA*

Corresponding Authors: mariano.tarantino@enea.it, kamil.tucek@ec.europa.eu, m.l.videla@iaea.org

The Lead-cooled Fast Reactor (LFR) Safety Design Criteria (SDC) document was prepared to identify a set of criteria specifically tailored to the Generation IV LFR system. The objective was to

present a set of reference criteria for the safety design of structures, systems and components of LFRs with the aim of achieving the safety goals of the Generation IV reactor system, and to support licensing processes. A set of eighty-two reference criteria for the LFR system is systematically and comprehensively laid out in the SDC. The full paper will: (i) describe the background, objectives, and formulation of principles; (ii) outline the safety approach to the LFR as a Generation IV reactor system; and (iii) summarize LFR-specific revisions to the IAEA SSR 2/1 requirements for the overall safety design and safety design of specific LFR structures, systems and components.

Confirm that the work is original and has not been published anywhere else:

kamil.tucek@ec.europa.eu

Email address:

mariano.tarantino@enea.it

108

Defense-in-Depth – Challenges today and tomorrow for I&C architecture

Author: Allan Carson[1]

Co-authors: Byung-Chan Na [1]; Johannes Pickelmann [2]

[1] *World Nuclear Association*

[2] *Framatome*

Corresponding Authors: byung-chan.na@world-nuclear.org, allan.carson@world-nuclear.org, johannes.pickelmann@framatome.com

Since the nuclear Industry started to grow as an international industry, inconsistencies in the definitions of terms, attributes, assessment methodologies, and scope associated with the concepts of 'defence-in-depth' and 'diversity' have led to significant challenges in design, licensing, and cost of nuclear power plants.

The differences between these definitions were first investigated in CORDEL report Safety Classification for I&C Systems in Nuclear Power Plants: Comparison of Definitions of Key Concepts and are further expanded upon in CORDEL report expanded upon in CORDEL report Defence-in-Depth and Diversity: Challenges Related to I&C Architecture.
Traditionally redundancy and separation of structures and components – such as the use of identical equipment in a four/three divisional arrangement – was an acceptable approach to meet the N+2 criterion2 and thereby demonstrate diversity. However, the N+2 criterion has now been extended by the conservative assumptions associated with digital I&C and thus digital CCF has come to replace redundancy as the main driver for designing diverse digital protection systems. This has in turn affected the development of I&C design for the main line of defence (e.g. protection systems).

The report calls for greater clarity among regulators and industry in relation to terms and terminology particularly around "Defence-in-Depth" and "diversity" which should be distinguished from each other and calls for further work on areas such as: quantification of diversity attributes, different defence-in-depth approaches between regulators and, clarification of rules for mitigation of CCF.
The recent CORDEL report on HDL programmed device (HPD) technology and implications for safety critical applications in nuclear power plants further outlines the lack of standards and common approaches to the treatment of CCF within digital devices and highlights inconsistences in the guidance that does exist.

The report highlights that the current trend of using HPD technology embedded within and used for control systems within nuclear power plants will only increase with the development of emerging designs and it is important that there is clear and consistent approach to the treatment of CCF within these technologies to facilitate wide deployment of these designs around the world.
As a result the report calls for consistency in industry standards that address the treatment of CCF in

systems performing Category A B or C functions, updating of regulatory requirements and common positions, which are currently limited to Category A functions, to include systems and components that perform Category B & C functions and, for harmonization across regulator and industry to provide clear and consistent guidance for the treatment of CCF in digital I&C for all safety related systems and components.

Email address:

allan.carson@world-nuclear.org

Confirm that the work is original and has not been published anywhere else:

Yes

109

A New Paradigm for Reactor Design Licensing – collaboration the key

Author: Allan Carson[1]

Co-authors: Claude Mayoral [2]; Byung-Chan Na [3]

[1] *World Nuclear Association*

[2] *Edvance*

[3] *World Nuclear Assocaition*

Corresponding Authors: allan.carson@world-nuclear.org, claude.mayoral@edvance.fr, byung-chan.na@world-nuclear.org

The Cooperation in Reactor Design Evaluation and Licensing (CORDEL) Working Group of World Nuclear Association report on different interpretations of regulatory requirements describes how despite the same fundamental safety requirements being agreed by all countries there remain a multitude of examples where different regulatory interpretations of requirements and how to best manage the perceived risks, result in significant design changes when licensing outside of country-of-origin.

The report reviews examples of GEN III LWR reactor designs where different interpretations or application of fundamental safety requirements such as application of Defence-in-depth, approach to DBA and DEC, and safety classifications of systems and components have resulted in significant design changes when licensing the same reactor design in different countries.

The report highlights that the nuclear industry is being held back by our national approaches to regulation that reset projects outside the reactors country of origin to FOAK deployments. To learn the lessons from this previous experience and support Nth of a Kind (NOAK) deployment of emerging reactor designs it is critical that greater collaboration between stakeholders is facilitated as soon as possible.

This point is further supported in the CORDEL report design maturity and regulatory expectations for SMRs which through analysis of member surveys identified best practices and recommendations for the nuclear industry and regulators when undertaking licensing of emerging reactor designs. These included: the importance of gap analyses when crossing national borders; a systematic approach to recording major design modifications, upgrades, safety decisions, and methodologies; development of safety case elements and early/frequent regulatory engagement; the benefit of engagement with international bodies; design maturity at appropriate point in regulatory engagement; and regulators' engagement with international peers.

Both reports identify the urgent need to streamline the existing national approaches to regulation through an expansion of international collaborative efforts by both regulators and industry to facilitate the wider deployment of evolutionary and innovative designs. Such efforts could minimize

the design changes required when licensing in new countries, reduce the project development risks, and thus facilitate the wide scale deployment of evolutionary and innovative designs. This would require a paradigm shift in cooperation and an alternative framework to facilitate the collaborative activities.

World Nuclear Association CORDEL report Harmonization for reactor design evaluation: lessons from transport outlines just such a framework in which regulators, supported by governments and industry, undertake joint regulatory assessments, joint recognition assessments, build trust and develop joint safety requirements and outcomes.

Such a framework requires a shift in current approaches but there are many steps along this journey and changes would not be overnight. The initial proposal would be to start small and develop the process and bounding criteria for a small number of diverse regulators, following which additional regulators would join and expand the framework.

Email address:

allan.carson@world-nuclear.org

Confirm that the work is original and has not been published anywhere else:

Yes

110

U.S. Nuclear Regulatory Commission Guidance on the Acceptability of Probabilistic Risk Assessment Used in Regulatory Decision Making for Non-Light-Water Reactors

Author: Anders Gilbertson[None]

Co-author: Hanh Phan

Corresponding Authors: hanh.phan@nrc.gov, anders.gilbertson@nrc.gov

The U.S. Nuclear Regulatory Commission (NRC) has developed Regulatory Guide (RG) 1.247, Revision 0, "Acceptability of Probabilistic Risk Assessment Results for Advanced Non-Light-Water Reactor Risk-Informed Activities," which provides the NRC staff positions on an acceptable probabilistic risk assessment (PRA) for non-light-water reactor (NLWR) licensing and risk-informed decision making. RG 1.247 endorses the recently published industry consensus standard for NLWR PRA, ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non Light Water Reactor Nuclear Power Plants," and industry guidance on PRA peer review in NEI 20-09, Revision 1, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard." RG 1.247 provides an approach for determining whether a NLWR design- or plant-specific PRA used to support an application to the NRC is sufficient to provide confidence in the results. RG 1.247 has been issued for trial use to gain experience with its use and implementation in risk-informed NLWR regulatory activities. This paper describes the purpose of and overall development process for RG 1.247 including its derivation from the NRC's RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," which provides the general structure and framework on the staff position on PRA acceptability for light-water reactors.

Email address:

Anders.Gilbertson@nrc.gov

Confirm that the work is original and has not been published anywhere else:

YES

111

DEVELOPING COMPETENCE OF NEW REGULATORS TO MANAGE (WATER COOLED) SMR REACTORS

Author: Fidel Ilizastigui[1]

[1] *Inversiones Gamma S.A.*

Corresponding Author: filizastigui@gmail.com

Due to their multiple advantages and potential uses, it is expected that the deployment of SMRs will soon reach developing countries, including those with no previous nuclear experience. For those countries, the creation of a sound nuclear regulatory framework, supported by an independent, competent and well-resourced regulator represents a considerable challenge and a time-consuming undertaking. The overall challenge is further compounded by scarce or non-existent availability of suitably qualified and experienced personnel in-country with the necessary technical competence to fulfill the nuclear regulatory function. To address this issue, the author suggests an organizational structure of the new SMR regulator which covers the totality of Core Regulatory Functions (CRFs) and Supporting Technical Functions (STFs) defined by the IAEA in the Safety Guide SGS-13, all supported by technical positions which, in his opinion will facilitate the acquisition of the minimum regulatory and technical expertise to allow the new SMR regulator to successfully address all technical areas associated with SMR reactor safety. The allocation of regulatory and technical responsibilities within the suggested organizational structure is meant to serve the basis for hiring staff and managing their competences. An example is given in the paper on the allocation of responsibilities for the review and assessment of a hypothetical Safety Analysis Report for a SMR, prepared as per the IAEA Safety Guide SSG-61. Overall, the suggested Organizational Structure for the new SMR regulator is expected to ensure minimum levels of regulatory and technical competences for the successful discharge of regulator's responsibilities in an effective and efficient manner.

Email address:

filizastigui@gmail.com

Confirm that the work is original and has not been published anywhere else:

YES

112

Application of Objectives-Driven Assurance Cases to System Development in an Evolving Acquisition Model

Authors: Robert Youngblood[1]; H. C. Everett[1]; Homayoon Dezfuli[2]

[1] *Idaho National Laboratory*

[2] *US National Aeronautics and Space Administration*

Corresponding Author: robert.youngblood@inl.gov

System properties such as "safety" and "dependability" cannot, in practice, be proven, and must be argued in an "assurance case" aimed at supporting risk-acceptance decisions that have to be made by system acquirers and/or regulatory authorities (see, e.g., ISO/IEC/15026). The present paper is concerned with applications of the "assurance case" idea early in design and development of new systems, when (apart from dedicated testing) the only available operating experience information derives from previous (non-identical) systems. Much of the present discussion is based on an evolving acquisition model at a US federal agency; previously, most major systems were developed in-house, but some major systems will now be developed by and acquired from commercial providers. Key points discussed include the following.
1. The risk acceptance decisions to be made are very significant, but many stakeholders will wish

to proceed as efficiently as possible through iterations of design in various stages of the system life cycle. Under the new acquisition model, these considerations imply a serious reconsideration of the way in which the development process is managed by both providers and acquirers.
2. By promoting a particular kind of focused discussion between acquirers and providers, the use of assurance cases should be particularly valuable under the new acquisition model.
3. In principle, application of objectives-driven (sometimes called "performance-based") approaches to assurance of performance have significant advantages in cases where they are applicable.

Email address:

robert.youngblood@inl.gov

Confirm that the work is original and has not been published anywhere else:

YES

113

Demonstrating the Inherent Safety Features of Modular HTGR by Simulators

Authors: Chunlin Wei[1]; Jun Sun[2]; Zhe Sui[1]

[1] *INET, Tsinghua University*

[2] *Tsinghua University*

Corresponding Author: sunjun@tsinghua.edu.cn

The modular High Temperature Gas-cooled Reactor (HTGR) technology is treasured by the inherent safety features and high temperature industrial applications. China has been involved in the research, development, deployment and promotion of the modular HTGR for more than 40 years. Institute of Nuclear and New Energy Technology (INET), Tsinghua University endorsed the concept, comprehended the design philosophy, matured the technologies and implemented several engineering projects in China. The Chinese central government fully supported the research and development of the modular HTGR by the test reactor of HTR-10 and the commercial demonstration nuclear power plant of HTR-PM (High Temperature gas-cooled Reactor Pebble-bed Module). The targets of modular HTGR in the future are electricity generation, even including coal-fired plant repowering, cogeneration, and nuclear hydrogen production.

During the research and development of the modular HTGR, various simulators are developed for many purposes, including operators training and certification, operating program verifications, education, etc. By advancing the key models of neutronics, thermal hydraulics, and the once through steam generators, the inherent safety features of modular HTGR are fully demonstrated.

As the demonstration nuclear power plant, HTR-PM should facilitate with its full scope simulator, to train and certificate operators. Therefore, INET firstly developed the first of its kind engineering simulator of HTR-PM, i.e. HTR-ESS, to master the key technologies of HTGR simulator and assist the operating analyses of the HTR-PM in the design and commissioning phases. The requirement of real time and high fidelity simulations were satisfied in HTR-ESS so that the steady states, dynamic operation conditions and accident scenarios could be performed.

Verified computer codes and reactor core models of the HTR-ESS were copied and implemented into the full scope simulator of the HTR-PM, which was used for the training and certifications of the HTR-PM operators from the end of 2015. Besides the training function, the HTR-ESS has supported the commissioning of the HTR-PM by finalizing the operating procedures and commissioning programs.

Based on the HTR-ESS, INET has developed the education tool in the classroom. Lectures on HTGRs' designs and operations would use the education tool to help student understand the advances of this

kind of Generation IV nuclear reactor technology. Practices on the education tool also impresses the students by dynamic operations and safety features in accidents.

The significant inherent safety features, high temperature process heat application potentials and implementation flexibility of the modular HTGR also absorb attentions from more nuclear power embarking countries, such as Indonesia. In the collaboration framework of China-Indonesia Joint Laboratory on HTGR, one HTGR simulator has also been jointly developed for education, training, demonstration, and research.

Since 2019, INET started to collaborate with International Atomic Energy Agency (IAEA) to develop the educational HTGR simulator based on the HTR-ESS. The models and functions are adapted to meet the IAEA requirement of basic educational purposes. The IAEA educational simulator is able to run on laptops and the user interfaces are also newly developed.

Meanwhile, more simulators are under research for better performance and wider applications of modular HTGR.

Email address:

sunjun@tsinghua.edu.cn

Confirm that the work is original and has not been published anywhere else:

YES

114

Numerical analyses of design extension conditions for sodium-cooled fast reactor designed in Japan

Author: Hidemasa Yamano[1]

Co-authors: Shigenobu Kubo [2]; Minako Tokizaki [3]; Hironori NAKAMURA [3]

[1] *Japan Atomic Energy Agency*

[2] *Deputy Director Advanced Fast Reactor Cycle System R&D Center, JAEA*

[3] *Mitsubishi FBR Systems, Inc.*

Corresponding Authors: yamano.hidemasa@jaea.go.jp, kubo.shigenobu@jaea.go.jp, hironori.nakamura.vz@nu.mhi.com, minako.tokizaki.7a@nu.mhi.com

Advanced sodium-cooled fast reactors (SFRs) with enhanced reliability and safety are being designed in Japan. Specific design features in one of SFRs designed in Japan are a passive reactor shutdown system, passive decay heat removal system and a recriticality free concept against anticipated transients without scram (ATWS) in design extension conditions (DECs). A core catcher in the reactor vessel allows us to attain an in-vessel retention, which can reduce the burden to the containment vessel as the last barrier against significant radionuclide release. Numerical analysis methodologies are necessary to demonstrate the effectiveness of safety design measures in DECs. This paper describes numerical analysis methodologies for event sequences studied in Japan and some numerical analyses of DECs to show the effectiveness of the passive shutdown system against a typical ATWS and severe accident mitigation measures for the in-vessel retention of core melt. A cutting-edge technology recently developed is a debris-bed cooling analysis methodology coupled with a computational fluid dynamics code, which can simulate accurate coolant flow near the debris bed with the decay heat removal system capability.

Email address:

yamano.hidemasa@jaea.go.jp

Confirm that the work is original and has not been published anywhere else:

YES

115

Internal flooding safety assessment of small reactor (ACP100) based on advanced 3D simulation software□CNIFA□

Authors: Chao WANG[None]; Chen hui WANG[1]; Xiaoyun HUANG[None]; Meng ZHAO[None]

[1] *China Nuclear Power Engineering Co., Ltd*

Corresponding Author: 1249072964@qq.com

Abstract□Internal flooding in nuclear power plant is a possible internal disaster, which will lead to serious accidents in nuclear power plant, so it is necessary to carry out safety assessment. The small reactor ACP100 independently developed by China National Nuclear Corporation is an advanced small pressurized water reactor, which was approved to start construction in June 2021. Through the self-developed advanced three-dimensional simulation software (CNIFA), this paper evaluates the internal flooding safety of small reactor (ACP100), and puts forward some suggestions for design improvement. Firstly, the physical model of the plant (reduction ratio 1:10) is built to simulate the water flooding spread in the room of the nuclear power plant. The physical experimental results are compared with the simulation results of CNIFA. The results show that CNIFA has good applicability and accuracy in internal flooding safety assessment. Secondly, the safety assessment of the possible flooding event of the electrical plant in ACP100 nuclear power plant is carried out through CNIFA, and the protective measures are taken for the important safety equipment in the plant according to the simulation results.

Key words: small reactor; internal flooding; 3D simulation; safety assessment

Email address:

1249072964@qq.com

Confirm that the work is original and has not been published anywhere else:

YES

116

Research and Development of Passive Emergency Cooling System in HPR1000 Nuclear Power Plant

Author: Hua Zhong[1]

Co-authors: Fengge Zhang ; Zhansheng Liu ; Chuancheng Wei

[1] *□86□13811911564*

Corresponding Author: sunshine399@163.com

This paper develops a Passive Emergency Cooling System for emergency cooling to important electrical safety-class cabinet rooms in the case of the failure of the ventilation system in the design extension conditions of HPR1000 nuclear power plant. The problem of heat export for electrical cabinet room and the determination of the volume of low-temperature liquid nitrogen storage tank were mainly solved by analysis and calculation. And Passive Emergency Cooling System was verified by numerical simulation and experimental research. Passive Emergency Cooling System is the first at home and abroad, which uses liquid nitrogen vaporization with a certain pressure for nitrogen transmission and convection heat exchange. It is a passive system. The process system is simple and effective. After the accident, only the operator of the main control room needs to confirm accident condition and start the system. The passive emergency cooling system increases the diversity of the final heat sink. In the design extension conditions, when the ventilation system fails, Passive Emergency Cooling System can effectively cool the important electrical cabinets within 24 hours, which ensures the reliable operation of the safety-class system and equipment in the design extension conditions, and achieves the purpose of defense in depth, thus ensuring the safety shutdown of nuclear power plant.

Email address:

sunshine399@163.com

Confirm that the work is original and has not been published anywhere else:

YES

117

Study on the configuration scheme of normal residual heat removal system based on SSG-30 safety classification guidelines

Authors: Jie Li[None]; Yue Pi[None]; Huixia Jiang[None]; Pei Yu[None]; Churan Feng[None]; Jiaming Zhao[None]

Corresponding Author: lee0734@sina.com

The new version of the safety classification guidelines, SSG-30, is a top-level guideline that comprehensively regulates and guides the classification of items in nuclear power plants, and reflects the trend of some safety requirements after the Fukushima nuclear accident. Through the application of the SSG-30 guidelines, the classification and design requirements of the normal residual heat removal system of a certain type of PWR nuclear power plant are studied, focusing on how the system meets the safety status requirements under the new safety design concept, how to apply the single fault principle, and the system performance design requirements. Finally, the method of combining deterministic theory and probability theory is used to evaluate and analyze the feasibility of various configuration schemes of normal residual heat removal system.

Email address:

lee0734@sina.com

Confirm that the work is original and has not been published anywhere else:

Yes

118

Standard study of the technology and safety performance evaluation on Emergency Core Cooling System Strainer

Author: Jingmei Zhu[1]

[1] [illegible]

Corresponding Author: 13366836240@163.com

Abstract: Emergency Core Cooling System (ECCS) strainer is the important equipment to ensure the effective and safe operation of ECCS, Containment Spray System (CSS) and the Cavity Injection System (CIS) etc. after PWR NPP accidents. Strainer equipment design technology, test technology and relevant core cooling safety evaluation technology are important design and research work of nuclear power projects, and also one of the important contents of nuclear safety review. In the past decades years, many countries have done a lot of research work in this field, and made great efforts to solve related nuclear safety issues. However, there is still a lack of a complete set of international standards to guide research, design and safety performance evaluation work related to ECCS strainer. Therefore, it is necessary to establish a set of standards covering the performance evaluation of ECCS strainer in accordance with the requirements of nuclear safety regulation, combined with the research results and engineering practices of NPPs owners. The standards can be the basis and reference for the nuclear regulatory authorities and the NPP owners to have unified standards to carry out nuclear safety work, greatly improving work efficiency and reducing unnecessary contradictions in nuclear safety review work.
Based on the research results of the ECCS strainer and its nuclear safety performance evaluation and the nuclear engineering practice research conducted by the author, this paper proposes to establish a set of standards on ECCS strainer which covering debris source workdown, upstream analysis, debris pressure drop test, chemical effect test, downstream effect (in-vessel) test and downstream effect (ex-vessel) analysis, and finally give out preliminary research results.
Keywords: design and performance evaluation of Emergency Core Cooling System Strainer,; debris source; upstream analysis; debris head loss test.

Email address:

13366836240@163.com

Confirm that the work is original and has not been published anywhere else:

YES

119

Application Of Passive Cold Source System In Nuclear Power Plant Design

Authors: KAI WANG[None]; WENBIN ZHAO[None]; XIAOXIA CHEN[None]

Corresponding Author: cnpe@sina.cn

Passive cold source system plays a very important role as one of the main accident prevention and mitigation measures for advanced nuclear power plant. The operation of passive cold source system is based on the passive principle, using the inherent physical properties of the medium to achieve automatic triggering and automatic operation. The passive cold source system can provide at least 24 hours of fully autonomous operation without human intervention. Between 24 hours to 72 hours, mobile equipment and back-up plant water are available on-site to ensure system operation.
At present, three passive systems have been designed in the "Hualong 1" nuclear power unit, passive containment heat removal system (PCS system), reactor cavity water injection cooling system (CIS system) and secondary side passive heat removal system system (PRS system). During the design, in order to meet the requirements of the passive containment heat transfer system, a hot water exchange tank is installed at a high place outside the containment of the reactor, and a valve and pipeline room is set at the bottom of the hot water exchange tank for PCS and PRS related valves and pipes. The water tank of the CIS system is arranged in the containment shell. In order to save the layout space in the shell, the water tank of the CIS system will be merged with the PCS water tank. At the same time, the water tank will be moved up to the top of the dome of the containment shell, and the passive cold source water tank will also be equipped with equipment and modular design.

Email address:

CNPE@SINA.CN

Confirm that the work is original and has not been published anywhere else:

YES

121

Research on heat pipe conduction device of containment dome

Authors: HU Bei[1]; JIAN KANG[None]; SHANSHAN QIU[None]

[1] *China Nuclear Power Engineering Co., Ltd.*

Corresponding Author: hubeicnpe@163.com

Pressurized water reactor (PWR) nuclear power plant containment shell is a tall space, and the containment dome is easy to gather heat and hydrogen, the dome air temperature is much higher than the average air temperature of the containment, continuous high temperature may cause adverse effects on the process equipment and concrete structure, hydrogen is easy to cause hydrogen explosion if there is no stirring or elimination measures. At present, forced convection mixing of dome air is mainly carried out by setting active fans, or passive cooling and air circulation are carried out by setting large volume water tank at the top of the containment vessel.
The dome of pressurized water reactor (PWR) nuclear power plant, as a radioactive containment barrier, needs to resist accident high pressure and radioactive leakage. The thickness of concrete is large, the thermal conductivity is very low, and only relying on the natural heat transfer of concrete can not effectively derive enough heat dissipation.
In this paper, a new passive residual heat extraction system for single-shell reactor is introduced. Heat pipes and enhanced heat transfer devices are installed in the containment dome to improve the heat transfer coefficient of the dome. The thermal conductivity system can passively export the residual heat of the containment vessel to the outdoor air source under different working conditions, and establish a natural ventilation circulation in the dome to relieve heat and hydrogen accumulation. The system has passive safety, the higher the temperature inside the containment, the more heat is removed and the stronger the air circulation. According to calculation, 1000kW heat can be derived under normal operating conditions, and 5000kW heat can be derived under accident conditions, which can be used as a beneficial supplement to the containment ventilation system and waste heat extraction system.

Email address:

hubeicnpe@163.com

Confirm that the work is original and has not been published anywhere else:

YES

122

Material Performance Metrics for Accident Tolerant Fuel Cladding in Pressurised Water Reactors

Author: Maithah Alaleeli[None]

Co-author: Yongsun Yi [1]

[1] *Khalifa University*

Corresponding Author: maithah.alaleeli@hotmail.com

Research conducted at Khalifa University in the UAE aimed to develop evaluation metrics on material performance of Accident Tolerant Fuel (ATF) cladding concepts in Pressurised Water Reactors (PWRs). The aim is to identify and quantify the material properties that affect plant response under Station Blackout (SBO) and Loss-Of-Coolant-Accident (LOCA) conditions. When the cladding is exposed to high-temperature steam, high oxidation resistance was found to significantly reduce the amount of by-product hydrogen but not increase the time to core meltdown This paper reports the extent of influence individual thermo-physical and thermo-mechanical cladding properties have on coping time in comparison to the conventional Zircaloy-4 via severe accident simulations. One result was that material with higher specific heat capacity (or enthalpy) absorbs a given amount of heat from the surroundings faster. It also stores this heat for longer before it is used to increase the cladding temperature. The rate at which this temperature rises was also observed to be slower and therefore delay melting. Additionally, it was found that a material with higher ultimate tensile strength permits the cladding to withstand steep pressure gradients in the fuel gap for longer and in turn delay tube burst. These simulation results are measured against the Zircaloy-4, and each other, in order to establish the weights of the evaluation metrics.

Email address:

100059509@ku.ac.ae

Confirm that the work is original and has not been published anywhere else:

YES

123

Application of the Objective Provision Tree tool for the Safety-security interface assessment

Author: Luca Ammirabile[1]

Co-author: GianLuigi Fiorini [2]

[1] *European Commission*

[2] *Nuclear-21*

Corresponding Author: luca.ammirabile@ec.europa.eu

National nuclear safety authorities and international organizations highlight the importance for enhancing and implementing the harmonization between safety and security approach with the objective of avoiding counterproductive interactions.
Nuclear safety and nuclear security with, for the latter, a special focus on physical protection, have the common purpose to protect the nuclear assets, the public and the environment. The protection is achieved by preventing and mitigating accidental and/or malevolent actions, which can lead, directly or indirectly, to large release of radioactive material.
Many elements or actions serve to enhance both safety and security simultaneously but there are circumstances in which actions or solutions to serve one objective can be antagonistic to the achievement of the other.
This paper discusses and motivates the need for an integrated approach for safety, security and its interfaces. The aim is to ensure that safety and security are dealt with together in a seamless and effective way. The paper suggests the concept of safety/security architecture as a key to addressing and resolving the safety-security integration. Then, the use of the Objective Provision Tree (OPT) tool is suggested to provide the architecture representation and to establish the link between the two domains.
For the security representation, in the logic of the OPT, the notion of "safety function" is replaced with that of "security function" while notions as "challenge / threat", "mechanisms" and "provision" remain fully applicable. A strong analogy remains between the concepts of "Equipment Target Sets" (for security) and "Line of Protection" (for safety).
A pilot application developed within the scope of this paper describes the simplified OPT for safety and security architecture of a hypothetical innovative reactor design. It clearly shows how something that is sufficient and acceptable from safety point of view is not necessarily appropriate or

sufficient from security point of view.
The implementation of safety - security OPT methodology, by allowing the joint and simultaneous analysis (i.e. the integrated analysis) of safety and security concerns, can enable designers and analysts to have a vision of the architecture of the system that is both adequate and comprehensive and that will allow optimizing the whole system while identifying – as requested - any inconsistencies and or conflicts. Complementary requirements can be identified and motivated, which will permit to optimize the whole plant architecture both from safety and security point of view while avoiding counterproductive interactions.
The analysis shall be exhaustive and iterative to consider simultaneously all the plausible plant conditions and all the plausible threats respectively from a safety and security point of view. But this does not affect the fact that the OPT is an instrument that can help responding concretely to meet the targets set by purpose and objective of the study.

Email address:

luca.ammirabile@ec.europa.eu

Confirm that the work is original and has not been published anywhere else:

YES

124

GIF Integrated Safety Assessment Methodology (ISAM) and Guidance for its Implementation for Novel Advanced Reactors

Author: Luca Ammirabile[1]

Co-authors: Tanju Sofu [2]; Paul Gauthé [3]

[1] *European Commission*

[2] *Argonne National Laboratory*

[3] *Commissariat à l'énergie atomique et aux énergies alternatives*

Corresponding Author: luca.ammirabile@ec.europa.eu

A principal focus of the Generation IV International Forum (GIF) Risk and Safety Working Group charter has been the development and demonstration of an integrated methodology that can be used to evaluate and document the safety of Gen IV nuclear systems.
Following its initial mandate on development of the "basis for the safety approach for design and assessment of Generation-IV nuclear systems," the RSWG prepared the Integrated Safety Assessment Methodology (ISAM) to support the entire novel advanced reactor technology development cycle.
The ISAM can be considered as a toolkit leveraged to develop a more detailed understanding of safety related design vulnerabilities and resulting contributions to risk. It includes Qualitative Safety Features Review (QSR), Phenomena Identification and Ranking Tables (PIRT), Objective Provision Trees (OPT), Deterministic and Phenomenological Analyses (DPA), and Probabilistic Safety Assessments (PSA).
Based on this detailed understanding of safety vulnerabilities using these tools, new safety provisions or design improvements can be identified, developed, and implemented, and performance requirements for the items important to safety can be established relatively early in the design cycle.
To help facilitate the use of the methodology, the RSWG also developed a supporting Guidance Document to provide the users with further help with the ISAM implementation. This paper will provide an overview of the ISAM and guidance for its implementation for novel advanced reactors.

Email address:

luca.ammirabile@ec.europa.eu

Confirm that the work is original and has not been published anywhere else:

NO

125

Regulatory challenges and licensing efforts for innovative molten salt reactors

Author: Badawy Elsheikh[1]

[1] *Atomic Energy Authority*

Corresponding Author: badawymel@yahoo.com

Molten salt reactors (MSRs) by nature and by good design have highly desirable safety features. The successful commercial deployment of MSRs depends largely on defining their safety characteristics in a well-coordinated licensing requirements program and how they can build on previous efforts and experience in licensing conventional solid fuel reactors.
This paper is intended to highlight the many challenges facing commercialization of molten liquid reactors, as well as the many perceptions and insights of the licensing and safety activities needed to provide that pathway.

Email address:

badawymel@yahoo.com

Confirm that the work is original and has not been published anywhere else:

yes

126

Natrium Reactor SSC Classification using the Licensing Modernization Project (LMP) Process

Authors: Dennis Henneke[1]; Jonathan Li[1]

[1] *GE Hitachi Nuclear Energy*

Corresponding Author: dennis.henneke@ge.com

The Natrium reactor features a 345MWe sodium-cooled fast reactor that can be optimized for specific markets. Natrium is a pool-type reactor, with a sodium primary system and a molten salt cooling supplying an innovative thermal storage has the potential to boost the system's output to 500MWe of power for more than five and a half hours when needed. This allows for a nuclear design that follows daily electric load changes and helps on peaking opportunities driven by energy fluctuations.

The Natrium design employs passive safety, digital instrumentation and control, and modular fabrication techniques to expedite plant construction. Passive features include use of a Reactor Auxiliary Cooling (RAC) system, Inherent Reactivity Feedback (IRF), and a passive intermediate air cooling (IAC) system. IAC utilizes air cooling and requires the opening of a damper (one per train) to initiate passive cooling. If IAC fails to function, RAC will operate without requiring any SSC operation. IRF ensures the reactor power level is reduced given heat up of the reactor occurs and the gravity driven rod drop does not function. The use of multiple passive features, the inherent characteristics of sodium, and the additional active features results in an overall plant risk that is several orders of magnitude lower than the existing reactor fleet.

The design of the Natrium reactor is supported by the development of a full-scope probabilistic safety assessment (PSA), which is being developed in phases as the design progresses. The PSA is being developed to meet the US Non-Light Water Reactor (NLWR) PRA standard ASME/ANS RA_S-1.4-2021 [1], and supports the Licensing Modernization Project (LMP) process described in NEI-18-04 [2]. This process endorsed by the US NRC in RG 1.233 [3] utilizes the PRA in a risk-informed approach to support Licensing-Basis Event (LBE) selection, SSC Classification and the defense-in-depth (DID) evaluation. The Natrium DID process is applying a rigorous defense line (DL) approach consistent with IAEA SSR-2/1 [4] to strengthen the NEI 18-04 DID process, This enhanced DID process would support application outside of the US market. The results of application of the LMP process is a risk-informed LBE selection supporting an optimized safety analysis, a reduced safety related SSC footprint, and an overall reduced cost for the plant design, construction and operation.

References:
1) ANS Standard
2) NEI 18-04
3) RG 1.233
4) SSR-2/1

Email address:

dennis.henneke@ge.com

Confirm that the work is original and has not been published anywhere else:

Yes

127

Development of Safety Design Criteria and Safety Design Guidelines for Generation IV Sodium-cooled Fast Reactors

Authors: Luca Ammirabile[1]; Paul Gauthé[2]; Satoshi FUTAGAMI[3]; Shigenobu Kubo[4]; Tanju Sofu[5]

[1] *European Commission*

[2] *CEA*

[3] *Japan atomic energy agency*

[4] *Deputy Director Advanced Fast Reactor Cycle System R&D Center, JAEA*

[5] *Argonne National Laboratory*

Corresponding Author: futagami.satoshi@jaea.go.jp

In the framework of the GIF, an effort to develop "Safety Design Criteria (SDC)" for SFR systems was initiated in 2011. For this purpose, an SDC task force (SDC-TF) was formulated in July 2011. The SDC-TF members consist of representatives of CIAE (China), CEA (France), JAEA (Japan), KAERI, KINS (Republic of Korea), IPPE (Russia), ANL, INL, ORNL (United States of America), EC and IAEA. The SFR SDC report was completed in 2013 and distributed to international organizations, namely IAEA, MDEP, NEA/CNRA, and regulatory bodies of the GIF member states with active SFR development programs (China, EC, France, Japan, Korea, Russia and the United States). In 2017, the SDC report was updated based on various comments on general matters e.g. safety approaches for Gen-IV reactor systems, the interface between safety and security and suggestions on specific criteria for dealing with e.g. sodium fire, design basis accident (DBA), and design extension condition (DEC).
In 2016, the SDC-TF completed the first SFR safety design guideline report titled "Safety Design Guidelines on Safety Approach and Design Conditions for Gen-IV Sodium-cooled Fast Reactor Systems". The guidelines are a set of recommendations on how to comply with the SFR SDC and address SFR-specific safety topics by clarifying technical issues and providing design options. The SDG on Safety Approach report was distributed to OECD/NEA's Ad-hoc Group on the Safety of Advanced Reactors (GSAR, currently WGSAR) and the IAEA for review. Leveraging the important and constructive feedback, the TF integrated the resolutions for GSAR and IAEA comments into the report in 2018.

The SDC-TF also developed the second safety design guidelines report, "Safety Design Guidelines on Structures, Systems and Components for Generation IV (Gen-IV) Sodium-cooled Fast Reactor Systems (SDG on SSC)", which provides recommendations in considering the design of structures, systems, and components (SSCs) important to safety and supports practical application of the SDC and the SDG on Safety Approach to the design of safety-related SSCs. The SDG on SSC specifies 14 focal points related to three fundamental systems: (1) reactor core system, (2) coolant system, and (3) containment system. These recommendations on the specific SSCs are developed to clarify the safety requirements for the Gen-IV SFR systems. The SDG on SSC is in the final process after reflecting the feedback from WGSAR and IAEA.
This paper describes the outline of the SDC and SDGs contents and its development background as shown above. These SDC and SDGs refer related IAEA safety standards, such as SSR-2/1 Safety of Nuclear Power Plants: Design, SSG-52 Design of the Reactor Core for Nuclear Power Plants. This paper focuses on both technology neutral aspects, which are common parts between the SDC /SDG and IAEA standards, and SFR specific aspects.

Email address:

futagami.satoshi@jaea.go.jp

Confirm that the work is original and has not been published anywhere else:

YES

128

Changes in PSA Models to Support the Licensing of Advanced Non-Light Water Reactors

Author: Jeffrey Julius[1]

Co-authors: Paul Amico ; Stephen M Hess [1]

[1] *Jensen Hughes*

Corresponding Author: jjulius@jensenhughes.com

The role of PSA in the licensing of nuclear power plants has changed significantly over the last 40-years. For plants licensed in the USA in the 1970's and 1980's, there was no requirement for a PSA as part of the plant's licensing basis. Currently in 2022, a PSA is required as part of risk-informing the licensing basis for new LWRs. For example, the PSA developed for the design certification of the Vogtle 3 & 4 plants in the United States consisted of internal events, internal fire, internal flood, and seismic hazards. The PSA developed for Vogtle 3 & 4 followed the current approach as embodied in the 2009 ASME/ANS PRA Standard and developed initiating events that could result in core damage or large early release of fission products to the environment. For advanced reactors, many of the proposed non-LWR designs include "less radioactive inventory, more stable fuel forms, higher system thermal capacities, and longer thermal constants, as well as passive safety features that rely on natural phenomena." In addition to the material properties changing, the physics associated with the progression to consequential releases has also changed. These changes have resulted in changes to the PSA requirements. For example, NEI 18-04 presents "a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. This guidance document provides one acceptable means for addressing … topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection." This paper describes the changes in the development of PSA models and the various roles that PSA could play in support of licensing advanced, non-light water reactors.

Email address:

jjulius@jensenhughes.com

Confirm that the work is original and has not been published anywhere else:

Yes

129

Thermal hydraulic analysis of a novel concept for a Passive Containment Cooling System

Authors: Muhammad ILYAS[1]; Nasir Hengra[1]

[1] *Pakistan Institute of Engineering and Applied Sciences*

Corresponding Author: milyas323@gmail.com

Reduction in containment pressure of a PWR in the follwing Loss of Coolant Accident (LOCA) is desirable for containment integrity. In this article, a novel Passive Containment Cooling System (PCCS) concept is proposed. In the proposed PCCS design, the containment is divided into two compartments with appropriate volumes. An external heat exchanger (HX) is connected between the two compartments of the containment for heat transfer and pressure reduction in the case of LOCA. This new design of the PCCS provides many advantages over previous designs. A RELAP5 model for a small reactor (998.6 MWth) has been developed to analyse thermal-hydraulic performance of the proposed PCCS design following LOCA under a variety of conditions. The results indicate that partitioning the containment creates a differential pressure that acts as a driving force for flow through the PCCS heat exchanger. Consequently, a decrease in long-term containment pressure is observed. In the case of equal volume fractions in the two containment compartments, the PCCS reduces long-term pressure in the containment by 16.5 percent. Sensitivity analysis was conducted in which the flow area of the PCCS inlet line, the heat transfer area, and the size of the HX tank were varied. As these parameters are increased, the PCCS heat transfer capacity increases. However, the former is less sensitive than the latter two. The performance of the PCCS HX under steady-state conditions was verified through RELAP5 simulations, empirical correlations, and analytical models. The evaluation of heat exchanger parameters by various methods was found in close agreement.

Email address:

milyas323@gmail.com

Confirm that the work is original and has not been published anywhere else:

Yes

130

A proposed approach for assessing multi-unit risk of innovative small modular reactors

Author: Ibrahim Alrammah[1]

[1] *King Abdulaziz City for Science and Technology*

Corresponding Author: iarammah@gmail.com

There is an increasing tendency to consider innovative Small Modular Reactors (SMRs) planned either as single modules in distant sites or as multi-units constructed in a single site, as this option is more practical and resource efficient. Some of these projects are modular reactors with a collection of four to eight reactor modules that share some supporting and auxiliary systems and structures. Multi-unit sites include reactors with essentially separate facilities as well as several that employ

highly integrated and shared support systems. Some have the capability to cross connect the emergency core cooling systems and others have shared control rooms. Though, such sites have probable risks of concurrent multiple plant hazards mainly due to external events.
In the past, safety assessment practices have used methods assuming that a single site with multiple units could be characterised by simply summing up the risk metrics of individual units. This way to assessing safety of a site has some drawbacks as it fully cannot reflect the various and complex interactions that would occur in case of a severe event threatening a multi-unit site. The tendency towards the utilising of a common site to house multiple reactors and supporting units requires the regulators to establish the way to assess the safety of such a site. Therefore, methodological solutions should be proposed in order to transit from a limited safety assessment for a single unit to a holistic safety assessment for a site.
Several nuclear countries have developed their regulatory systems mainly considering large NPPs yet there are significant differences between those typical reactors and innovative SMRs that need to be acknowledged when licensing such designs. In terms of SMR licensing, there are several areas which determine the need for alterations from traditional large NPPs licensing. Evolutionary reactor designs demand special licensing requirements. It is recommended that probabilistic insights should be involved in regulating and licensing novel reactor concepts.
Probabilistic Safety Assessment (PSA) represents a well-established tool to support nuclear regulators with insights on potential accident scenarios of a certain plant design. PSA has gained good experience from application to large Light Water Reactors (LWRs). While for SMRs that grasp increasing attention nowadays, the application of PSA should be refined according to the special characteristics of such concepts.
SMR designs takes place with multiple modules that can share components and structures or located at a single site within a plant. The consideration of multi-unit site risk needs to be emphasised by regulatory authorities. Certain aspects have been examined in a somewhat ad hoc manner yet no integrated approach is present. It is to be noted that conducting PSAs for multi-unit sites based on a single reactor at a time can result in misguided perceptions on the level of risk.
This work presents a holistic framework that integrates deterministic and probabilistic safety assessment approaches in order to evaluate the hazards that could threaten a multi-unit nuclear site. The proposed framework could assist as a roadmap to support the process of using PSA in licensing evolutionary SMRs.

Email address:

irammah@kacst.edu.sa

Confirm that the work is original and has not been published anywhere else:

YES

131

CONCEPTUAL DESIGN OF MITSUBISHI SMALL MODULAR REACTOR

Author: Junichi Takeuchi[None]

Corresponding Author: junichi.takeuchi.vd@nu.mhi.com

Small modular reactors (SMRs) attracts a growing attention due to its capability of providing a solution to the needs of distributed sources for power grid with its enhanced safety features and low capital cost. A concept of MHI Integrated Small Reactor is currently under investigation based on Pressurized Water Reactor technology by Mitsubishi Heavy Industries, Ltd. The main feature of the reactor is its integrated design in which all of primary circuit is built into a reactor vessel and reactor coolant is driven by two-phase flow natural circulation. This simplified reactor concept eliminates several postulated accident scenarios, but creates technical challenges in its reactor design: modelling of two-phase flow and heat transfer characteristics under pressure higher than conventional boiling water reactors. To demonstrate a feasibility of the concept, an experimental program has been in progress to validate its preliminary modellings and analyses. This paper presents the design concept and the characteristics of the reactor as well as an experimental plan for further developments.

Email address:

junichi.takeuchi.vd@nu.mhi.com

Confirm that the work is original and has not been published anywhere else:

YES

132

STATUS OF THE INDEPENDENT VALIDATION OF TRACE CODE FOR SMR SAFETY ANALYSES

Author: Fulvio Mascari[1]

Co-authors: A. Bersano [1]; Brian G. Woods [2]; Jose N. Reyes [3]; Kent Welter [3]; Hideo Nakamura [4]; Francesco D'Auria [5]

[1] *ENEA*

[2] *Henry W. and Janice J. Schuette Chair in Nuclear Engineering and Radiation Health Physics School of Nuclear Science and Engineering*

[3] *NuScale Power*

[4] *Japan Atomic Energy Agency*

[5] *University of Pisa - DESTEC*

Corresponding Author: fulvio.mascari@enea.it

Small Modular Reactors (SMRs) are considered one of the most promising technologies for the near-term development of nuclear power generation. In particular, the integral Pressurized Water Reactors (iPWR) appear to be the closest to market deployment among SMR technologies because they start from the well- established operating large Light Water Reactor (LWR) technology and include evolutionary designs aiming to enhance the plant safety. The lower SMR core power, compared to large reactors, allows to exploit several passive safety features. Deterministic safety analyses conducted by best-estimate thermal-hydraulic system codes demonstrate a fundamental role in designing mitigation strategies and evaluating plant safety. SMRs are generally characterized by some common features of the operating reactors and by other unique features, e.g., containment interactions with the RCS, low pressure phenomena, and phenomena specific to new system components or reactor configurations. Therefore, it is needed to qualify these computational tools for the phenomena characterizing SMR operation and transient conditions. Among the available best-estimate thermal-hydraulic system codes, TRACE (TRAC/RELAP Advanced Computational Engine) is being developed by USNRC to simulate the thermal-hydraulic behavior of operating reactors and advanced designs like SMR. In the past few years an independent validation of a SMR design, led by ENEA, has been carried out in the USNRC CAMP (Code Application and Maintenance Program) framework following the TRACE code development in related areas. The experimental data developed in the OSU-MASLWR facility was used for these validation activities. More recently a numerical scaling analysis was conducted to give insights about the TRACE scaling-up capability against single-phase natural circulation in integral type reactors. The present paper summarizes the TRACE independent validation activities of OSU-MASLWR tests against the natural circulation phenomena inside the integral RPV, RPV/containment coupling phenomena taking place in the mitigation of SBLOCA scenario, and the status of the current code capabilities.

Email address:

fulvio.mascari@enea.it

Confirm that the work is original and has not been published anywhere else:

yes

133

Innovative Passive Safety Features of the HeFASTo Reactor Concept

Authors: Petr Vácha[1]; Roman Koryčanský[2]; Petr Hájek[2]; Jan Klouzal[1]

[1] *UJV Rez, a. s.*

[2] *UJV Rez, a.s.*

Corresponding Author: petr.vacha@ujv.cz

HeFASTo is a concept of an advanced modular reactor based on the gas-cooled fast reactor (GFR) technology with a thermal output of 200 MW and nominal core outlet temperature 900℃. It has been under development by UJV Rez, Czech Republic.

It features many innovative modular solutions in both its design and the approach to the passive safety. It features four key passive safety systems:
- Dedicated decay heat removal system
- Emergency coolant injection system
- System for prolongation of primary blower rundown period
- System for keeping elevated residual pressure in LOCA conditions

In this paper, main features of these passive safety solutions are described, and their compliance with the existing regulatory requirements are discussed.

Email address:

petr.vacha@ujv.cz

Confirm that the work is original and has not been published anywhere else:

YES

134

ARKADIA-Safety, an overall risk assessment simulation tool for risk integrated and performance based decision making

Author: Yasushi OKANO[1]

Co-authors: Masateru Sonehara [2]; Shinya Ishida [2]; Akihiro Uchibori [2]; Masaaki Tanaka [2]; Yasuhiro Enuma [2]; Takashi Takata [3]; Tai Asayama ; Hiroyuki Ohshima [2]

[1] *Japan Atomic Energy Agency*

[2] *JAEA*

[3] *University of Tokyo*

Corresponding Author: okano.yasushi@jaea.go.jp

ARKADIA (Advanced Reactor Knowledge- and AI-aided Design Integration Approach through the whole plant life cycle) is a tool to realize a significant improvement in development process more efficiently by automatic optimization of innovative reactor systems including fast reactors. ARKADIA consists three main systems: KMS (Knowledge Management System), VLS (Virtual plant Life System), and EAS (Enhanced and AI-aided optimization System) with an AI-aided platform as a common fundamental structure. For the application in safety-related analyses and design optimization, a numerical simulation tool, named SPECTRA, to evaluate from transient to severe accidents situations is now under development where fuel/core/plant/containment dynamics are seamlessly simulated in coupling from initiating events to fuel-to-core damage and potentially to containment response against the challenges. Namely, the simulation covers from AOO to DBA or DEC continuously with accounting event-branches and plant condition changes as a dynamic PRA from level 1 to 2. From these features of SPECTRA, potential users, e.g. developers of innovative reactor systems, will be able to proceed safety and design optimization to meet defence-in-depth concept and safety standards/criteria/goals and also to obtain risk-informed, performance-based features into the reactor designs from the beginning of their development process. In this paper, numerical methods and modules, example of results, and future development plan of SPECTRA are presented as well.

Email address:

okano.yasushi@jaea.go.jp

Confirm that the work is original and has not been published anywhere else:

YES

135

An alternative emergency preparedness regulatory framework for small modular reactors and other new technologies

Authors: Eric Schrader[1]; Todd Smith[None]

[1] *USNRC*

Corresponding Author: eric.schrader@nrc.gov

The current regulatory framework for emergency preparedness is a prescriptive regime that focuses on emergency plan maintenance and compliance with pre-determined criteria based on assumptions about large light-water reactor designs. Evolutionary designs are expected to have design features that provide simplified, inherent, passive, or other innovative means to accomplish their safety and security functions. For example, when compared to traditional large light-water reactors, small modular reactor designs are expected to have slower transient response times, and relatively small and slow release of fission products associated with postulated accidents. Emergency preparedness follows a risk-informed, graded approach in which the requirements and criteria are based on the importance of several factors. This suggests that innovative reactor technologies would greatly benefit from a regulatory framework that provides flexibility for considering the various improvements and advancements in design and is performance-based in meeting regulatory requirements.
This paper will discuss the benefits and challenges of shifting from the current prescriptive, compliance based regulatory structure to a performance-based, technology-inclusive, risk-informed, and consequence-oriented regulatory approach to emergency preparedness for small modular reactor, non-light-water reactors, and other new technologies.

Email address:

eric.schrader@nrc.gov

Confirm that the work is original and has not been published anywhere else:

yes

137

Practical application of uncertainty and sensitivity analysis methodologies for the analysis of severe accidents in VVER reactors

Authors: Pavel Kanin[None]; Ivan Melnikov[1]

[1] *NRC "Kurchatov Institute"*

Corresponding Author: kanin_pk@nrcki.ru

Nowadays numerical simulation is an aspect of great importance for safety demonstration of nuclear power plants. An integral best estimate computer code SOCRAT-B1/B2 is applied for safety justification of VVER in Russia.
In accordance with the world's best practices, deterministic analysis of severe accidents should be complemented by uncertainty and sensitivity analyzes of the calculation results. The GAUSS module within brand new TSAR code is developed in NRC "Kurchatov Institute" for this purpose. The module is based on Monte Carlo method with a random sampling of input parameters and provides graphical and tabular representation of analysis results and corresponding statistical measures.
The report presents the results of uncertainty and sensitivity analysis of a severe accident scenario "rupture of the pressurizer surge line with simultaneous station blackout" for VVER-1000 (V-320) implemented by means of code SOCRAT-B1/B2 and GAUSS module.

Acknowledgment
This research is supported by the International Atomic Energy Agency under the Coordinated Research Project award I31033 on Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors, launched in 2019.

Email address:

kanin_pk@nrcki.ru

Confirm that the work is original and has not been published anywhere else:

YES

139

The ASME/ANS Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants

Authors: David Grabaskas[1]; Karl Fleming[2]

[1] *Argonne National Laboratory*

[2] *KNF Consulting Services LLC*

Corresponding Author: dgrabaskas@anl.gov

Due to growing interest in non-light water reactor (NLWR) concepts, the predecessor to the American Nuclear Society (ANS)/American Society of Mechanical Engineers (ASME) Joint Committee on Nuclear Risk Management (JCNRM) initiated development of a NLWR probabilistic risk assessment (PRA) standard in 2006. The developed standard, released for trial use in 2013, was the first integral PRA standard created by the JCNRM, containing requirements ranging from the development of initiating events to the analysis of offsite radiological consequence. Based on feedback from pilot applications of the trial standard, a revised version of the standard was developed and balloted in 2020, with unanimous passage by the JCNRM and formal publication as an American National Standards Institute (ANSI) standard in 2021. This article provides an overview of the standard and its development process. In addition, the article reviews the latest status of U.S. Nuclear Regulatory Commission endorsement efforts through Regulatory Guide 1.247.

Email address:

dgrabaskas@anl.gov

Confirm that the work is original and has not been published anywhere else:

YES

140

BELGIAN APPROACH FOR LICENSING NEW INNOVATIVE REACTORS

Authors: François Henry[1]; Simon Coenen[2]; Irina Sanda[3]

[1] *Bel V*

[2] *FANC - Federal Agency for Nuclear Control*

[3] *SCK CEN*

Corresponding Author: simon.coenen@fanc.fgov.be

BELGIAN APPROACH FOR LICENSING NEW INNOVATIVE REACTORS
Pre-Licensing experience of the MYRRHA Research Reactor
S. Coenen , F. Henry , I. Sanda
Belgium

Abstract
The Belgian nuclear research centre SCK CEN is preparing to build MYRRHA, a heavy metal cooled fast reactor as part of an accelerator driven system (ADS). Due to the complexity of the facility and to build up sufficient knowledge and expertise for such an innovative design, the SCK CEN, FANC and Bel V launched a pre-licensing project as a preparatory phase for the licensing of the facility. The pre-licensing process allows for early interaction with the future operator to converge to a design that meets all expectations of the safety authorities. regarding safety, security and safeguards. The process also allows the development of specific regulation if needed, as well as to extend an independent knowledge base for all parties. For example, the use of lead-bismuth eutectic as a coolant leads to challenging radiation protection and safety issues (e.g. production of Po-210 and other elements of high radiotoxicity). The use of an opaque coolant medium introduces specific requirements with respect to corrosion which will add further safety, security and safeguards concerns for the regulatory body.
The pre-licensing process allows for an early interaction between the regulator and the future licensee with the objective to converge to a final design concept that meets all expectations of the safety authorities. The pre-licensing ends with a "final conclusion" from the regulatory body expressing the level of compliance of the design with the regulatory framework and the level of how all safety, security and safeguards issues are addressed, etc. This "final conclusion" is not binding and does not guarantee a successful licensing process, but aims to identify and resolve any issues for the licensing process.
The full paper presents in detail the methodology, scope and objectives of the pre-licensing process in general and illustrates its usefulness for both operator and regulator using specific results from its application to the development of MYRRHA.
llustrates its usefulness for both operator and regulator using specific results from its application to the development of MYRRHA.

Email address:

Simon.Coenen@fanc.fgov.be

Confirm that the work is original and has not been published anywhere else:

YES

141

Interest of functional analyses for internal hazards at the beginning of a project

Author: Hélène FLAMENT[None]

Corresponding Author: helene.flament@edvance.fr

In the frame of design and protection for internal hazards, the associated analyses are identified as a high stake for the new French reactor EPR2. The major stake is to prove that sufficient separation exist between components which failure may trigger an event (Anticipated Operational Occurence or Design Basis Accident) and components required to perform the safety functions following such an event. Indeed, the internal hazard analyses are strongly linked to layout in the buildings and late modifications aiming at protecting the components are difficult to manage, in particular when construction is started. Generally, hazard analyses verification are performed in a late phase of a project to take into account the most complete and finalized input from layout. In order to secure the design and layout as early as possible, functional analyses are performed at the beginning of the basic design phase to identify the possible common cause failures combining the initiation of an event with the loss of safety means required to mitigate this event.
In EPR2, preliminary analyses are performed during the first system implementation phase in the mock-up. It means that, whenever a proposal of system implementation in the mock-up is presented, a functional internal hazard analysis is performed. It is particularly efficient when assessing local consequences of hazards such as those of High Energy Pipe Break (HEPB) or missile. The challenge at the beginning of a project is to perform analyses without input data stabilized. Indeed, only rough information about systems is available (main design choices but not detailed identification of all the components required to perform a safety function) and also no demonstration that the civil structures can withstand the loads. Then some assumptions have to be made in order to compensate for the lack of data. All these preliminary conclusions have to be confirmed later on, when validated data is available.
These analyses necessitate to compile lots of input and output data. With the objective of performing comprehensive and traceable analyses with limitation of human error, a dedicated tool named SFDB (SaFety DataBase) is used. It allows defining volumes affected by the internal hazard, identifying all the components located in this volume (which failure can initiate an AOO or DBA or jeopardize an accident mitigation means) and then confirming that possible common cause failure remain acceptable according to the safety analysis rules.

Email address:

helene.flament@edvance.fr

Confirm that the work is original and has not been published anywhere else:

Yes

142

The US DOE Advanced Sensor and Instrumentation program

Author: Pattrick Calderoni[1]

[1] *INL*

Corresponding Author: pattrick.calderoni@inl.gov

In 2011, the Department of Energy’s Office of Nuclear Energy (DOE-NE) initiated the Nuclear Energy Enabling Technologies (NEET) initiative to conduct research, development, and demonstration (RD&D) in crosscutting technologies that directly support current reactors and enable the development of new and advanced reactor designs and fuel cycle technologies. The Advanced Sensors and Instrumentation (ASI) program is the element of NEET dedicated to Instrumentation and Control (I&C) technology.

The ASI Program has spurred innovation in the measurement science field by funding research to advance the nuclear industry’s monitoring and control capability. These capabilities are crucial in developing research solutions that enable reduced costs, improved efficiencies, and increased safety for advanced reactors operations. They also serve a vital role in Materials Test Reactors (MTR) to measure environmental conditions of irradiation experiments and to monitor aspects of advanced fuel and materials behavior that in turn contribute to accelerate deployment of advanced nuclear systems.

The ASI program is coordinated with DOE-NE's other research and development (R&D) programs to ensure that developed technologies and capabilities are part of an integrated investment strategy aimed at improving the safety, reliability, and competitiveness of domestic nuclear technologies. The ASI program provides the crosscutting technologies research in four main areas:

1. Sensors and Instrumentation. Research and develop reliable and cost-effective instruments to provide real-time, accurate, and high-resolution measurements of the performance of existing and advanced reactors' core and plant systems.

2. Advanced Control Systems. Research and develop real-time control of plant or experimentation process variables to enhance performance and reduce operation and maintenance (O&M) costs through advanced risk-informed approaches to monitoring and control.

3. Nuclear Plant Communication. Research and develop a resilient, secure, and real-time transmission of sufficient data enabling online monitoring, advanced control strategies, and big data analytics.

4. Big Data Analytics, Machine Learning, and Artificial Intelligence: Research and develop machine learning and artificial intelligence capabilities to enable autonomous operations and maintenance by design using heterogeneous and unstructured data.

This contribution highlights the accomplishments of the program research activities in recent years in all four research areas. Technical outcomes are discussed, with focus on the development of innovative sensor technologies for in-core instrumentation and their demonstration in irradiation experiments to reach operational conditions relevant to advanced reactors (including combined temperature and neutron flux). This includes demonstration test of mature technologies aimed at extending operational boundaries (for example, thermocouples and self-powered neutron detectors) and proof of feasibility of lower readiness technologies for nuclear applications, such as optical fiber and acoustic sensors. The ASI program engagement with other DOE programs and industrial partners is also discussed to emphasize the role of innovative I&C solutions to the deployment of advanced reactors, with focus on transportable, small footprint systems with less than 20 MWth energy output (microreactors).

Email address:

pattrick.calderoni@inl.gov

Confirm that the work is original and has not been published anywhere else:

yes

144

An Integrated Design Approach to Address Safety of the Westinghouse LFR: an Innovative Pool-Type, Liquid Lead-Cooled Fast Reactor

Authors: Cory Stansbury[1]; Paolo Ferroni[1]; Jun Liao[2]; Sung Jin Lee[3]; Emre Tatli[1]; Michael Epstein[4]

[1] *Westinghouse Electric Company LLC*

[2] *Westinghouse Electric Company*

[3] *Fauske & Associates, LLC*

[4] *Fauske & Associates*

Corresponding Author: stansbca@westinghouse.com

The Westinghouse Lead-Cooled Fast Reactor (LFR) is a highly compact, pool type reactor with a core power of 950 MWt and net electrical output of 450 MWe produced through use of a supercritical steam (sH2O) power conversion system with an air-cooled condenser and integral energy storage system. Westinghouse has targeted IAEA Passive Category B safety, as it is felt that successfully achieving this goal would result in a level of robust simplicity which also correlates to a widely deployable, rugged, and economic power plant for customers around the world. Pure lead-coolant technology is supportive of this goal and Westinghouse selected this coolant and design direction after consideration of many competing technology types. The benefits of this technology were predicted at the time and continue to show an ability to simplify reactor safety systems, enable optimized secondary efficiency, enhance fuel burn-up, reduce component count (especially in safety), and (through these improvements) drive towards a competitive cost of electricity. Further innovations in reactor configuration and heat exchanger design have resulted in a uniquely power dense reactor system and plant design which can transfer heat from the primary fluid circuit to secondary side directly in micro-channel heat exchangers while elegantly addressing concerns associated with secondary breaks; strongly supporting the goal of eliminating a large, high-pressure containment and safety-grade instrumentation and controls with their associated active component actuation. This paper describes the key design features which work in concert to deliver the targeted reactor system power density, reduced complexity, and passive safety. These are considered to address major safety events and all accident categories; specifically prioritizing loss of offsite power, in-vessel break of secondary pressure boundary, loss of reactivity control, and spent fuel safety. As appropriate, heat removal, reactivity control, radionuclide retention, enabling design decisions, and features are discussed. Lastly, the comprehensive suite of supporting design, simulation, and modeling tools are described along with ongoing testing work used to demonstrate performance of key LFR systems and components and inform and qualify these tools. This includes full height testing of passive decay heat removal, a highly instrumented lead loop for testing of primary and secondary components, and testing of primary heat exchanger failure and lead freezing phenomena.

Email address:

stansbca@westinghouse.com

Confirm that the work is original and has not been published anywhere else:

YES

145

Multi-Module Probabilistic Safety Assessment (PSA) for the PRISM Sodium-Cooled Fast Reactor

Author: Dennis Henneke[1]

[1] *GE Hitachi Nuclear Energy*

Corresponding Author: dennis.henneke@ge.com

This paper describes the development of a Multi-Unit PSA (MUPSA) for a PRISM power block with two identical reactor units. The PRISM is a pool-type, metal-fueled, small modular Sodium Fast Reactor (SFR). PRISM employs passive safety, digital instrumentation and control, and modular fabrication techniques to expedite plant construction. The analyzed PRISM design includes a single turbine generator fed from the two reactors, through a common steam inlet to the turbine generator. The goal of this MUPSA is to develop an analysis of events that involve more than one reactor on a multi-unit site for a non-LWR PSA. This analysis is documented Annex VI of the IAEA Safety Report 96 on MUPSA [Ref. 1].

The MUPSA methodology developed for the PSA focuses on the extension of a single reactor PSA model to account for interaction with other reactor units or generating units at the same power plant. The methodology used is patterned after the method outlined in IAEA guidance [Ref. 2] used

as input to SR 96, and is intended to satisfy the requirements in the ASME/ANS Non-LWR PSA standard [Ref. 3]. The PRISM MUPSA includes the performance of similar steps as performed in the single unit PSA, such as the initiating event analysis, accident sequence analysis, system modeling, etc. However, the main difference for the PRISM reactor PSA compared to a traditional LWR PSA is the performance of MU Common Cause Failure (CCF) analysis for passive reliability features, such as multi-unit failure of the reactor vessel auxiliary cooling systems (RVACS).

The main result of the MUPSA is that, given a single-unit PSA is completed, the performance of a MUPSA can performed without adding significant time to the overall PSA effort. For a reactor utilizing passive features, specialized models are needed for the analysis of passive reliability CCF which can be based on the single unit passive reliability modeling. In the case of RVACS MU CCF, the use of Monte Carlo analysis is expanded using a similar model as used in the single-unit PSA, taking into account an estimate of the dependency between units for the reliability factors modeled in the passible reliability, such as flow area, air temperature, inlet pressure, friction, surface emissivity, etc. The quantitative PSA results for the plant internal events analyzed showed a low likelihood of MU releases due to the overall low CCF for the passive reliability features in the PRISM plant. However, when external hazards are considered such as seismic events, the PRISM results will likely be similar to LWRs used as the pilot application for the MUPSA Guidance [Ref 1], where there is a high probability of two units releasing significant radioactivity given a single unit releases significant radioactivity.
References:
1) IAEA Safety Report 96, Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units, 2019,
2) International Atomic Energy Agency, "Technical Approach to Multi-Unit Site Probabilistic Safety Assessment (MUPSA)," 2014.
3) ASME/ANS Joint Committee on Nuclear Risk Management, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," 2013.

Email address:

dennis.henneke@ge.com

Confirm that the work is original and has not been published anywhere else:

Yes

146

Uncertainty and Sensitivity Analysis of Severe Accidents Simulations at VVER in the Framework of the IAEA CRP I31033

Author: Nikolai Ryzhov[1]

Co-authors: V.N. Semenov [1]; R.V. Chalyy [1]; Khaled Naffaa [2]; Oleksandr Balashevskyi [3]; Anna Nikolaeva [4]; Stanislav Pantyushin [4]; Alexander Lityshev [4]; Dmitry Gasparov [4]; Valeriy Astahov [4]; Pavel Kanin ; Ivan Melnikov [5]; Tatjana JEVREMOVIC [6]; H.U. Rheman [6]; Fulvio Mascari [7]; A. Bersano [8]; FABRIZIO Gabrielli [9]; David Luxat [10]; L. I. Albright [11]; Javier Ortiz [12]; Kwang-Il Ahn [13]; Sergei Petoukhov [14]; J. M. García [15]

[1] *IBRAE RAN*

[2] *NNEGC "ENERGOATOM"*

[3] *SE "Scientific and Technical Center" of SE NNEGC "Energoatom"*

[4] *OKB Gidropress JSC*

[5] *NRC "Kurchatov Institute"*

[6] *IAEA*

[7] *ENEA*

[8] *ENEA, Italy*

[9] *Karlsruhe Institute of Technology*

[10] *Sandia National Laboratories*

[11] *SNL, USA*

[12] *ININ*

[13] *Korea Atomic Energy Research Institute (KAERI)*

[14] *Canadian Nuclear Laboratories Ltd.*

[15] *CNEA, Argentina*

Corresponding Author: rni@ibrae.ac.ru

The International Atomic Energy Agency (IAEA) launched in 2019 the Cooperative Research Project (CRP) I31033 entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors" [1]. The main goal is to advance the understanding and characterization of sources of uncertainty and investigate their effects on the key figure-of-merits (FOMs) of the severe accident codes predictions in water cooled reactors (WCRs). Having this in mind, within the five-year duration of the CRP, 22 organizations representing 18 Member States have been developing and evaluating calculation platforms based on the currently-available severe accident codes and relevant uncertainty tools for uncertainty and sensitivity (U&S) analyses of severe accident scenarios in different WCRs, i.e. PWR, BWR, CANDU, VVER, and SMR. In the CRP framework, the U&S analysis of the QUENCH-06 experiment [2] has also been performed to give relevant insights.
The VVER group consists of four institutions from two Member States, one from Ukraine NNEGC "ENERGOATOM" and three from Russian Federation OKB GIDROPRESS, IBRAE RAN, and NRC "KURCHATOV INSTITUTE". Two severe accident codes are used MELCOR 1.8.5 and SOCRAT [3] together with in-house developed uncertainty tools. Each participant has selected a specific severe accident to investigate and developed a list of input uncertain parameters. NNEGC "ENERGOATOM" simulates a station blackout (SBO) at the first unit of South-Ukrainian NPP. OKB GIDROPRESS investigates a SBO scenario with account for accident management actions consisting in the opening of 3 PORVs 2 h since the initiating event. IBRAE RAN has opted for LB LOCA (cold leg rupture) that happens simultaneously with unmitigated SBO. NRC "KURCHATOV INSTITUTE" analyze LB LOCA (rupture of the pressurizer surge line) with simultaneous SBO. One paid attention to the justification of PDFs of uncertain input parameters giving preference to literary sources or technical documentation rather than expert judgment. The U&S analysis is focused on in-vessel stage of severe accident progression and key FOMs at VVER-1000 reactor. Determined FOMs include hydrogen production, times of key events and other reactor and containment parameters. By now these institutions have obtained preliminary results on U&S analysis of selected FOMs and uncertainty associated with mesh refinement. This paper presents the description of the applied methodologies for U&S analysis, base case calculation results, and U&S analysis results performed by the group.

[1] https://www.iaea.org/projects/crp/i31033
[2] Sepold, L., et al., 2004, Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45), Report FZKA-6664, Forschungszentrum Karlsruhe.
[3] Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, IAEA-TECDOC-1872, Vienna, 2019.

Email address:

rni@ibrae.ac.ru

Confirm that the work is original and has not been published anywhere else:

YES

147

Nuclear floating power unit: ensuring safety during transportation

Author: Daria Doronkova[None]

Co-authors: Vladimir Malev ; Nadezhda Salnikova [1]; Igor Shmelev ; Konstantin Veshnyakov

[1] *JSC "Afrikantov OKBM"*

Corresponding Author: doronkova@okbm.nnov.ru

The existence of the stage 'transportation' is a feature of the life cycle of floating power units, when the object moves between the construction site and the operation site. The projects of both Russian and foreign developers assume that the floating power unit can be transported with a reactor plant loaded with fuel. One of the issues that needs to be taken into account when developing an FPU is the issue of transportation and safety requirements at this stage of the life cycle. This issue is actively discussed within the IAEA working groups. To date, special norms and requirements for floating power units exist only in Russia, and are absent at the international level.
The stage of transportation with fuel inside is also assumed in the life cycle of packages; there is an extensive international regulatory framework that regulates the conditions for the design, testing and movement of packages.
One of the approaches to justify the safety of transportation of floating power units is a comparative analysis with the requirements for the safety of transportation of packages. Given the fundamental differences between the objects, a direct assessment of the compliance of floating power units with the requirements for packages is methodologically incorrect.
This report will consider the option of transporting a floating nuclear power unit loaded with nuclear fuel, show the difference in safety assessment methodologies for nuclear floating power units and packages, and propose approaches to confirming the safety of transportation coupled with the requirements for packages.
The conclusion of the report will also include proposals for approaches to international regulation of the transportation of floating power units.

Email address:

doronkova@okbm.nnov.ru

Confirm that the work is original and has not been published anywhere else:

Yes

148

Experimental and computational research of the containment passive emergency pressure decrease system in the floating NPP with reactor KLT-40S and universal nuclear-powered icebreaker with reactor RITM-200

Author: Oleg Tyurikov[None]

Co-authors: Mikhail Kamnev [1]; Akhmir Hizbullin

[1] *JSC "Afrikantov OKBM"*

Corresponding Author: tyurikov@okbm.nnov.ru

JSC "Afrikantov OKBM" specialists have developed the containment emergency pressure decrease system (EPDS), which is based on natural circulation and does not require additional power sources. EPDS of the KLT-40S floating nuclear power plant (NPP) has been studied at the large-scale experimental SPOT ZO test facility (hereinafter the SPOT). The containment of SPOT test facility has a volume of ~59 m3 and height of about 8 m. Passive system cooling circuit was modeled in full scale. The heat exchanger-condenser (HX-C) of the system is located in the containment model. Water evaporation tank has a volume of 25 m3. The relation of system power in range of 120 – 600 kW as a function of containment pressure for various partial air pressures in range of 40 – 170 kPa (abs.) was obtained. Various operation modes of the passive system in case of primary coolant leakage into the containment were studied. As a result of experimental study a validation of KUPOL-MT v 1.0 code (hereinafter the KUPOL-MT) has been performed. KUPOL-MT is a lumped parameter code (LP-code) and is used for analysis of heat and mass transfer inside the containment volume of small reactor plants (RP). Numerical simulations were performed using the KUPOL-MT code to substantiate safety of the floating NPP with KLT-40S RP and universal nuclear-powered icebreaker with RITM-200 RP in accidents of type loss-of-coolant accident (LOCA).

Email address:

tyurikov@okbm.nnov.ru

Confirm that the work is original and has not been published anywhere else:

YES

149

Integrated design analyses of beyond-design-basis accidents at VVER-1200, including fuel severe damage

Authors: Daria Buzykina[None]; Samokhin Aleksei[1]; Nadezhda Kozlova[1]; Roman Fuks[1]

[1] *SEC NRS*

Corresponding Author: asamokhin@secnrs.ru

A brief description and results of a pilot application of a VVER-1200 simulator intended for end-to-end integral simulation of emergency processes in VVER-1200 reactors from the initial event to the processes of nuclear fuel severe damage, including simulation the internal and external stages of the accident, as well as for simulation the behavior of the environment in the containment, are encapsulated in the report.
The NPP simulator provides simulation of the main systems of a VVER-1200 NPP power unit, including innovative passive safety systems. The calculation models of the simulator are based on simulator technologies. It makes possible to carry out calculation analysis of beyond-design-basis accidents in a reactor facility, both in real time and in an accelerated mode (5 - 7 times faster than real time). This makes it possible to use this simulator both for assessing the effectiveness of actions provided in the accident management procedures, and for predicting the development of an emergency in the information and analytical center of Rostechnadzor, as a tool for performing quick assessment and forecast calculations.
Thus, a universal calculated tool has been developed that allows performing independent expert evaluation calculations as part of justification review of safety and predictive calculations in conditions of emergency training.

Email address:

asamokhin@secnrs.ru

Confirm that the work is original and has not been published anywhere else:

Yes

150

Evaluation of safety margins to eliminate the cliff edge effect

Authors: Ivan MELNIKOV[1]; Pavel Kanin[1]

[1] *NRC "Kurchatov Institute"*

Corresponding Author: melnikov_ia@nrcki.ru

Development and justification of measures to eliminate the cliff edge effects occurrence is one of the supervisory authority requirements for a nuclear power plant design. The presence of sufficient safety margin allows to analyze the cliff edge effect probability and evaluate the effectiveness of measures aimed at the avoidance of it and large or early radioactive release. To prevent damage to the last safety barrier and release of radioactive fission products into the environment, power units with VVER gen 3+ are equipped with core catcher.
This work is devoted to the exploration of the safety margin with the CC operation during the accident with fuel melting, by means of mathematical simulation for AES-2006 (VVER-1200) design. The safety margin justification is performed considering the results uncertainty, which is obtained based on statistical analysis of variant calculations.

Email address:

Melnikov_IA@nrcki.ru

Confirm that the work is original and has not been published anywhere else:

YES

151

Uncertainty and Sensitivity Analysis of a CANDU 6 Plant by Means of Severe Accident Codes in the Framework of the IAEA CRP I31033: Preliminary Results

Authors: A. Bersano[1]; Fulvio Mascari[2]; S.M. Petoukhov[3]; Sergei Petoukhov[4]; D. Luxat[5]; FABRIZIO Gabrielli[6]; H.U. Rheman[7]; J. M. García[8]; Javier Ortiz[9]; K.I. Ahn[10]; L. I. Albright[11]; Nikolai Ryzhov[12]; Tatjana JEVREMOVIC[7]

[1] *ENEA, Italy*

[2] *ENEA*

[3] *CNL*

[4] *Canadian Nuclear Laboratories Ltd.*

[5] *SNL*

[6] *Karlsruhe Institute of Technology*

[7] *IAEA*

[8] *CNEA, Argentina*

[9] *ININ*

[10] *Kaeri*

[11] *SNL, USA*

[12] *IBRAE RAN*

THIS PAPER WAS RECEIVED OFFLINE
The International Atomic Energy Agency (IAEA) launched in 2019 the Cooperative Research Program (CRP) I31033 entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors" [1], with the main goal to advance the understanding and characterization of sources of uncertainty and investigate their effects on the key figure-of-merits (FOMs) of severe accident code predictions in water cooled reactors (WCRs). Within the 5-year duration of the CRP, 22 organizations representing 18 Member States have been developing and evaluating calculation platforms based on severe accidents codes and relevant uncertainty tools or uncertainty and sensitivity (U&S) analyses of severe accidents scenarios in different WCRs, i.e. PWR, BWR, CANDU, VVER, and SMR. In the CRP framework, the U&S analysis of the QUENCH-06 experiment [2] has also been performed to give relevant insights.

With respect to the pressurized heavy water reactor CANDU (CANada Deuterium Uranium) analyses, three organizations in three Member States are participating in this CRP task. The overall goal is to share own methods for the U&S analysis and relevant results, as shown in Table 1. All three CANDU group participants analyzed the same accident scenario (station blackout) in a generic CANDU 6 station.

PLEASE REFER TO TABLE (enclosed)
The key code input parameters, which might affect the defined FOMs, including boundary and initial conditions, material properties and code correlations, were identified after a careful screening process by each institution: 12 key parameters for KAERI, 30 key parameters for UPB, and 26 key parameters for CNL. The corresponding probability distributions functions and ranges for these key parameters were assigned by the respective organizations based on the relevant code manuals, literature survey, engineering judgment, and parametric sensitivity analysis wherever necessary. The studies were carried out separately by each organization including: i) plant modeling and nodalization, ii) simulation of the base case, iii) calculation of the U&S through coupling of the relevant severe accident codes with the corresponding uncertainty quantification tools, iv) estimation of resultant FOMs' range, and v) sensitivtitiy analysis of each input to the relevant FOMs. From the results of these analysis general conclusions were drawn.
This paper presents preliminary results of the U&S analyses performed by the CANDU group within the IAEA CRP and discusses all relevant insights in view of the application of the U&S methodologies for severe accident analyses in CANDU.

Email address:

sergei.petoukhov@cnl.ca

Confirm that the work is original and has not been published anywhere else:

Yes

152

CAREM25: an Integral Methodological Approach to Coherently Internalize Defence in Depth in the Design Process

Authors: David Quiroga[1]; Marcelo Gimenez[1]; Juan Matías García[1]; Mariela Stella Grinberg[1]; Pablo Zanocco[1]

[1] *CNEA*

Corresponding Author: gimenez@cab.cnea.gov.ar

This paper discusses the methodological approach used in CAREM25 design, to establish clear and coherent design criteria, requirements and specifications from the nuclear safety technical area. Defence in Depth (DiD) is the basis for this process.
Criteria were defined for DiD internalization in the design taking into account CAREM25 general design characteristics, to properly and in a balanced way prevent, control and mitigate the postulated initiating events. WENRA RHWG [1] proposal for DiD levels definition was adopted to consider Multiple Failures Events (Level 3b) in order to prevent core damage. A strategy was defined for each one of the levels; two stages are considered for Level 3a and 3b. For Levels 3a and b, the first one was accomplished by means of passive safety systems. The second stage, for Level 3a, was implemented by active systems to achieve the final safe state and for Level 3b by simple system with external water supply.
Subsequently, Low Level Safety Functions (LLSFs) including monitoring ones were identified for each one of the DiD levels and stages, starting from the Fundamental Safety Functions, by means of attributes. Safety functional groups (SFGs) -set of structures, systems and components (SSCs) that fulfill those functions- were defined for each one of the LLSFs. Afterward, safety classification process was executed. A methodology developed in harmony with IEC-61226 [2] and IAEA SSG-30 was applied. Criteria for safety categories allocation to LLSFs and classes to SFGs/SSCs were established, based on the strategy defined to internalize DiD in CAREM25 design. Three categories for LLSFs and classes for SFGs/SSCs were defined, with rules for class reduction that were applied for example, to

SSC that constitute the Diverse Line of Protection (Level 3b). As part of this process, deterministic and probabilistic evaluations were also considered to support the evaluation of the relative importance of SSCs.
Finally and by setting deterministic and probabilistic design acceptance criteria for each DiD Level, systems were evaluated to verify that the assigned functions were properly fulfilled. In a complementary way and by means of PSA and DSA, design feedback was provided for Levels 2 and 3 systems; the Chemical and Volume Control System was proposed to control LOHS and the whole postulated spectrum of LOCAs within level 2, and a RPV depressurization system and others to cope with Multiple Failures Events (Level 3b).
A pyramidal set of technical documentation was implemented. IAEA safety standards were taken as reference to carry out the whole process.
It is also important to mention that this approach, based on a consistent DiD internalization, has provided a clear road map to support the design of systems important to safety and their balanced integration into the plant. This has considerably facilitated engineering development and regulatory licensing processes.
[1] Safety of new NPP designs, Study by Reactor Harmonization Working Group, Western, European Nuclear Regulators Association (WENRA RHWG), August 2013.
[2] Nuclear power plants – Instrumentation and control systems important to safety – Classification of instrumentation and control functions, International Standard IEC 61226, Third Edition, 2009-07.

Email address:

gimenez@cab.cnea.gov.ar

Confirm that the work is original and has not been published anywhere else:

YES

153

An Innovative Methodology for Designing and Regulating Small and Modular Reactors

Author: Noreddine Mesmous[None]

Corresponding Author: noreddine.mesmous@cnsc-ccsn.gc.ca

An Innovative Methodology for Designing and Regulating Small and Modular Reactors

Abstract of the technical paper presented at:
IAEA: International Conference on Topical Issues in Nuclear Installation Safety: Strengthening Safety of Evolutionary and Innovative Reactor Designs
October 18–21, 2022

Prepared by:
Noreddine Mesmous
Canadian Nuclear Safety Commission

Abstract:

The Canadian Nuclear Safety Commission (CNSC) has a risk-informed regulatory framework that allows for flexibility and fosters innovation while ensuring that nuclear safety is maintained. CNSC requirements allow applicants/licensees to put forward a case to demonstrate that the intent of a requirement is addressed by other means and demonstrated with supportable evidence.

With the purpose of safe, effective, and efficient design of Small Modular Reactors (SMRs), this innovative risk-informed, consequences-acceptable, and technology-neutral methodology clarifies the application of the practical elimination concept by focusing on strengthening the implementation of defence-in-depth.

In addition, with the objective of addressing regulatory uncertainties, this methodology enables regulators, designers, proponents, and licensees in using the graded approach where the regulatory requirements and guidance are applied in a graded manner, commensurate with the risk posed by the regulated activity.

This methodology is a paradigm shift opportunity to developing SMRs with implementing more safety early in the design stage.

Key words:
Small Modular Reactor, practical elimination, design extension condition, source term, safety function

Email address:

noreddine.mesmous@cnsc-ccsn.gc.ca

Confirm that the work is original and has not been published anywhere else:

YES

154

Calculation of the Neutron Parameters for the Accelerator-Driven Subcritical Reactor using Pb and Pb-Bi mixtures as both Target and Coolant

Author: Quoc Dung Tran[None]

Co-author: Minh Tien Tran

Corresponding Author: trandungquoc@gmail.com

In this paper we present some recent results regarding the use of Pb (lead) and Pb-Bi mixtures as (p,n) reaction targets and coolant in the Accelerator Driven Subcritical Reactor. The accelerator-driven subcritical reactor (ADSR) is simulated based on structure of the TRIGA-Mark II reactor by MCNPX program. A proton beam is accelerated and interacts on the lead and the target mixture of Molten Pb-Bi. Three cases are considered here: firstly, solid lead is referred to as spallation neutron target and water as the coolant; secondly, molten lead is considered both as a target and as a coolant; thirdly, mixture of Molten Pb-Bi lead is considered both as a target and as a coolant. The proton beam interacts on the targets with various energy levels from 0.115 GeV to 2.0 GeV. The neutron parameters were calculated, including the neutron yields, the effective neutron multiplication factors keff, the distributions of the neutron fluxes along the height and the radial of ADSR with Pb and molten Pb-Bi targets in the U-Th mixed and UZrH fuels.
The study results have shown that, in the ADSR system molten Pb (lead) and Pb-Bi should be considered both as targets and as a coolant. The possibility of using molten lead and Pb-Bi with fuel mixtures containing thorium and uranium is a very promising ADSR system. The results also show that the proportions of uranium in the mixture needs to be carefully calculated so that the ADSR generates positive energy and remains subcritical.

Email address:

trandungquoc@gmail.com

Confirm that the work is original and has not been published anywhere else:

Yes

155

Design measures aimed at eliminating cliff-edge effects as a necessary condition for effectiveness of plant defence-in depth

Authors: Artur Lyubarskiy[1]; Gennady Ershov[2]; Mikhail Bykov[3]

[1] *JSC Atomenegoproekt, Moscow*

[2] *JSC Atomproekt, St. Petersburg*

[3] *OKB GIDROPRESS*

Corresponding Author: lyubarskiy_av@aep.ru

The article analyzes the content of the of “cliff edge effect” term.
This term presents either in IAEA Safety Standards and in various international and national documents (including Russian regulatory documents). Cliff-edge effect is defined usually as an instance of severely abnormal NPP behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter.
The article substantiates the close connection of the measures taken in plant design aimed at eliminating the cliff-edge effect with ensuring the overall effectiveness of plant defence-in depth.
The article shows that cliff-edge effects requiring the adoption of design measures to protect against them can manifest themselves in various initial plant states – both in operational states and accident conditions.
The article considers deterministic and probabilistic aspects in the interpretation of the cliff edge effect. It is shown that the provisions of the IAEA safety standards allow us to conclude that both situations with real consequences and situations when the risk of undesirable consequences increases, including situations when the actual occurrence of undesirable consequences is significantly delayed in time, can be regarded as a manifestation of the cliff-edge effect.
The concept of the effectiveness of the specific defence-in-depth level is given and it is justified that situations when a small change in the NPP parameter leads to a loss of effectiveness of two or more levels of defense-in-depth or to a significant change in the magnitude of radiation exposure to personnel, the population, and the environment, as well as situations where the probability of such consequences significantly increases should be considered as cliff-edge effect.
The paper suggests interpretations of the concepts of “significant negative change in plant status”, “parameter”, “small deviation of plant parameter” (for both internal and external parameters), “abrupt transition” used in IAEA Safety Standards when defining the term “cliff edge effect”.
The article provides examples of the manifestation of cliff-edge effects in various plant states and describes the measures that can be taken in plant design to avoid appearance of cliff-edge effects of different types.
Having in mind that it is not possible to completely exclude the deterioration of NPP status due to a small change in the parameter, the decision process whether it is reasonable to protect against such deterioration should take into consideration the significance of degradation of Defense in Depth, deterioration of radiation exposure, as well as risk considerations
Based on the results of the analysis of the cliff edge effect concept the paper presents a matrix for determining the necessity of taking measures in NPP design aimed at cliff-edge effect exclusion.

Email address:

GAErshov@atomproekt.com; lyubarskiy_av@aep.ru

Confirm that the work is original and has not been published anywhere else:

YES

156

A Response to the Regulatory Safety Challenges for New Reactor Technologies

Author: Abel Julio Gonzalez[1]

[1] *AUTORIDAD REGULATORIA NUCLEAR*

Corresponding Author: abel_j_gonzalez@yahoo.com

In addressing a main objective of the International Conference on Topical Issues in Nuclear Installation Safety, the paper present a response to the regulatory safety challenges for new reactor technologies, including evolutionary and innovative reactor designs, on the basis of the Argentine regulatory experience.
It is now recognized that new reactor technologies and designs generate increased interest, as technologies are maturing and plans to deploy these technologies are being made. In Argentina this interest started many years ago: a prototype of this types of reactors was developed and it is in construction: the reactor CAREM. The challenge for the Argentine Nuclear Regulatory Authority (ARN), therefore, refers to a real present rather than to a supposed future.
There are two main elements that have dominated the Argentine regulatory approach to evolving new nuclear technologies:
• the promotion of an international undertaking proclaiming renewed safety principles, the Vienna Declaration on Nuclear Safety; and,
• a renewed emphasis on safety criteria for accident mitigation (vis-à-vis prevention criteria), including the regulatory standardization of quantitative probabilistic criteria.
The paper summarizes the Argentine promotion of the Vienna Declaration on Nuclear Safety on principles for the implementation of the objective of the Convention on Nuclear Safety to prevent accidents and mitigate radiological consequences, which was adopted by the Contracting Parties of the Convention on Nuclear Safety (CSN) meeting at the Diplomatic Conference in Vienna, Austria, on 9 February, 2015. The Declaration was negotiated, and the Diplomatic Conference presided, by the then IAEA's Argentine Governor and now Director General. Ambassador Rafael Mariano Grossi. While the Declaration's principles reflect a consensus of all Contracting Parties, the Argentine leadership was apparent.
The paper analyses the Declaration's main principle, which applies to new reactors, namely that new nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions. It also discusses the implications of the subsidiary but important principle of establishing that national requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards. The Declaration also establishes that the principle should be reflected in the actions of the CSN's Contracting Parties, in particular when preparing their reports on the implementation of the CNS; in this respect, the paper summarizes the unique ad hoc actions by Argentina at the CSN.
Over many years of regulatory activity, ARN and its preceding bodies have emphasized the importance of mitigation and, to that effect that quantitative probabilistic criteria be an integral part of the regulatory requirements. The ethical basis and rational of the Argentine probabilistic requirements are discussed in the paper and their necessity for the safety of the forthcoming new technologies will be scrutinized.

Email address:

abel_j_gonzalez@yahoo.com

Confirm that the work is original and has not been published anywhere else:

YES

157

Development of uncertainty and sensitivity approaches for the analysis of severe accidents in a small modular reactor

Author: Pablo Hernan Ilardo[1]

[1] *Comision Nacional de Energia Atomica (Argentina)*

Corresponding Author: pablo.ilardo@cab.cnea.gov.ar

A research was conducted by the Nuclear Safety Department of CNEA on the analysis and comprehension of uncertainties on severe accidents code calculations for a Small Modular Reactor design. This work was accomplished due to the participation in the Cooperative Research Project (CRP) I31033 [1] launched by IAEA entitled "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors". Insights on best practices and a possible methodology to be used for uncertainty and sensitivity calculations associated with severe accident codes applications are included.
The present study is applied to a CAREM-like plant model based on CAREM-25 design using simplified hypothesis and approximated geometrical data, with the aim to improve the knowledge regarding severe accident management. MELCOR 1.8.6 [2] was considered in this work as the simulation tool to perform severe accident tests in both uncertainty and parametric analysis, and DAKOTA was chosen as the calculation tool for sampling and managing the interface with the plant code.
After having the MELCOR plant model developed and the coupling between codes properly established, a series of uncertainties studios were performed. A PIRT-like analysis was conducted to select relevant parameters affecting the defined Figure of Merits (FOM) and their corresponding probability distribution functions. The selected FOM represent key times of the in-vessel severe accident phase.
Simple and partial correlation tables, minimum and maximum FOM values and FOM distributions were obtained based on Wilks' formula approach. Four out of eleven initial parameters were obtained as relevant factors in the accident progression and their individual impact on each FOM and other relevant variables was evaluated by performing a centered parametric studio.
Failure rates of simulations were calculated and parameter dependence patterns were searched. MELCOR plant model improvement taken into account to drop down the number of failed tests is also presented.
Lessons learnt throughout the investigation regarding best engineering practices on uncertainty analysis are discussed and some recommendations are also presented.

[1] https://www.iaea.org/projects/crp/i31033
[2] NUREG/CR-6119, MELCOR Computer Code Manuals Vol. 1 Rev. 3 (Primer and Users' Guide) and Vol. 2 Rev. 3 (Reference Manual), versión 1.8.6, Septiembre 2005

Email address:

pablo.ilardo@cab.cnea.gov.ar

Confirm that the work is original and has not been published anywhere else:

YES

158

An Overview of the Nuclear Regulatory Framework for a Molten Salt Reactor Project in Turkey

Author: Tahir Akbaş[1]

[1] *TENMAK Nuclear Energy Research Institute*

Corresponding Author: tahir.akbas@tenmak.gov.tr

Nuclear power is an important option for the sustainable energy production and studies are ongoing to develop and deploy advanced nuclear reactor technologies in the word. Turkey has developed nuclear power plant projects based on the conventional light water reactors. Akkuyu NPP Project composed of four units WWER-1200 Reactor is under construction phase and is expected to start the operation in coming years. Nuclear Regulatory Authority has focused on the regulations for the existing and future nuclear power projects based on light water reactors.
On the other hand, interests in the advanced nuclear energy technologies have increased in recent

years due to the changing needs such as climate change and environmental impacts. Molten Salt Reactors (MSR) have many features that can support to meet these needs and MSR is one of the main technologies studied within the Generation IV International Forum. Turkish Energy, Nuclear and Mineral Research Agency has commenced recently a feasibility study for an MSR Project. In this paper a preliminary review study has been carried out for applicability of current nuclear regulations for the development of a typical Molten Salt Reactor project.

Email address:

tahir.akbas@tenmak.gov.tr

Confirm that the work is original and has not been published anywhere else:

YES

159

Enhanced Safety of Current and Future Nuclear Plant using Integrated Mechanistic Models and Data-Centric Assessment of Risk

Author: Michael Martin[1]

Co-authors: Martin McMann [2]; Simon Williams [2]

[1] *Rolls-Royce*

[2] *Rolls-Royce plc*

Corresponding Author: michael.martin@rolls-royce.com

This paper describes a data-centric approach to the design, manufacture, operation, maintenance and decommissioning of nuclear plant that can be used to manage a fleet of assets in-service and inform next generation designs. Key to this approach are advanced integrated models of through-life degradation mechanisms and their propagation to structural failure modes together with a consistent reliability-based approach for understanding the worth of all through-life data sources from raw material and manufacturing through to decommissioning. The key benefits of this are an improved understanding of conservatisms and opportunity to better focus resources to where they are most effective throughout the lifecycle.
The approach is illustrated by recent examples related to environmentally-assisted degradation mechanisms. Where necessary, the mechanistic models are informed by non-continuum influences such as microstructure and are validated by dedicated test-programmes that ensure optimal use of a sparse set of specimens. Surrogate models are used to provide a practical engineering assessment whilst retaining the key characteristics of the underpinning physical phenomena, coupled to a structural-reliability analysis of failure, the surrogate models provide a product-level digital twin whereby a conceptual design justification based on anticipated operational duty evolves to an as-built justification as manufacturing data is accumulated, leading to an as-operated justification as operational experience builds. For example, Non-Destructive Examination (NDE), In-Service Inspection (ISI) and Equipment Health Monitoring (EHM) data can be assimilated by the surrogate models together with wider fleet data leading to improved risk-based management of current plant and optimisation of next generation designs.
Integration of the outputs of such analyses with modern probabilistic risk evaluation methods can provide further benefits in terms of increased flexibility for through-life risk management. The peak failure frequency for components can be calibrated against target values, assigned to ensure that risk values are acceptably low at all times during life. The through-life distribution of failure frequency can then be mapped throughout the product lifecycle using a method such as phased mission analysis. This can release significant margins compared to current methods that generally apply a single limiting failure frequency value across much or all of the product lifecycle. This approach is summarised, to show where conservatism in risk analyses can be reduced, what benefits this can provide in terms of equipment or operational management, and how a framework can be put in place to allow appropriate target failure frequencies to be derived for individual components in a larger risk model, or for the totality of components in a risk model.

Email address:

michael.martin@rolls-royce.com

Confirm that the work is original and has not been published anywhere else:

YES

160

State of the art of IRSN's studies regarding fuel behaviour during LOCA

Authors: Antoine Lejosne[1]; Cristina Dominguez[1]; George Repetto[1]; Tony Glantz[1]

[1] *IRSN*

Corresponding Author: cristina.dominguez@irsn.fr

The loss of coolant accident (LOCA) safety case was reviewed in France during a rulemaking process which was finalized in April 2014. In this context, the French utility developed a new method to study LOCA considering physical phenomena related to fuel behaviour occurring during the transient: the fuel behaviour modelling is improved and more accurate. The first results obtained by this new method, assessed by IRSN, show a significant sensitivity of the cladding temperature to the input parameters, appearing from 750 to 800 °C: this sensitivity is linked to the fuel rod phenomena activation (ballooning, burst, contact between rods and fuel relocation).
However, as of today, it remains a challenge to confirm that this sensitivity of the fuel modelled phenomena is representative of reality or rather due to conservatisms chosen.
To answer this question, further studies are needed. IRSN's strategy regarding this topic is to rely on adequate numerical tools able to simulate those fuel rod phenomena during a LOCA. That is why, the DRACCAR software of the FUEL+ platform has been developed by IRSN since several years. Meanwhile, an advanced experimental fuel program called PERFROI (including thermomechanical and thermalhydraulic aspects) is under way to study specific issues relative to fuel behaviour under LOCA situation and to get experimental data for DRACCAR modelling and validation.

Email address:

antoine.lejosne@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

161

Natural circulation HERO-2 experiment simulations in the EU funded PASTEL Project

Author: Christophe Herer[1]

Co-author: To be determined

[1] *IRSN*

Corresponding Author: christophe.herer@irsn.fr

The recent development of innovative passive safety systems in the design of nuclear power plants, in particular small modular reactors, has made this topic an important area of investigation.

The PASTELS project has defined a framework for a multilateral cooperation between several organisations in the European Union aiming at reaching further advance in the knowledge on passive systems issues, related phenomena and validation of thermalhydraulic codes. Several aspects are being investigated, including studies on simple natural circulation configurations. Decay heat removal (DHR) passive systems connected to the secondary side (Steam Generator), known as Safety Condenser (SACO), are also considered. Studies on Containment Wall Condenser to control pressure in the reactor building complement this European Commission funded project.
This paper presents, using the thermalhydraulic system code ATHLET, CATHARE, RELAP5 and the CFD codes neptune_cfd and ANSYS, the analyses of experimental data measured on the HERO-2 loop. This is a simple loop supervised by ENEA (Italy), designed for basic studies of bayonet tubes and associated physical phenomena such as natural circulation and pool condensation under several operating conditions, aiming at their possible application in a DHR systems.

Email address:

christophe.herer@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

162

Challenges in Defining Emergency Planning Zones for Small Modular Reactors and Advanced Reactors: Review of Canadian Practices and Research at Canadian Nuclear Laboratories

Authors: Luke Lebel[1]; David Hummel[1]

[1] *Canadian Nuclear Laboratories*

Corresponding Author: luke.lebel@cnl.ca

There is a general sentiment that small modular reactors (SMRs) will have less offsite emergency preparedness and response (EPR) requirements than contemporary, large nuclear power plants. This has been attributed to their relatively small size, enhanced passive and inherent safety characteristics, and overall reduced risk. This also aligns with the Generation IV International Forum's safety and reliability goals for the next generation of nuclear energy systems, which are to eliminate the need for any offsite emergency response. Historically, offsite emergency response has been the fifth and final level of defence in depth for nuclear power plants, and therefore foundational to the concept of nuclear safety. The significance of the assertion that an entire level of defence may be eliminated begs the question of how EPR requirements will be determined in general for SMRs and advanced reactors. Most stakeholders agree that the requirements should be commensurate with the risk, but it is not obvious in all cases how the risks will be evaluated, and what levels of risk will be tolerated. This paper presents a review of EPR planning practices in Canada as manifested in the definition of emergency planning zones (EPZ) around presently operating nuclear power plants. Challenges that face the definition of SMR EPZs in the context of this recent experience are highlighted. Summary descriptions of some of the relevant recent and ongoing research at Canadian Nuclear Laboratories are also included.

Email address:

luke.lebel@cnl.ca

Confirm that the work is original and has not been published anywhere else:

YES

163

Argentine Experience in the Licensing of CAREM 25 Prototype Reactor

Authors: Pablo Torano[1]; Víctor Ibarra[2]

[1] *Autoridad Regulatoria Nuclear Argentina*

[2] *Autoridad Regulatoria Nuclear*

Corresponding Authors: vibarra@arn.gob.ar, ptorano@arn.gob.ar

The CAREM 25 is the first nuclear power reactor entirely designed and built-in Argentina. It is a prototype of a light water low power reactor with some features that simplify its design and contributes to a higher safety level. The construction of this prototype must demonstrate feasibility and safety on a smaller scale than that planned for commercial modules.
The CAREM 25 prototype reactor licensing process, carried out by the Nuclear Regulatory Authority (ARN), is based on the Argentine nuclear legal framework and its performance regulatory approach. Regarding the licensing process to construction and preliminary tests, the ARN has established a licensing scheme applicable to this stage. This framework defined the milestones for the beginning of construction and, in order to issue the construction authorization, the ARN established additional mandatory documentation requirements (License Conditions) regarding the traditional nuclear power plant licensing process to reinforce authorization.
ARN construction authorization is granted against a safety demonstration based on comprehensive deterministic and probabilistic safety analysis. One of the most challenging aspects of the licensing process of the CAREM 25 prototype reactor, was assessing the safety demonstration of a new reactor design, unique in Argentina. The regulatory approach of CAREM 25 licensing was developed under the concept of an "integral" evaluation of mandatory documentation, essentially the PSAR.
This assessment links the Demonstration of Safety and the Safety Classification of Structures Systems and Components (SSCs). From the Safety Classification of SSCs, engineering requirements are derived and the compliance of these have to be demonstrated in the chapters of the PSAR, within the description of design and engineering of SSCs. Acceptance criteria set by ARN include the adequacy of the Design Criteria and engineering requirements considering regulatory standards, current regulatory requirements, and good practices.
This way, engineering features are verified as consistent with the demonstration of safety and consolidated enough to allow ARN to grant the construction authorization of the CAREM 25 prototype reactor.

As a result of the assessment, ARN was also able to verify the requirements associated with:
• The Defense in Depth Principle, internalized in the design.
• Ample safety margins regarding design limits in all design base scenarios.
• A systematic methodology for the Safety Classification of SSC.
• The safety requirements derived from the post-Fukushima accident lessons that were already included in the design.
ARN could also verify additional features of CAREM design:
• Elimination by design of some initiating events
• Handling a wide range of multiple failure scenarios without exceeding Safety Limits.
• Extension of the grace period (operator actions not required to avoid core damage) up to 36 h.

Email address:

ptorano@arn.gob.ar

Confirm that the work is original and has not been published anywhere else:

YES

164

Regulatory Criteria Proposal for the Mission Time of the Sequences of the Event Trees of the L1 PSA for New Nuclear Power Reactors.

Authors: Federico José Carlín LLorente[1]; Pablo Torano[2]

[1] *Argentina Nuclear Regulatory Authority*

[2] *Autoridad Regulatoria Nuclear Argentina*

Corresponding Authors: ptorano@arn.gob.ar, fllorente@arn.gob.ar

In the framework of the licensing of the Argentine CAREM 25 prototype nuclear reactor, the Argentine Nuclear Regulatory Authority (ARN) agreed with the Responsible Entity a work plan to carry out the evaluation of the Level 1 Probabilistic Safety Assessment (L1 PSA), which is considered as Mandatory Documentation related to the Commissioning Licensing Milestone.
As a starting point of the work plan, from the regulatory point of view, the need arose to focus on the analysis of the Mission Time used for the modeling of the initiating events postulated in the L1 PSA, due it varies between active and passive systems. The particularity of this type of nuclear reactor is that the time for passive systems (it´s called: the grace period of the CAREM 25 reactor) is longer than for active systems. Therefore, this publication presents a regulatory criteria proposal for the Mission Time of the sequences of the event trees of the L1 PSA for new power nuclear reactors.
The work consisted of the development of a definition of the ARN for Mission Time term, the review of the bibliography of several sources, and the state of the art in the matter. Later, an analysis of the information collected was carried out by the experts, taking into account the design of the CAREM 25 prototype reactor.
As a conclusion, it is proposed that the Mission Time to determine the plant states, from the initiating events in L1 PSA, should be established according to the following:

• The review of the thermal-hydraulic results of the accidental sequences should not be limited to verifying the fulfillment of the acceptance criteria by the plant parameters within a predefined Mission Time.

• The Mission Time for each accidental sequence must be justified based on the trend of the relevant parameters and the dynamics of the accidental sequences. That is, the time analyzed must be extended until a stable situation or a trend towards a stable situation is reached.

Email address:

ptorano@arn.gob.ar

Confirm that the work is original and has not been published anywhere else:

YES

165

Evaluation of neutronic and thermal hydraulic parameters of an innovative molten salt reactor design using MCNPX and RELAP5 codes

Author: TARIQ Malatim[1]

[1] *Libyan Atomic Energy Establishment*

Corresponding Author: tariqmolatim@gmail.com

Abstract:
In this study an innovative reactor design of molten salt reactor type with power of 200 MWatt was chosen to be simulated, the design of the reactor was modificated to new dimensions that ensure

the simplicity and safe operation, the fuel materials composition also was changed, a multi cases of fuel composition were implemented, each case was simulated to find the most safe and proper fuel composition to be used. By using MCNPX, a comparison was implemented to evaluate the neutronic parameters of the reactor core in each case, while RELAP5-3D, was used for the thermal hydraulic analysis, furthermore the results were compared with other types of reactors. And since the concept of molten salts reactor is one of generation IV six technologies, the safety criteria of the generation IV will be applied.

Email address:

Tariqmolatim@gmail.com

Confirm that the work is original and has not been published anywhere else:

yes

166

Argentine Experience in the Application of the IAEA SSG-3 for the Evaluation of Procedures to Develop L1 PSA for CAREM 25.

Authors: Federico José Carlín Llorente[1]; Pablo Torano[2]

[1] *Argentine Nuclear Regulatory Authority*

[2] *Autoridad Regulatoria Nuclear Argentina*

Corresponding Author: fllorente@arn.gob.ar

In the framework of the licensing of the CAREM 25 nuclear power reactor prototype, the Argentine Nuclear Regulatory Authority (ARN) agreed with the Entity Responsible for the project a Work Plan to carry out the evaluation of the Level 1 Probabilistic Safety Assessment (L1 PSA). The PSA is considered by the ARN as a Mandatory Documentation requirement related to the Commissioning Licensing Milestone.
The approach adopted in the Work Plan for the evaluation of the L1 PSA of the CAREM 25 prototype reactor is to carry out an ON-LINE review as proposed in the IAEA "Safety Reports Series N°25 (2002)" and consists of the developer of the L1 PSA immediately after finishing a certain task sends it to the regulator so that it can evaluate it. The main advantage of this approach is that many of the findings found in the review can be taken into account before developing the following tasks in the PSA.
Currently, the Work Plan is in Stage 1 – Consolidation, completed. This means that the ARN completed the evaluation of the technical aspects of the procedures developed by the Responsible Entity, to carry out L1 PSA, in relation to the guidelines proposed by the IAEA SSG-3 guide.
As the scope of the evaluation of the procedures, it was defined to take into account the guidelines of the SSG-3 referring to operation at full power for internal events, since it will be the same scope to be considered by the ARN for the evaluation of the L1 PSA of the CAREM 25 prototype reactor.
This work compiles the challenges, experience, and conclusions obtained by the ARN through the process of applying the IAEA SSG-3 guide for the evaluation of the procedures for carrying out the L1 PSA for the CAREM 25 prototype reactor.

Email address:

ptorano@arn.gob.ar

Confirm that the work is original and has not been published anywhere else:

YES

167

Multi Agent Systems Application Concept for Enhancing Reactor Safety and Operations in an Innovative SMR

Authors: Jae-Cheon Jung[1]; philip Gyan[2]

[1] *KEPCO International Nuclear Graduate School*

[2] *NUCLEAR REGULATORY AUTHORITY*

Corresponding Author: philsbea@gmail.com

The initiative for Gen IV nuclear reactors had lead many designers to devise several advanced design features to improve the visibility of inherent reactor safety targeting Small Modular Reactor (SMR). Stakeholders have identified high levels of automation and some human performance issues related to the design and operation of these reactors, despite the advancements. The complexity in multi-module man-machine interface operations, difficulties transitioning between automation and manual control workloads, MCR operator monitoring, and control navigation complexity are only a few of these issues. To overcome these concerns, it is critical that further research is required. Nowadays Multi Agent Systems (MAS) are used extensively in the industry to solve several complex engineering problems, therefore a proven verification and validation procedure is required before deploying this technology in an innovative SMR to improve safety and provide intelligent operator support. As a result this research is aimed at demonstrating the concept of how the application of this proposed technology can be verified and validated using a systems engineering approach.

Email address:

p.gyan@gnra.org.gh

Confirm that the work is original and has not been published anywhere else:

YES

168

Contextual Integrated Risk-Informed Decision-Making

Author: Gueorgui Petkov[1]

[1] *Kozloduy NPP*

Corresponding Author: petkovgi@yahoo.com

Probabilistic risk assessment and management seeks to reach the standards of theoretical systematicity and empirical accuracy achieved in the models of natural sciences. To this end, a set of standards has been developed and improved in the form of statistically validated measures and probabilistic principles based on deterministic and probabilistic analysis. This set allows systematically derive potentially objective risk assessments and to apply them for the integrated risk-informed decision-making. But it must also be borne in mind that much of the data and models are subjectively influenced by the uncertainty of the context in which they are obtained and linked. In other words, PRA in its current state contains a large amount of risk information, but without context these models and results may not have real predictive and explanatory power and a common solid basis for comparative and integrated decisions. Therefore, thousands of sequences of events and transitions between complex system states need to be monitored and analyzed by integrated code simulations with specific boundary conditions and actual configurations to understand and predict risk. These sequences of events must be checked, grouped, and restricted to make the task practical for PRA purposes. And the detailed deterministic safety analyses should be also reduced to a number of representative event sequences and identified bounding cases that have similar accident progressions. For the entire multi-unit NPP site, the cases become even more numerous and complex, as it is necessary to take into account not only the technological conditions of the NPP, but also the whole context of the site, including human, organizational and environmental factors. A dynamic symptom-based context evaluation procedure could be used as a probabilistic tool for deterministic-probabilistic safety analyses interface in order to contextualize and supplement the existing risk metrics but not to replace them. The paper presents opportunities for the context quantification procedure of the Performance Evaluation of Teamwork method for contextual integrated risk-informed decision-making.

Email address:

gipetkov@npp.bg

Confirm that the work is original and has not been published anywhere else:

YES

169

INPRO studies on Transportable Nuclear Power Plants and Modules and key legal issues for their regulations

Authors: Alexander BYCHKOV[1]; Brian Boyer[None]

[1] *Senior Nuclear Engineering Expert - INPRO*

Corresponding Author: a.bychkov@iaea.org

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) established in 2000 serves as the IAEA's integrated, forward-looking activity in nuclear power examines cross cutting nuclear fuel cycle issues. INPRO performed a key collaborative project called Legal and Institutional Issues of Transportable Nuclear Power Plants (TNPP): A Preliminary Study. Consideration of different crosscutting issues: Infrastructure, safeguards, legal, nuclear liability, nuclear safety and security drove IAEA efforts in examining TNPP and small modular reactors (SMRs). Building on the results of that effort INPRO started the Case Study for the Deployment of Factory Fuelled SMRs in 2015. The study aimed to examine legal and institutional issues for export of a factory fuelled, tested and sealed Transportable Nuclear Module (TNM) for deployment as a TNM power plant (TNMPP). The TNM/TNMPP concept is novel in applying IAEA safeguards, complying with international legal norms, and maintaining IAEA standards in safety and security over its lifecycle. The study analysed a submersible, a floating and a land based TNM. TNMs will be manufactured as serial and industrial units, so the expected TNMPP could meet advanced safety solutions and reduce the time from decision making to deployment in the Host State. The study considered scenarios of "maximum outsourcing" as examples to demonstrate potential advantages of the factory manufactured, fuelled, and sealed TNMs, specifically addressing affordability and sustainability. Legislative issues included maritime law and civil liability for nuclear damage relating to nuclear safety and security, along with safeguards and licensing. The examination of the responsibilities of Supplier versus Host States challenged present standards and regulations. Despite the absence of international regulations developed for TNM/TNMPP, the existing legal and regulatory framework is generally applicable for all TNM/TNMPP through all lifecycle stages. Safeguards, civil liability for nuclear damage, etc. may require modifying present legal frameworks, however, intergovernmental agreements may cover these identified "gaps". The study fixed "the gap" in two legal approaches related to marine application of a TNM, as well as the maritime law during TNM relocations between Supplier and the Host States. The elaboration of specific recommendations for national regulators and for import/export cooperation would be useful for safe deployment of the TNM/TNMPP's concept.

Email address:

a.bychkov@iaea.org

Confirm that the work is original and has not been published anywhere else:

yes

170

The Design and Application of the Intelligent Accident Analysis System for the Nuclear Power Plant

Author: Guoyang Ma[1]

[1] *China nuclear power operation technology corporation,LTD*

Corresponding Author: magy@cnnp.com.cn

This paper introduced an intelligent accident analysis system for the PWR nuclear power plant, which is developed to: (1) provide the accident cause diagnosis and the safety status analysis through the designed diagnostic algorithms, (2) provide synchronized tracking and rapid prediction of the accident processes with the advanced module accident analysis program MAAP5,(3) as a tool for evaluating and prioritizing accident management strategies, (4) as a tool for visual monitoring of the accident phenomenon using 3D modeling method, (5) assist in execution of the severe accident management guidelines(SAMG) of the plant. This intelligent accident analysis system can be used in the nuclear accidents to alleviate the accident risks to the public and the environment. And it also can be used as a convenient way to train utility and plant personnel with respect to nuclear accidents. Therefore this system not only can improve the plant safety performance, but also promote the growth of profits and social stability.

Email address:

magy@cnnp.com.cn

Confirm that the work is original and has not been published anywhere else:

Yes

171

Development of a structural integrity model for the steam generator tubing of the SMART100 small modular reactor

Authors: Russell Cipolla[1]; Minki Cho[2]; Jongha Kim[2]; Michael Liu[1]

[1] *Intertek AIM*

[2] *Doosan Heavy Industries and Construction*

Corresponding Authors: jongha2.kim@doosan.com, minki1.cho@doosan.com, russell.cipolla@intertek.com, michael.c.liu@intertek.com

The proposed System-Integrated Modular Advanced Reactor (SMART100) small modular reactor (SMR) designed by the Korea Atomic Energy Research Institute and Korea Hydro & Nuclear Power is a pressurized water reactor (PWR) that uses once-through steam generators (SG). In the SMART100 SG design, heat is transferred from the primary to secondary loops across helically-coiled tubes. Unlike typical large PWRs, the SMART100 SG circulates high-pressure primary water over the outer surface of the tubes, generating steam at a lower pressure within the tubing. The novel tubing geometry and pressure arrangement mean that conventional PWR SG analyses and guidelines used to ensure the structural integrity of SG tubing against failure by burst may require further analysis for application to SMART100.

This paper describes the development of structural integrity criteria for the SMART100 SG, including a review of prior experiments and similar helically-coiled SG design history, postulated degradation and collapse failure modes, computer modeling using finite element analysis, and laboratory testing of tube segments to failure to confirm the model results. In addition, a simple analytical prediction of collapse pressure was hypothesized and compared with laboratory testing results.

Email address:

michael.c.liu@intertek.com

Confirm that the work is original and has not been published anywhere else:

YES

172

Development and Application of A Severe Accident Management Guidelines VerIfICATION and Drilling Platform at CNNP

Author: Wei Wei[1]

[1] *China National Nuclear Power Operation Technology Corporation, LTD.*

Corresponding Author: weiwei02@cnnp.com.cn

With the rapid growth of nuclear power plants in China, the simulator has been used as one of the tool for personnel training to meet the requirement of sufficient supply of nuclear power employees. Current simulators oriented to different audience for corresponding training purpose in China has been described in the paper. The directions of improvement for simulators are addressed as well. After the Fukushima severe nuclear accidents in Japan, China national nuclear safety regulatory authorities have promoted the requirements for severe accident management of nuclear power plants, and the new version of the nuclear power plant operational safety requirements of China (HAF103) have been proposed that regulate nuclear power plant operators must accept severe accidents related training(NNSA, 2013). This requires that the FSS (full-scope simulator) of new plants should have severe accidents simulation capabilities, and the old ones should be updated to expand the simulation scope to the stage of severe accidents. Existing severe accidents analysis programs, such as MELCOR (Merrill et al., 2010),MAAP4 (Ferng et al., 2010), SCDAP-RELAP5 (Sharm et al., 2011), CONTAIN (Carroll et al., 1987),VICTORIA (Makynen et al., 1997), CATHARE/ICARE (Bandini et al.,2011), are too large scale and complex structureto be applied to the FSS.

Email address:

weiwei02@cnnp.com.cn

Confirm that the work is original and has not been published anywhere else:

YES

173

Using INPRO Methodology for a Holistic Sustainability Assessment in Safety, Safeguards and Security (3S)

Author: Carolynn SCHERER[1]

Co-author: Brian BOYER

[1] *IAEA - INPRO*

Corresponding Author: c.scherer@iaea.org

The International Project for Innovative Nuclear Reactors and Fuel Cycle Facilities (INPRO) has a methodology for assessing the sustainability of innovative and evolutionary nuclear energy systems

(NES) in the areas of infrastructure, economics, environmental impact of stressors and depletion of resources, safety for nuclear reactors and fuel cycle facilities, waste management, and proliferation resistance. The INPRO methodology uses a hierarchical approach to assess sustainability, based on fundamental basic principles (BP), which defines the goal or target for each area that the NES needs to achieve to be sustainable. Each BP has user requirements (UR) and criteria (CR) which enables the assessor to evaluate the NES. The INPRO Methodology identifies weaknesses and gaps in a countries NES. This methodology applies a holistic approach to perform a sustainability NES assessment (NESA) and is particularly useful in supporting application of the 3S approach in safety, security, and safeguards. Specific areas of assessment are in safety of nuclear reactors, and most recently updated in proliferation resistance, which covers the safeguardability and proliferation by State actors, and in nuclear security. The NESA identifies gaps and weaknesses that designers, technology holders and States can address in the design phase of innovative nuclear reactors.

Email address:

c.scherer@iaea.org

Confirm that the work is original and has not been published anywhere else:

Yes

174

ETSON - The European Technical Safety Organisations Network

Author: Uwe Stoll[1]

[1] *GRS gGmbH*

Corresponding Author: uwe.stoll@grs.de

Technical safety organisations (TSO) play an important role in nuclear safety and security. They are science-based organisations which support the nuclear regulatory body of their respective country by performing technical safety assessments. This requires a high level of competent, reliable and impartial technical expertise in many different subject areas.
Furthermore, many TSO strongly engage in furthering the state of science and technology by conducting research and development, thus providing both the knowledge and the analytic tools needed to ensure a high level of safety and security in the nuclear field.
In 2006, the European Technical Safety Organisations Network ETSON was founded with the aim to develop common approaches for harmonized nuclear safety assessments and to share technical and scientific knowledge and experiences among TSOs from different countries. Since 2011 ETSON is an independent legal entity and as this serves as a common platform for its member organisations for working together on current issues on nuclear safety.
Currently, ETSON consists of 16 TSOs from different countries, whose members exchange information in 14 expert groups on various topics of nuclear safety assessment and research, ranging from generic aspects such as safety concepts or emergency preparedness and response to specific technical fields like safety fluid systems or mechanical and electrical systems. The output of the expert groups' activities is published in Technical Safety Assessment Guides as part of ETSON's publications. Furthermore, workshops on specific technical and scientific issues are organized by individual member organisations on behalf of the network.
With a view to supporting further the definition and the implementation of nuclear safety research programmes in Europe, the network has also established the ETSON Research Group. Since 2000, the group´s experts collaborate to develop and communicate common positions on relevant research tasks, both within the networks and to stakeholders on the European and international level. This includes the identification of research needs as well as the definition and implementation of respective research programmes.
As the third pillar of the network´s technical and scientifical activities, the ETSON Knowledge Management Group is tasked to foster the mutual exchange on concepts and tools to share relevant knowledge to enhance cooperation within the network as well to contribute to enhancing nuclear safety in general.

With a view to foster the development of the next generation of experts in nuclear safety and security, ETSON has established its Junior Staff Programme (JSP) which brings together young experts from member organisations to learn from each other and form personal networks across national borders. For example, they organize annual workshops and the competition for the annual ETSON award as part of the Eurosafe conference, which is organized by ETSON.
This contribution will give an overview of ETSON, its current activities in terms of safety assessments and research projects and its further contribution to international IAEA activities mainly in the GNSSN TSO Forum.

Email address:

uwe.stoll@grs.de

Confirm that the work is original and has not been published anywhere else:

Yes

175

Incorporation of passive system functional reliability in Probabilistic Safety Assessment

Authors: Clément Laclouere[1]; Elisabeth Duvillard[1]

[1] *EDF*

Corresponding Author: elisabeth.duvillard@edf.fr

Advanced nuclear reactor designs incorporate safety passive systems in addition to active ones. Assessment of the capability of such passive systems in performing satisfactorily their functions, also called "functional reliability", is an important issue for the safety analysis. Operating of passive system functions is based on thermal-hydraulic (T-H) principles, which are not generally considered to be subject to any kind of failures in conventional Probabilistic Safety Assessment (PSA). But due to the environment and to the physical phenomena that may deviate from expectation, passive systems may fail to meet their required functions. Quantification process of the unreliability figures of passive systems has to consider uncertainties of numerous parameters which can affect the accomplishment of system functions during a transient. RMPS (Reliability Method for Passive System) is one of the methodologies that address the reliability assessment of a passive system and its incorporation into a PSA model. Thus, PSA insights will provide risk contributions of mechanical, electrical, and I&C system failures, human errors, and functional reliability of passive systems as well. RMPS methodology uses a statistical approach to assess the reliability of a passive system taking into account the uncertainty variation of the key physical and code parameters. Based on the probability distribution of each main parameter, a Monte Carlo simulation drawing gives a set of randomized values for each parameter.

This paper will present the example of the use of RMPS for the development and risk quantification of PSA accident sequences for PWR loss of main feedwater transient, crediting the operation of a SACO (SAfety COndenser) passive system to remove the decay heat. To assess the passive system functional reliability, which is highly dependent on physical phenomena, the PSA analyst considers all the possible T-H states that can be reached by the reactor depending on the passive system's performances. Then, for each identified T-H state, all the possible protection systems and mitigation safety features, and human actions that can be credited are identified. RMPS results correspond to a probability distribution for each T-H state, and for all the relevant functional sequences of the event tree. In this way a full event tree development and risk quantification are possible by integrating these probabilities as a multiple branch event.

The results of this exercise will be discussed in the paper, and several ways of optimization will be presented in an effort to reduce the overall number of sequences, to increase the PSA model's readability and to possibly reduce the full calculation run time.

Email address:

elisabeth.duvillard@edf.fr

Confirm that the work is original and has not been published anywhere else:

YES

176

Machine learning applications for nuclear safety: An overview

Author: Samuele Meschini[1]

Co-authors: Raffaella Testoni [1]; Massimo Zucchetti [1]

[1] *Politecnico di Torino*

Corresponding Author: samuele.meschini@polito.it

Machine learning (ML) algorithms are rapidly spreading to many different fields, such as computer vision, automotive, finance, and safety. The capability to learn and execute tasks without being explicitly programmed makes ML algorithms a tool with an enormous potential. Yet, the black box nature of ML algorithms arises doubts and concerns when applied to safety. Lack of transparency, model interpretability, safe space exploration, and robustness to distributional shift, to cite few, are all open issues in ML safety. This paper presents an overview of published works on ML applications for safety in different fields. A special focus is given to ML applications for nuclear safety, which are still limited in number if compared to other fields. Good practices and possible weaknesses of these applications are highlighted. The choice of the algorithm(s) is usually case-dependent, but a general approach for the analysis can be depicted. Then, guidelines for the development of robust ML applications in the nuclear safety field are proposed.

Email address:

samuele.meschini@polito.it

Confirm that the work is original and has not been published anywhere else:

YES

177

IRSN research and development strategy for the assessment of passive systems and the safety of future small modular reactors.

Author: Olivier Marchand[1]

[1] *IRSN*

Corresponding Author: olivier.marchand@irsn.fr

Passive systems are expected to be increasingly used in future nuclear power plants and should even be central to the safety of small modular reactors. In this context, the technical safety organizations (TSO), and in particular the IRSN, will have to be able to assess the proper functioning of these systems in any situation.

These systems are most often based on natural physical phenomena (natural convection, condensation, etc.) which must be modelled in a satisfactory way. This is not necessarily the case for all foreseen designs and all incidental or accidental situations.
IRSN has chosen to concentrate its research and development activities on systems that could be installed in France in the near future, i.e. those relating to pressurized water reactors (Gen-III+ and SMR).
IRSN's general policy on this subject is therefore organized around four main lines of action:
-Collaborate at the European level in the framework of EURATOM and SNETP to promote European activities in this field;
-Discuss with French designers to work on relevant systems;
- Strengthen the activity of the European TSO network, ETSON, on this subject;
-Develop its own R&D capabilities.
For each of these axes, this document presents the context, the state of progress and the main perspectives.

Email address:

olivier.marchand@irsn.fr

Confirm that the work is original and has not been published anywhere else:

YES

178

Reliability-Based Optimisation with Nuclear Data Uncertainty

Author: Enrique Miralles-Dolz[1]

Co-authors: Ander Gray [1]; Edoardo Patelli [2]

[1] *University of Liverpool*

[2] *University of Strathclyde*

Corresponding Author: enmidol@liverpool.ac.uk

Neutronics simulations are an essential part of both fission and fusion reactor design. Simulations are affected by two main sources of uncertainty: from the nuclear data, and from the inherent statistical uncertainty of the Monte Carlo methods. With the Total Monte Carlo methodology, it is possible to propagate the nuclear data uncertainty through the simulations, accounting for both nuclear data imprecision and statistical uncertainty.

Neutronics simulations are used in the calculation of the safety and reliability of the nuclear components and systems. These designs are typically found defining some constraints (e.g. safety margins) and finding the solution that optimises certain figure of merit in the model. With reliability-based optimisation, it is possible to find the best design solution while considering the effects of uncertainty in the neutronics simulations. However, taking into account uncertainties in the neutronics simulations can be computationally demanding, and therefore it is desirable to apply optimisation methods that perform the minimum possible model evaluations. Building a surrogate model of the reliability function using Gaussian Processes, and applying the expected improvement acquisition function to perform the sampling, is a method that can do both: taking into account uncertainties in the simulation, and finding the optimum solution in a computationally efficient way.

This paper presents a methodology for reliability-based optimisation, where Total Monte Carlo has been coupled with OpenMC, and a Bayesian optimisation approach has been taken to account for both nuclear data imprecision and statistical uncertainty in the neutronics simulations of the optimisation problem.

References:

Gray, Ander, Andrew Davis, and Edoardo Patelli. "Uncertainty propagation in SINBAD fusion benchmarks with total Monte Carlo and imprecise probabilities." Fusion Science and Technology 77.7-8 (2021): 802-812.

Valdebenito, Marcos A., and Gerhart I. Schuëller. "A survey on approaches for reliability-based optimization." Structural and Multidisciplinary Optimization 42.5 (2010): 645-663.

Janusevskis, Janis, and Rodolphe Le Riche. "Simultaneous kriging-based estimation and optimization of mean response." Journal of Global Optimization 55.2 (2013): 313-336.

Email address:

enmidol@liverpool.ac.uk

Confirm that the work is original and has not been published anywhere else:

YES

179

Remote Monitoring of IoT Integrated Reactor Temperature and Visual Condition in Small Modular Reactor Safety

Authors: Isna Syilmi Qaira[1]; Muh. Suhalmin Sriwa[1]

[1] *Indonesian Polytechnic of Nuclear Technology*

Corresponding Author: isnasyilmiqaira@gmail.com

As many countries use nuclear small modular reactors (SMR), the safety aspect of nuclear reactors is now very important, in anticipation of nuclear accidents due to operator error or natural disasters. Many efforts can be made to improve monitoring from time to time on the condition of the SMR plant to identify errors in the system that are not in accordance with procedures, especially in regulating the reactor temperature. Overall this safety system is monitored with a Raspberry Pi camera as a liaison between the camera and the admin and can be monitored online. This monitoring system will send information detected by sensors and cameras to the admin in real time based on IoT. The data from the temperature sensor which is connected to the NodeMCU is then forwarded to the MQTT Broker using the publish topic. Clients (android/PC) who want to monitor temperatures must subscribe to published topics so they can get temperature data and visualize plant conditions along with warnings if the temperature exceeds the specified limit. The method used in this research is design, while the materials used in this research are Raspberry Pi 3, Raspberry Pi camera, DHT-22 sensor, and MCU Node.

Email address:

isnasyilmiqaira@gmail.com

Confirm that the work is original and has not been published anywhere else:

YES

180

Easing the nuclear safety demonstration integrated into projects using MBSE and Artificial Intelligence.

Author: Emir Roumili[1]

Co-authors: Jean Francois BOSSU [2]; Marinos PANAYIOTOU [3]; Robert PLANA [2]; Lavinia RAGANELLI [3]; Jas SIDHU [3]

[1] *Assystem/IMT Alès*

[2] *Assystem*

[3] *CRA*

Corresponding Author: roumili.emir@gmail.com

Nuclear industry projects are becoming increasingly complex. Alongside the project complexity, nuclear safety is a priority in all nuclear projects which have varied constraints of scope, schedule, budget, quality, resources [1]. Indeed, the amount of effort required to demonstrate nuclear safety in a complex project such as a NPP can escalate quickly and compromise the project success. IAEA-TECDOC-1919 [2] supports using information technology to simplify management of the different phases of a NPP lifecycle. This paper presents how digital tools can facilitate demonstrating nuclear safety through a hybrid approach based on Model Based System Engineering (MBSE) and Artificial Intelligence.
Systems engineering [3], [4] has proven advantages in various industrial fields for coordinating complex systems engineering projects. MBSE [5] is the practice of developing a set of related system models that help define, design, and document a system under development. These models provide an efficient way to explore, update, and communicate system aspects to stakeholders, while significantly reducing or eliminating dependence on traditional documents.
Artificial Intelligence is defined as the study of "intelligent agents" [6]: any system that perceives its environment and takes actions that maximize its chance of achieving its goals. In the last few years connectionist AI, mostly neural nets, and deep learning, have been widely adopted enabled by the rapid growth of computational power.
This project aims to create a MBSE metamodel (A model about a model. [4] i.e. the concepts, their attributes, the links between them which will allow the system of interest to be modelled later on) to extend metamodels already used in other industries and in their modelling approaches and software. This will enable the development of a DSML adapted to nuclear safety modelling which will be directly integrated into MBSE software. Modification of existing MBSE models is key to support the demonstration of nuclear safety. A test case on a NPP system is used to demonstrate the viability of our approach. The AI contribution will take the form of induction processing which is highly efficient but not adapted to all type of activities. Our metamodel also allows us to identify the parts of nuclear safety demonstration process where connectionist AI can contribute. These inductive approaches are based either on supervised learning models leading to the creation of datasets composed by domain experts or on unsupervised approaches. Our current usage of Natural Language Processing (AI applied on language) is mainly aimed at identifying safety requirements in unstructured documentation (regulations etc.) and processing these requirements.

[1] A guide to the project management body of knowledge (PMBOK guide), Fifth edition, 2013.
[2] Application of Plant Information Models to Manage Design Knowledge through the Nuclear Power Plant Life Cycle IAEA TECDOC No. 1919. IAEA, 2020.
[3] ISO, « ISO/IEC 15288 Systems and software engineering — System life cycle processes », 2015.
[4] « Guide to the Systems Engineering Body of Knowledge (SEBoK) », INCOSE.
[5] B. Schindel, « INCOSE Model-Based SE Transformation », p. 30.
[6] S. J. Russell, P. Norvig, et E. Davis, Artificial intelligence

Email address:

eroumili@assystem.com

Confirm that the work is original and has not been published anywhere else:

YES

181

The Regulatory Treatment of Low Frequency External Events as part of a Risk-Informed Performance-Based Approach

Authors: David Grabaskas[1]; Ben Chen[1]; Richard Denning[2]

[1] *Argonne National Laboratory*

[2] *Consultant*

Corresponding Author: dgrabaskas@anl.gov

To assist the advanced reactor industry in future licensing efforts, the U.S. Department of Energy Advanced Reactor Demonstration Program Regulatory Development area initiated a project at Argonne National Laboratory to examine the regulatory treatment of external hazards as part of a risk-informed performance-based (RIPB) licensing framework. A RIPB licensing framework for advanced reactors built on establishing an affirmative safety case offers the benefits of increased flexibility regarding key design and licensing decisions based on a detailed assessment and understanding of plant risk.

Historically, reactor licensing addressed events of very low frequency primarily through the application of design margin and defense-in-depth philosophy. In contrast, RIPB approaches attempt to evaluate these scenarios at a level of detail commensurate with their risk, which often necessitates an explicit treatment of their frequency and associated consequence. While the detailed analysis of low frequency events provides insights that can help justify alternative treatments to past conservatism, the findings are dependent on the quality and confidence associated with the analyses. The assessment of external hazards presents a unique challenge, as their potential frequency of occurrence, especially of large magnitude events, is inherently uncertain given the long return periods in question.

This project aims to identify the benefits and challenges of such approaches for advanced reactor vendors and aid in the development of consistent and appropriate analysis methodologies. The paper summarizes project findings and explores the application of various approaches for different external hazards. In addition, the current work also evaluates the application of the quantitative health objectives as a limit on external event risk, as they are a potential regulatory requirement under the current draft 10 CFR Part 53, which is a new technology-neutral reactor licensing pathway in the U.S.

Email address:

dgrabaskas@anl.gov

Confirm that the work is original and has not been published anywhere else:

YES

182

Markov Decision Processes for Intelligent, Risk-Informed Asset-Management Decision-Making

Authors: David Grabaskas[1]; Yunfei Zhao[2]; Carol Smidts[2]; Moiseytseva Vera[1]; Roberto Ponciroli[1]; Pascal Brocheny[3]

[1] *Argonne National Laboratory*

[2] *The Ohio State University*

[3] *Framatome*

Corresponding Author: dgrabaskas@anl.gov

Advanced nuclear reactors are a promising option for aiding the world in achieving its net-zero carbon emission goals, however, there are significant challenges to attaining and maintaining economic competitiveness with other sources of electricity. To improve the economic competitiveness of advanced reactor designs, a project was initiated to explore the use of Markov Decision Processes (MDPs) to guide asset-management decision-making during advanced reactor operation. MDPs are a powerful tool for optimizing decision-making in complex environments and their application to advanced reactors can aid in planning maintenance and repair activities to minimize downtime and maximize generation. The described approach expands on previous work regarding the use of MDPs for operational decision-making through the direct incorporation of real-time plant information. The integral MDP analysis includes information from online component diagnostic tools and the plant's real-time generation risk assessment (GRA) and probabilistic risk assessment (PRA), which evaluate plant risk from both an economic and safety perspective. The result is an asset-management optimization framework that is based on real-time data regarding plant component status and the current best-estimate of plant risk. The paper presents an overview of the theoretical framework to incorporate the different information pathways into an integral MDP analysis, along with example analyses.

Email address:

dgrabaskas@anl.gov

Confirm that the work is original and has not been published anywhere else:

YES

183

Remote Imaging of The IoT Integrated Smartphone Display for Analysis Of Leaks in Reactor Pipes and Vessels As An Initial Command for Complete Automatic Shutdown Mode in The Small Modular Reactor Plant

Authors: Isna Syimi Qaira[1]; Muh. Suhalmin Sriwa[None]

[1] *Indonesian Polytechnic of Nuclear Technology*

Corresponding Author: isnasyilmiqaira@gmail.com

Leakage of fluid flowing in the pipes and reactor vessels in the small modular nuclear reactor (SMR) is currently a very serious problem. crucial especially for novice countries using nuclear technology. So that monitoring of the leak system in real time is very necessary. This research aims to produce a method to design a tool that can detect the location of fluid leaks in pipes and vessels quickly and accurately. The method is carried out by using two flowmeter sensors which are placed before and after the leak points of pipes and vessels to record data on the difference in fluid flow in and out (ΔQ). The resulting data is transmitted to a computer using a network based on Transmission Control Protocol/Internet Protocol (TCP/IP). Then the operator can see and can immediately determine where the leak is in the pipe and vessel system by looking at the danger signs displayed on the piping and instrumentation diagrams in the TCP/IP system. The imaging results are converted into simpler hazard sign information via a smartphone display.

Email address:

muhsuhalminsriwa@outlook.com

Confirm that the work is original and has not been published anywhere else:

YES

184

Lessons Learned from Exploring Safety, Security, and Safeguards Interfaces in Advanced and Small Modular Reactor Technologies

Authors: Adam WILLIAMS[1]; Alan Evans[1]; Benjamin Cipiti[1]; Douglas Osborn[1]; Jesse Bland[2]

[1] *Sandia National Laboratories*

[2] *Sandia National Labs*

Corresponding Author: adwilli@sandia.gov

Evolutionary and innovative reactor technologies are moving toward smaller and advanced designs that will impact the ability of nuclear installations to achieve desired levels of safety, security, and safeguards (3S). Yet, such technologies also present new opportunities to address 3S interfaces. For example, as installation operators seek reduced physical protection footprints, they may apply knowledge of safety systems to mitigate potential sabotage scenarios and align material monitoring for safeguards with accounting for security. Similarly, the anticipated increase in digitization in nuclear installations to support these reactor technologies enhances the importance of addressing cyber interfaces to ensure adequate 3S operations.

In response, Sandia National Laboratories (Sandia) has supported a range of research projects and design-related engagements better understand different 3S interfaces for advanced reactor technologies and facilities.

This paper will review a set of Sandia-developed examples demonstrating opportunities related to 3S interfaces. Supported by both research-based analysis and practice experience, risk reduction strategies for such reactors and installations are enhanced when accounting for interfaces in both 3S operations and uncertainty. The conclusions, insights, and implications from these examples help frame approaches to better address 3S interfaces in designing, deploying, licensing, operating, and decommissioning evolutionary and innovative reactors.

Email address:

adwilli@sandia.gov

Confirm that the work is original and has not been published anywhere else:

Yes

185

Experimental Investigation of Iodine Removal in a Lab Scale Setup of Filtered Containment Venting System

Author: AMJAD Farooq[1]

Co-authors: Jawaria Ahad [2]; Masroor Ahmad [2]; Kamran Qureshi [3]; Waseem Siddique [3]; Khalid Waheed [2]; Naseem Irfan [2]

[1] *Pakistan Institute of Engineering & Applied Sciences, Pakistan Atomic Energy Commission*

[2] *Department of Nuclear Engineering*

[3] *Department of Mechanical Engineering*

Corresponding Author: aaffqq@gmail.com

Nuclear powerplants are a clean and reliable source of energy. They are designed very conservatively with wide safety margins, and they contain multiple safety systems for handling design-basis accidents. After severe accidents like Three Mile Island, Chernobyl and Fukushima, many concerns were raised regarding the safety of people and environment from beyond design-basis accidents. Majority of countries conducted stress tests and critical reviews of the existing safety systems and prepared action plans in response to these accidents. As a result of these action plans, for prevention and mitigation of severe accidents in both short- and long-term view, many countries installed a passive safety system named Filtered Containment Venting System (FCVS). The main objectives of this system were to maintain the integrity of containment, control over-pressurization by transferring some portion of steam and gas mixture from containment to this system, removal of hazardous gases and safety of people and environment. Main focus of this system was removal of iodine because of its hazardous nature and major potential contribution to source term to the environment. Upon exposure, it can get accumulated inside the thyroid gland of human beings and can cause thyroid cancer. Removal of this radioactive product is very important to ensure safety of people. As a response to Fukushima, Pakistan also presented a Fukushima Response Action Plan (FRAP) and following this plan, PIEAS initiated its research on removal of iodine. A lab scale setup was developed at PIEAS to study removal of iodine for understanding the mechanism involved in wet scrubbing and to improve removal efficiency. Removal of both elemental iodine and methyl iodide was investigated. Setup was equipped with a compressor to simulate the high pressure in accidental scenario. Compressed air was passed onto a moisture separator for moisture removal and a rotameter for flow control. Compressed air at a specific flowrate was then heated above 100oC in order to keep iodine and methyl iodide in gaseous form. Heated air was mixed with iodine that was being injected from iodine chamber and this air-gas mixture was then passed onto the scrubbing column containing scrubbing solution. Samples were taken from both inlet and outlet, and they were analyzed by using UV-VIS spectroscopy for estimation of removal efficiency. Bubble column was around 2 m high, and its diameter was 0.13 m

This paper discusses the influence of different hydrodynamic parameters such as gas flowrate, liquid level height, gas holdup and different additives on retention of iodine and methyl iodide. Strong dependance of removal efficiency on gas flow rate and liquid level height was observed. Overall, removal efficiency of greater than 98% was achieved for both iodine and methyl iodide.

Email address:

aaffqq@gmail.com

Confirm that the work is original and has not been published anywhere else:

Yes

186

Uncertainty assessment of a Carem25 Passive Safety System: life-cycle management study case

Authors: Cristian Fabián Aramayo[1]; Federico Mezio[1]; Marcelo Gimenez[1]; Pablo Zanocco[1]

[1] *CNEA*

Corresponding Author: federico.mezio@cab.cnea.gov.ar

The implementation of passive safety systems in the design of advanced SMRs, depending on natural phenomena rather than active safety systems, presents many challenges, as performing a reliability assessment is not straightforward in the context of safety studies for licensing or design feedback. For instance, large uncertainties, exclusive failure mechanisms and little operational experience are a few of the issues to be conveniently addressed, in order to accurately model the reliability of passive safety systems.

In addition to the challenges presented in this kind of assessments, it must be considered their life-cycle management, in order to keep the results updated as system engineering evolves, thought the engineering design phases as well as during the whole plant life.

In this work it is showed one particular example of how to update one of this assessment, when the system engineering phase changed and more information became available.

The assessment presented in this work analyze the uncertainties impact on the performance of the

Medium Pressure Injection System of Carem25 reactor, during a Small Break LOCA combined with a Station Black-Out.

Email address:

federico.mezio@cab.cnea.gov.ar

Confirm that the work is original and has not been published anywhere else:

yes

187

Advanced Computer Vision Machine Learning for Automatic Microstructural Analysis

Author: Edmanuel Torres[1]

[1] *Canadian Nuclear Laboratories*

Corresponding Author: edmanuel.torres@cnl.ca

Microstructural characterization plays an important role in the assessment of materials properties, which is essential to ensure the performance and service life of the materials and components. The number of techniques for imaging experiments is continuously improving and increasing. While the analysis of microstructural images traditionally involves a human criteria for the detection and quantification of features, recent advances in computer vision (CV) and machine learning (ML) offer new approaches for extracting information from microstructural images. Supervised ML techniques require dataset of annotated images for training. The annotation is usually performed by humans. However, humans are a source of bias in the data, and also the datasets are becoming larger and larger, with a vast number of possible annotations. In this work, we present and CV technique for feature detection and annotation. It demonstrated that a fully automated processing and training based on CV and ML enable the efficient and accurate assessment of microstructural images of materials.

Email address:

edmanuel.torres@cnl.ca

Confirm that the work is original and has not been published anywhere else:

YES

188

DEVELOPMENT OF MULTIPHYSICS MODELLING CAPABILITIES FOR SMALL MODULAR REACTORS IN CANADA

Author: Krishna Podila[1]

Co-authors: Qi Chen [1]; Waddington Geoff [1]; Xianmin Huang [1]; Alexandre Trottier [1]; David Wang [1]; Ali Siddiqui [1]; Peter Pfeiffer [1]

[1] *Canadian Nuclear Laboratories*

Corresponding Author: krishna.podila@cnl.ca

Pursuant to Canada's small modular reactor (SMR) action plan to achieve net-zero emissions, a number of SMR reactor technologies are under consideration for deployment in Canada. Using a technology agnostic approach, Canadian Nuclear Laboratories (CNL) is developing an integrated toolset that loosely couples computational fluid dynamics (CFD), system thermalhydraulics (TH) and neutronics codes to handle a range of multi physics capabilities and model a range of transient scenarios for various SMR concepts. This paper provides an overview of the capabilities of a new toolset that is being developed at CNL under the Federal Nuclear Science and Technology (FNST) program. The toolset has been demonstrated through steady-state and transient simulations of a prismatic block gas-cooled reactor (GCR) and a molten salt reactor (MSR). A step wise approach was undertaken in which the models were first tested using a stand alone code, followed by coupled simulations. A unique aspect of this study is the assessment of coupled analyses (CFD + system TH) against integral test data from the Oregon State University High Temperature Test Facility (HTTF, a scaled facility based on the General Atomics Gas-Turbine Modular High Temperature Reactor). Overall, the results obtained are satisfactory, and demonstrate the potential of the CNL toolset for the analysis of normal and upset conditions of GCR and MSR concepts.

Email address:

krishna.podila@cnl.ca

Confirm that the work is original and has not been published anywhere else:

YES

189

Development of experimental and modelling capabilities and tools at Canadian Nuclear Laboratories for investigations on inherent and passive safety designs of advanced reactor concepts

Author: Chukwudi Azih[1]

[1] *Canadian Nuclear Laboratories*

Corresponding Author: chuk.azih@cnl.ca

A robust technical program that supports the research, development, and licensing activities is an underpinning platform to meet Canada's ambitions for deployment of Small Modular Reactors (SMR). Canada aims to achieve these ambitions through technical, environmental, economic and policy landscapes as outlined in the Canadian SMR Roadmap and Canadian SMR Action Plan. Although there are still uncertainties with respect to the technologies which will be deployed, CNL proactively pursued research activities to support a broad technical program. Once such activity relates to the development of experimental and modelling capabilities for the assessment of passive and inherent safety systems for decay heat removal in postulated accident scenarios, which feature in various SMR designs. Particular experimental and modelling capabilities being developed pertain to natural convection heat removal with molten salts as the working fluid, including investigations with self-heating fluids that simulate fuel salts. Likewise, heat removal with heat pipes, interactions and inherent instabilities of multiple-coupled natural circulation loops, and the effectives of radiation heat transfer from graphite samples are pursued as areas of interest. In addition, instrumentation technologies for such systems are investigated, which often involve novel challenges due to chemical, thermal, or flow environments. Facilities have been developed to target generic systems that cover a broad range of heat removal systems for accident scenarios such as: natural convection driven flows got Reactor Cavity Cooling Systems (RCCS), Reactor Vessel Auxiliary Cooling Systems (RVACS), Direct Reactor Auxiliary Cooling systems (DRACS). Also of interest is the performance of cores that are cooled by Alkali-metal heat-pipe components during concurrent loss of cooling capacity of multiple heat pipe units. The facilities are also intended to investigate particular phenomena related to experiments such as: the effect of intrusive instrumentation on low-flow measurements and phenomenological differences in experiments that use convectional fluids to simulate the behaviour of self-heating fluids. The progress, lessons learned, and preliminary outcomes of several capabilities being pursued simultaneously are described in this paper.

Email address:

chuk.azih@cnl.ca

Confirm that the work is original and has not been published anywhere else:

YES

190

Revival of a Practical Sodium Safety Culture through Experiential Training

Author: Daniel LaBrier[1]

Co-authors: Wade Marcum [2]; Chad Pope [1]

[1] *Idaho State University*

[2] *Oregon State University*

Corresponding Author: labrdani@isu.edu

In the United States of America (U.S.), sodium-cooled fast-neutron reactors (SFRs) are currently being pursued as candidates for next-generation nuclear power production, as part of the Generation IV initiative. Successful operations of SFR prototypes have been demonstrated in the past in the United States, such as the Experimental Breeder Reactor-II and Fast Flux Test Facility. Both of these programs were scuttled in the 1990s, however, which has led to the eventual decay of experiential knowledge related to handling molten sodium and their associated systems relative to SFRs. With the advent of the concept of the Versatile Test Reactor (VTR), a proposed pool-type SFR design that would be utilized as a test reactor for advanced materials and safety analyses, this gap in institutional knowledge related to handling sodium has been further exposed.

As part of the VTR program initiative, several U.S. universities were tasked with designing and developing a variety of necessary components to revive relevant SFR technology in the U.S. One of these tasks involved the development of a "unique and comprehensive safety culture program…associated with the working fluid of the VTR." In this paper, the developers of that program discuss the efforts that went into producing such a product that would be utilized by a diverse group of technical workers, along with an associated hands-on training program for students, technicians, and scientists that was launched in 2021.

Email address:

labrdani@isu.edu

Confirm that the work is original and has not been published anywhere else:

NO (some introductory material was presented at an ANS conference in 2020)

191

A study report on the identification and evaluation of possible regulatory challenges towards licensing and regulation of small modular reactors in India

Authors: Abhinav Kumar[1]; Tanay Walhekar[2]; Vivek Gupta[2]; Utkarsh Chikkanagoudar[2]

[1] *Atomic Energy Regulatory Board*

[2] *Atomic Energy Regulatory Board, India*

Corresponding Author: abhinavkumar@live.in

Atomic Energy Regulatory Board (AERB) framework for licensing as well as regulation of nuclear power plants (NPP) is developed comprehensively for water cooled reactor based NPPs and to a large extent is also technologically neutral which allows its implementation towards licensing as well as regulation of different types of nuclear reactor technologies. With the advent of new age small and modular power reactors (SMR) and the advancement in their licensing globally, a need has arisen to map the current AERB regulatory framework and practices to SMRs which are currently under advance stages of deployment/ licensing/ design globally, so as to identify major regulatory challenges that SMRs might present during their licensing as well as further regulation.

The majority of SMR technologies which are currently being pursued towards deployment globally are seen to be predominantly Light Water Reactor variants which are either scaled down from existing large NPP designs or are entirely new concepts. Several other SMR variants are also seen to be employing liquid metal coolants, molten salt modules, high temperature gas cooled modules etc. Some of the possible challenges identified under the current study and those identified by various international organisations and bodies such as, International Atomic Energy Agency etc. as well as by various national regulatory bodies were evaluated. Some of the areas which were studied in order to ascertain and evaluate the possible regulatory challenges presented by them include; regulatory oversight, use of advanced passive safety features, manpower requirement, operating experience feedback, siting, emergency response planning boundary, application of graded approach, SMR design specific requirements, safety assessment as well as research and development aspects etc.

It is seen that to a large extent, the available AERB regulatory framework can effectively be implemented towards licensing as well as regulation of water cooled SMR technologies. In addition, AERB is developing regulations for sodium cooled fast reactor based NPPs which can further be utilized towards licensing and regulation of various fast reactor based SMR technologies. However, there are certain specific aspects dependent on the technology being used that need a detailed review and assessment. For such aspects, there might be a need to develop technology specific requirements and guidance which would require a systematic approach and extensive expert consultation as well as engagement with stakeholders.

Email address:

abnv@aerb.gov.in

Confirm that the work is original and has not been published anywhere else:

YES

192

PRELIMINARY GAP ANALYSIS OF ARMENIAN REGULATORY BASIS FOR THE LICENSING OF SMR TYPE REACTORS

Authors: Armen Amirjanyan[1]; Surik Bznuni[2]

[1] *AA*

[2] *Leading Specialist, Nuclear and Radiation Safety Center*

Corresponding Author: s.bznuni@nrsc.am

The National Strategy of development of the energy sector of Armenia mandates keeping nuclear energy as the base load in the energy mix. Due to the small capacity of the Armenian grid, the Strategy envisages replacing existing NPP with SMR type of reactors. Within this work, a preliminary

gap analysis of Armenian legislation was carried out against IAEA SMR Forum working group reports as well as IAEA TECDOC 1936 suggestions and recommendations on the example of NuScale SMR.

Email address:

s.bznuni@nrsc.am

Confirm that the work is original and has not been published anywhere else:

Yes

193

Risk-Informed Safety Strategy and Fault Evaluation for the GE Hitachi BWRX-300 Advanced Reactor

Authors: Dennis Henneke[1]; Shelby Small[1]

[1] *GE Hitachi Nuclear Energy*

Corresponding Authors: dennis.henneke@ge.com, shelby.small@ge.com

GE-Hitachi is in the design stage of a small, simplified Boiling Water Reactor (BWR), named BWRX-300 (~300 MWe). The design is the latest iteration in GE's history of BWR development and introduces unique design features to address many significant risk contributors associated with the operating fleet of BWRs. The BWRX-300 generally employs a passive approach to reaching a safe-and-stable state in response to initiating events. This is achieved, in part, through isolation of the Reactor Pressure Vessel (RPV) from attached piping using valves directly connected to the RPV. Additionally, a robust, fail-safe Isolation Condenser System (ICS) activates to provide passive cooling to the reactor without any need for inventory makeup or active decay heat removal. This design addresses several major sources of risk-significant sequences traditionally analysed for BWRs: losses of power and loss of coolant accidents, by greatly reducing the frequency of classically important sequences to BWRs and therefore the overall plant risk.

The BWRX-300 design is supported by a risk-informed safety strategy and fault evaluation process consistent with the IAEA process described in IAEA GSR Part 4, "Safety Assessment for Facilities and Activities", and SSR-2/1, "Safety of Nuclear Power Plants: Design." The fault evaluation determines which plant scenarios will be included in the deterministic safety analysis (DSA) and the severe accident analysis (SSA). In general, these fault evaluations start in parallel with or ahead of hazard evaluations and PSA activities, and result in initial assignments of plant functions to defence lines to facilitate design maturation prior to availability of mature PSA models.
As hazard and PSA evaluations mature with the design, the fault evaluation is reconciled with the postulated initiating event (PIE) lists from the hazard and PSA evaluations, and the complex and severe accident sequences identified in the PSA results. Complex sequences include PIEs with failures of items relied on for safety as well as PIEs involving common cause failures. Quantitative frequencies are adopted as the final, governing measure of the category to which an event sequence is assigned (e.g., AOO, PA, or DEC). These frequencies are determined based on Level 1 PSA results, when modelled in the PSA. The PSA is also used to support the selection of the defence line measures credited in the safety analysis. Additionally, the PSA is used to confirm the adequacy of the independence of defence line measures in the design.

The resulting risk-informed safety analysis helps ensure the analysis is comprehensive and complete, while removing extraneous conservatism in the category assigned to the PIEs. The interface between the safety analysis and PSA also ensures the developed safety case meets the requirements in IAEA GSR Part 4.

Email address:

dennis.henneke@ge.com

Confirm that the work is original and has not been published anywhere else:

Yes

194

Approach to Small Modular and Advanced Reactor Regulatory Readiness

Author: Tarek Tabikh[1]

[1] *Canadian Nuclear Safety Commission*

Corresponding Author: tarek.tabikh@cnsc-ccsn.gc.ca

SMR deployment potential in Canada has accelerated towards reality with recent industry announcements on technology selection and significant progress towards a licence to construct (LTC) application expected by the end of 2022. There has also been an increasing rise in interest in SMRs and advanced technologies across Canada at unprecedented levels, including SMR projects in the provinces of Alberta, Ontario, New Brunswick, Saskatchewan, and transportable and marine-based power plants. The Canadian Nuclear Safety Commission (CNSC) has been preparing for the regulation and licensing of SMRs and advanced reactors for over a decade and is continuing to invest efforts towards readiness activities. This paper discusses the ongoing and future planned readiness activities including regulatory refinement, ramping up capabilities, addressing first to a kind aspects, and international collaboration activities with the UK Office of Nuclear Regulation, the US Nuclear Regulatory Commission, and the Nuclear Energy Agency.

Email address:

tarek.tabikh@cnsc-ccsn.gc.ca

Confirm that the work is original and has not been published anywhere else:

YES

195

KORSAR/GP Validation Based on the Experiments of OECD/NEA PKL-4 Project

Authors: Mikhail Suslov[None]; Ivan Petkevich[None]

Corresponding Author: suslov_mv@mail.ru

The "PKL Phase 4" project was held between June 2016 and September 2020 within the framework of Nuclear Energy Agency of Organization for Economic Cooperation and Development (OECD/NEA) international cooperation. Many participant organizations from more than 12 different countries took part in the project activities. The project included three experimental programs held at PKL test facility (Framatome GmbH, Erlangen, Germany), PMK test facility (MTAEK, Budapest, Hungary), PACTEL test facility (LUT, Lappeenranta, Finland). Participants used a lot of different thermal-hydraulic system codes for simulating of the experiments: ATHLET, CATHARE, KORSAR/GP, RELAP5, SOCRAT and others.
OKB Gidropress JSC, being an organization of ROSATOM, took part in the project activities with Russian system code KORSAR/GP. This paper presents the main results of KORSAR/GP validation

based on the experimental program of PKL facility. The plenty of scenarios were simulated with use of KORSAR/GP, the following ones are among them: IBLOCA, parametric study with nitrogen injection, facility cooldown in loss-of-power conditions, 2Φ-flow investigations during different LOCAs, and others. In general, obtained results showed a good agreement with experimental data and expanded the validation base of KORSAR/GP code.

Keywords: Best Estimate Code, KORSAR, PWR, System Code, Thermal-hydraulic, Validation, VVER.

Email address:

suslov_mv@grpress.podolsk.ru

Confirm that the work is original and has not been published anywhere else:

YES

196

Steps Towards Efficiently Demonstrating Adequate MSR Safety

Author: David Holcomb[1]

[1] *Oak Ridge National Laboratory*

Corresponding Author: holcombde@ornl.gov

The substantial technical differences of liquid-salt fueled nuclear power plants from previously licensed reactors results in significant divergence in the evidence necessary to demonstrate adequate protection of the health and safety of the public and the environment. The US Government is supporting a diverse set of activities to improve the efficiency and effectiveness of molten salt reactor safety adequacy evaluation. Key current projects include developing a fuel salt thermophysical and thermochemical properties database and performing integral and separate effects experiments to validate accident progression models.

Email address:

HolcombDE@ornl.gov

Confirm that the work is original and has not been published anywhere else:

YES

197

Innovative Regulatory Approach for Licensing a new Nuclear Power Plant

Authors: Al Falahi Rashid [1]; Raoul Awad[1]

[1] *FANR*

Corresponding Authors: raoul.awad@fanr.gov.ae, rashid.alfalahi@fanr.gov.ae

September 2009, Federal Authority for Nuclear Regulation (FANR) was officially established. Since that date, FANR has made significant progress towards establishing FANR as world-class and globally recognized nuclear safety, security and safeguards regulator. The guiding principles of FANR are based on the principles of the Policy of the United Arab Emirates on the Evaluation and Potential Development of Peaceful Nuclear Energy (Nuclear Policy). FANR has continued its strong regulatory oversight of the UAE's nuclear programme in accordance with Federal Law by Decree No. 6 of 2009, Concerning the Peaceful Uses of Nuclear Energy (Nuclear Law).
FANR has achieved remarkable success in regulating the peaceful nuclear programme through transparency in its operations, dedication to building the capacities of Emiratis in the nuclear sector, and development of effective regulations to oversee the nuclear sector in the UAE. In addition, FANR has gained international recognition as a competent regulatory body and for its close cooperation with the International Atomic Energy Agency (IAEA).

In 2012, a decision was made by the FANR Board of Management to issue a construction licence for the Barakah Nuclear Power Plant Units 1 and 2. In 2014, FANR approved a licence application to begin constructing Units 3 and 4 at the Barakah Nuclear Power Plant. In 2020 and despite the Covid-19 pandemic, FANR issued the first operation licence to operate reactor Unit 1 followed in 2021 by the operation licence of Unit 2.

Due to the fast pace of the development of UAE peaceful nuclear programme and the pandemic situation, FANR had to innovate in its regulatory approach to licensing and oversight of the activities at Barakah Nuclear Power Plant. This paper presents how FANR used the innovative approach to review of the Licence applications and to assure the proper regulatory oversight during the construction and the operation of the first two units.

Email address:

raoul.awad@fanr.gov.ae

Confirm that the work is original and has not been published anywhere else:

YES

199

An Experimental Program for ACP100 Passive Containment Air Cooling System

Authors: Ji Xing[1]; Yidan Yuan[1]; Xu Han[1]; Mingrui YU[1]; Zhuo Liu[1]; Hongliang Wang[1]; Yu Feng[1]

[1] *China Nuclear Power Engineering Co.,Ltd*

The passive containment air cooling system (PAS) is one of the key engineering safety features in the ACP100 design, a small modular reactor developed by China National Nuclear Corporation (CNNC). The function of the system is to removal heat released to the containment following postulated design basis accidents that result in containment heatup and pressurization. The system employs passive air cooling to transfer heat from the steel containment to the environment. In order to assess the PAS heat removal capacity, an experimental program is introduced which includes the following series of tests: (i) Heated Plate Test; (ii) Wind Tunnel Test; (iii) Air Flow Path Pressure Drop Test; (iv) Large Scale Passive Containment Cooling System Test. All the tests were to provide test data for use in developing and verifying analytical models used in the analysis of the ACP100 containment air cooling system.

Email address:

yumingrui2010@sina.com

Confirm that the work is original and has not been published anywhere else:

YES

200

Feedback of HTR-PM demonstration project in China and safety improvement for future HTGR application

Authors: Bin Chen[1]; Qi Shi[None]

[1] *Engineer*

Corresponding Author: toyarmybin@gmail.com

HTR-PM demonstration project has reached several milestones in recent years with the first power generation last year. The experience accumulated and feedback through manufacturing, construction, commissioning as well as experiments performed on the reactor, steam generator, turbines and etc for dedicated tests are vital for the improvement of HTGR safety in its further application as a comprehensive nuclear energy system with a more competitive cost and more flexible utilization. The safety enhancement of HTGR development and application of not only power generation but also supplying process heat are discussed to present the challenge and potential of this technology for massive deployment as a carbon-free energy source.

Email address:

toyarmybin@gmail.com

Confirm that the work is original and has not been published anywhere else:

YES

201

Example for paper submission

Authors: Francisco Parada[1]; Laura VIDELA[1]

[1] *IAEA*

Corresponding Author: f.parada-iturria@iaea.org

a

Email address:

Confirm that the work is original and has not been published anywhere else:

202

Insights on Application of Some Probabilistic Considerations for Licensing of New Nuclear Power Plants

Author: Irina Kuzmina[1]

Co-authors: D. Noskov ; D. Simin ; S. Afonin ; Y. Aleksyeyev ; K. Avdjiev

[1] *RAOS project OY*

Corresponding Author: irina.kuzmina@rosatom.fi

The paper discusses some insights dealing with interpretation of IAEA requirements concerning the use of Probabilistic Safety Assessment (PSA) and risk-related aspects in the context of licensing new NPPs (Generation III+) with advanced safety features providing increased redundancy and diversity in safety systems (e.g. VVER-1200 NPP, the Russian Federation) that contributes to achieving very low risk metrics such as Core Damage Frequency (CDF) and Large Release Frequency (LRF). Specifically, the paper is concerned with the applicability of IAEA and national regulatory requirements focused on demonstration of a balanced risk profile in the specific case when the risk profile is dominated by irreducible contributors (i.e. initiating events contributing to CDF/LRF) such as Reactor Pressure Vessel rupture or an extreme seismic event. Another issue discussed in the paper is concerned with interpretation of the term 'practically eliminated conditions' often used in IAEA Safety Standards (e.g. SSR2/1 Rev.1, GSR4 Rev.1, etc.). The insights presented in the paper are largely based on practical work dealing with licensing activities accomplished in the framework of Hanhikivi project and might be especially useful in the context of the ongoing work on updating IAEA Safety Guides on PSA (SSG-3 and SSG-4) as well as for future revisions of IAEA Safety Requirements publications.

Email address:

irina.kuzmina@rosatom.fi

Confirm that the work is original and has not been published anywhere else:

YES