The 11th European Review Meeting on Severe Accidents Research

May 13-16, 2024

KTH Royal Institute of Technology, Stockholm, Sweden

BOOK OF THE ABSTRACTS

11TH EUROPEAN REVIEW MEETING ON SEVERE ACCIDENTS RESEARCH

May 13-16, 2024 – Stockholm, Sweden

ERMSAR2024

Funded by the European Union

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FOREWORD

The 11th edition of the ERMSAR (European Review Meeting on Severe Accident Research) Conference, will be held from 13 to 16 May 2024 in Stockholm, hosted and locally organized by KTH, after 2 years for the previous 10th edition in Karlsruhe. For the first time it has been organized in the frame of the EURATOM SEAKNOT project and the Technical Area 2 of the NUclear GENeration II and III pillar (NUGENIA TA2) of SNETP Association, together with IAEA and OECD/NEA.

The Technical Programme Committee involved 19 researchers from diverse organizations (BT, CEA, CIEMAT, ENEA, Framatome GmbH, GRS, IAEA, INRN, IRSN, JSI, KIT, KTH, LGI, NEA and University of Pisa).

ERMSAR 2024 will gather about 146 participants from nearly 77 organizations settled in 26 countries worldwide (EU, USA, Canada, Republic of Korea, Brazil, Japan, India), which highlights ERMSAR as the reference international conference on Severe Accidents. 71 papers will be orally presented and 33 more will be exhibited in the posters stand. The conference is focused on the latest progress of international knowledge on severe accidents and is mainly an opportunity for researchers to discuss about future R&D priorities in this field. The Conference topics are highlighted in the following technical session list.

1. In-vessel corium and debris coolability
2. Ex-vessel corium interactions and coolability
3. Containment behavior incl. H2 explosion risk
4. Source term issues
5. Analysis, Management, and Consequences of Severe Accidents for Gen I/III reactors
6. Severe accidents in innovative reactor concepts such as Gen IV and Small Modular Reactors (SMRs)
7. Other applications (fusion reactors, interim SNF storage, Accident Tolerant Fuels (ATFs), etc.

In addition, the technical programme includes three plenary sessions, in addition to specific presentation on SEAKNOT and NUGENIA/TA2 latest activities and status:

- Singular careers in SA research
- Regulatory perspective and approaches for Severe Accidents in Small Modular Reactors
- Looking ahead in severe accident research

Next edition ERMSAR2026 will be again organized in the frame of the SEAKNOT project by CIEMAT from the 18th to the 23rd of May 2026, in Madrid.

Editors: Fabrizio Gabrielli (KIT), Luis E. Herranz (CIEMAT), and Sandro Paci (University of Pisa)

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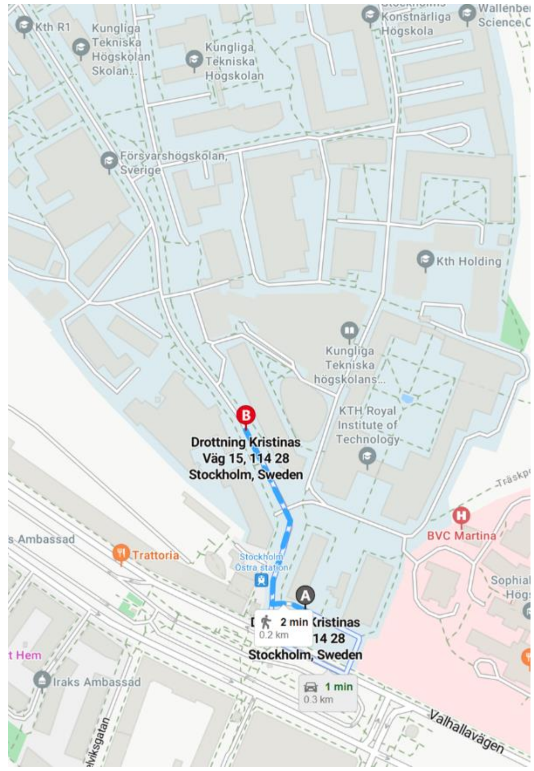
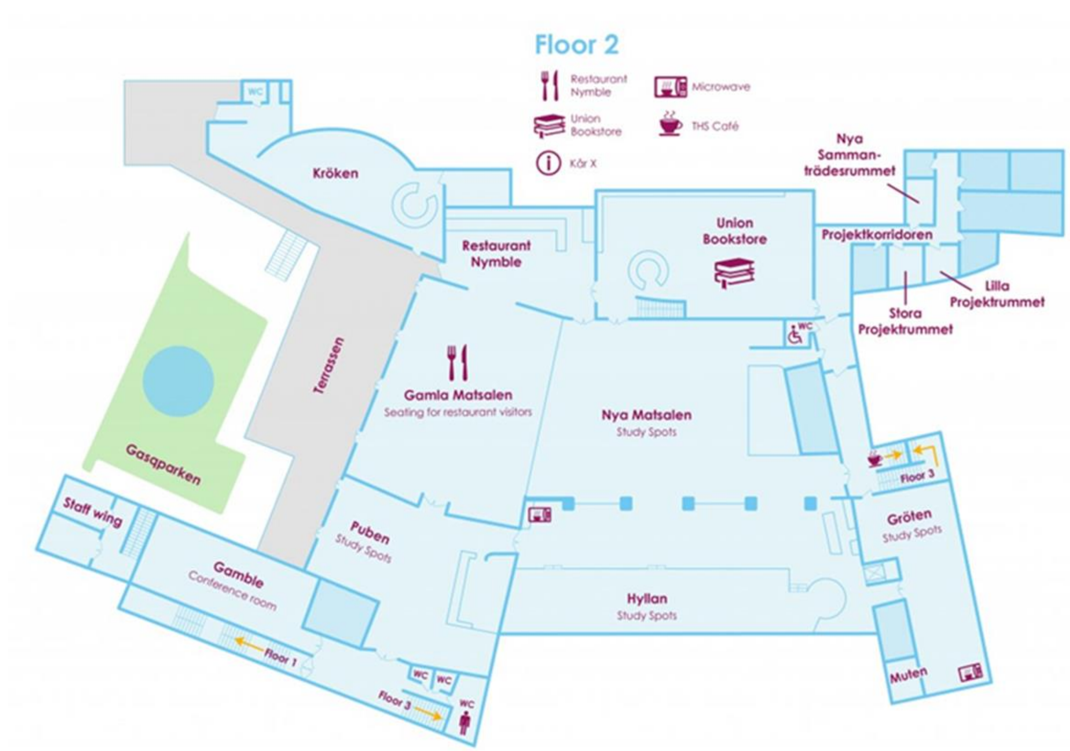
VENUE

Nymble is located a short distance from the Tekniska högskolan subway station on Drottning Kristinas väg 15-19, 114 28 Stockholm.

On October 11, 1930. THS's new Student Union Building, Nymble, was inaugurated on Drottning Kristinas Väg. Rumours said that the old union building was called "Gamble" and in connection with the construction of the new union building a student-like proposal was to call it "Nymble".



CONFERENCE ROOMS



Organization

Technical Program Committee

- Luis E. Herranz (Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas, CIEMAT, Spain)
- Fabrizio Gabrielli (Karlsruhe Institute of Technology, KIT, Germany)
- Pascal Piluso (Commissariat à l'énergie atomique et aux énergies alternatives, CEA, France)
- Sami Strola (LGI Sustainable Innovation, France)
- Ivo Kljenak (Jozef Stefan Institute, IJS, Slovenia)
- Sanjeev Gupta (Becker Technologies, Germany)
- Federico Rocchi (Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile, ENEA, Italy)
- Sandro Paci (University of Pisa, Italy)
- Ahmed Bentaib (Institut de Radioprotection et de Sûreté Nucléaire, IRSN, France)
- Sevostian Bechta (Royal Institute of Technology, KTH, Sweden)
- Victor H. Sanchez Espinoza (Karlsruhe Institute of Technology, KIT, Germany)
- Walter Tromm (Karlsruhe Institute of Technology, KIT, Germany)
- Thorsten Hollands (Gesellschaft für Anlagen- und Reaktorsicherheit, GRS, Germany)
- Pavlin Groudev (Institute for Nuclear Research and Nuclear Energy, Bulgaria)
- Markus. Hupp (Framatome GmbH, Germany)
- Alexei. Miassoedov (IAEA, Austria)
- Stephan. Brumm (European Commission's Joint Research Centre, Netherlands)
- Martina Adorni (Nuclear Energy Agency, France)
- Didier Jacquemain (Nuclear Energy Agency, France)

Organizing committee

- Patrick Isaksson (Swedish Radiation Safety Authority, SSM, Sweden)
- Nils Sandberg (Swedish Radiation Safety Authority, SSM, Sweden)
- Sean Roshan (Royal Institute of Technology, KTH, Sweden)
- Marta Marko-Tisch (Royal Institute of Technology, KTH, Sweden)

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May 13-16, 2024

KTH Royal Institute of Technology, Stockholm, Sweden

Technical Program

MONDAY, MAY 13th, 2024

8:00 Registration

WELCOME AND OPENING (Room Nya Matsalen)

Chairs: S. Bechta (KTH), F. Gabrielli (KIT)

09:00	**Welcoming Addresses** S. Östlund, Vice President of the KTH
09:10	M. Knochenhauer, Director General, Swedish Radiation Safety Authority (SSM)
09:20	**Opening of the ERMSAR2024 Conference** L.E. Herranz (CIEMAT), Coordinator of the SEAKNOT project

PLENARY SESSION: 'SINGULAR CAREERS IN THE SEVERE ACCIDENT RESEARCH' (Room: Nya Matsalen)

Chairs: L. E. Herranz (CIEMAT), T. Lind (PSI)

09:30	**Reflections on a career in reactor safety: Severe Accident Code Development** Dr Randall Gauntt (MELCOR Code Development Manager, Retired Sandia National Laboratories)
09:50	**Progress in Predicting Reactor Pressure Vessel Failure** J. Rempe (Rempe and Associates, LLC)
10:10	**Your Research Maps: Needs and challenges with scaling consideration** H. Nakamura (JAEA)

10:50 Coffee break

	Room: Nya Matsalen	Room: Gröten
	Session 2.1 Severe Accident Scenarios	**Session: 1.1 In-vessel Corium and debris coolability**
	Chairs: S. Gupta (Becker Tech. GmbH), D. Jacquemain (OECD/NEA)	**Chairs: F. Gabrielli (KIT), A. Miassoedov (IAEA)**
11:10	**SEAKNOT: Looking Ahead of Severe Accident Research** L. E. Herranz (CIEMAT)	**IVMR Modelling with Transient Effects during Molten Pool Formation and Stabilization – Outcomes from Models' Comparison Performed in the IAEA CRP J46002** L. Carénini (IRSN)
11:35	**ASTEC core degradation calculations in support of Level-2 Probabilistic Safety assessment for 1300MWe French reactors: methodology and preliminary results** M. Monestier (IRSN)	**Numerical Analysis of Melt Penetration Behavior in the Control Rod Drive Housing of Fukushima Daiichi Nuclear Power Station Unit-2** X. Li (JAEA)
12:00	**MELCOR Analyses for Investigation on Hydrogen Management during BWR Severe Accidents with the Filtered Containment Venting System** Y. Kojima (Waseda University)	**Reactor pressure vessel integrity during severe accident with core meltdown: characterization of material parameters, structure integrity assessment and thermal-hydraulic assessment** P. Gal (UJV)
12:25	**On the progress made in safety assessment and severe accident management as part of the French Fukushima post-accident research programme** A. Bentaib (IRSN)	**Improvement of thermochemical corium stratification accounting for uranium and zirconium composition difference in metallic and oxide phases** R. Le Tellier (CEA)

12:50	**Lunch break**

	Room: Nya Matsalen	Room: Gröten
	Session 2.2 Severe Accident Scenarios	**Session: 1.2 In-vessel Corium and debris coolability**
	Chairs: A. Bentaib (IRSN), L. E. Herranz (CIEMAT)	**Chairs: L. Carénini (IRSN), T. Hollands (GRS)**
14:30	**Horizon Euratom ASSAS project: Can machine-learning make fast and accurate severe accident simulators a reality?** B. Poubeau (IRSN)	**Stainless Steel Oxidation at both Solid and Liquid State Under Ar-H_2O Gas Mixture in Severe Accident Conditions** M. Nasselahsen (CEA)
14:55	**Preliminary Strategies for Training Dataset Generation and Surrogate Modeling of SBO Management Measures in PWR** C. D'Alessandro (PSI)	**A Thermodynamic Study of Molten Pool Stratification Morphology** E. Chen (China Nuclear Power Eng. Co.)

15:20	**Analysis of accident progression behavior and simulation of FP release into the environment using MELCOR code for the Fukushima Daiichi Nuclear Power Station Units 1 to 3** M. Himi (CRIEPI)	**In-Vessel Corium Thermochemistry Benchmark based on MASCA Experimental Data** R. Le Tellier (CEA)
15:45	**Long-term severe accident management at Loviisa NPP** M. Harti (Fortum Power and Heat Oy)	**Coolability of a Corium Pool in a Debris Bed – Calculation of Critical Heat Flux (CHF) Tests with the ASTEC Code** J. A. Zambaux (IRSN)

16:10 Coffee break

	Room: Nya Matsalen	**Room: Gröten**
	Session 2.3 Severe Accident Scenarios: U&S analysis	**Session 4.1 Severe Accident Scenarios: Small Modular Reactors**
	Chairs: O. Coindreau (IRSN), S. Paci (University of Pisa)	**Chairs: F. Gabrielli (KIT), D. Jacquemain (OECD/NEA)**
16:30	**Major Achievements of the EC MUSA Project** L. E. Herranz (CIEMAT)	**SASPAM-SA: Assessment of the relevance and applicability of existing experimental databases to iPWR** T. Lind (PSI)
16:55	**Source Term Uncertainty Analysis of Filtered Containment Venting Scenarios in Nordic BWR** S. Galushin (Vysus Group)	**Numerical Investigation of Natural Circulation Inside a Scaled-Down Prismatic Modular Reactor By RHYS** M. Shewitah (Minia University)
17:20	**Uncertainty and Sensitivity Analysis of the ASTEC Source Term Results of a MBLOCA Scenario with the Activation of Severe Accident Management Actions in a Generic KONVOI Plant** A. Stakhanova (KIT)	**Development of a LW-SMR dry containment model with containmentFOAM** C. Vázquez-Rodríguez (FZJ)
17:45	**Source Term Uncertainties in unmitigated SBO sequences in a PWR-1000: Insights from the EU-MUSA project** R. Iglesias (CIEMAT)	**Comparison of a DBA sequence in a generic iPWR between MELCOR and ASTEC codes** G. Grippo (ENEA)

18:10 Adjourn

TUESDAY, MAY 14th, 2024

	Room: Nya Matsalen	Room: Gröten
	Session 2.4 Severe Accident Scenarios: U&S analysis	Session 4.2 Severe Accident Scenarios: Small Modular Reactors
	Chairs: M. Angelucci (University of Pisa), L. E. Herranz (CIEMAT)	Chairs: T. Lind (PSI), A. Bentaib (IRSN)
09:00	**Synthesis of Source Term Assessments for a Loss-Of-Cooling Accident in a Spent Fuel Pool: Uncertainty and Sensitivity Analyses and Potential Benefit of Water Injection by Spray System** O. Coindreau (IRSN)	**Analysis of Postulated Severe Accidents in Generic Integral PWR Small Modular Reactors in the frame of the Horizon Euratom SASPAM-SA Project** F. Gabrielli (KIT)
09:25	**Uncertainties on Fission Product Release for a Loss-Of-Cooling Scenario in a Spent Fuel Pool with MELCOR2.2** M. Garcia (CIEMAT)	**Update on Severe Accident Analysis Research at CNL for Small Modular and Advanced Reactor Designs** A. Morreale (CNL)
09:50	**Uncertainty and Sensitivity Analyses of Severe Accident Codes Using the ACE Algorithm-based Surrogate Model** Kwang-Il Ahn (KAERI)	**Application of the Probabilistic Method to Propagate Input Uncertainty on a DBA Sequence in a Generic iPWR** G. Grippo (ENEA)
10:15	**System Identification and Ranking Table (SIRT) for chemical thermodynamics of severe accidents** C. Journeau (CEA)	**Comparison between EDF MAAP5.04 and ASTECv3 codes on an hypothetical Severe Accident on the ELSMOR project NUWARD-Like SMR Design** J. Bittan (EdF)

10:40 Coffee break

	Room: Nya Matsalen	Room: Gröten
	Session 3.1 Ex-vessel corium interactions and coolability	Session 6.1 Hydrogen risk and Containment behavior
	Chairs: L. Carénini (IRSN), M. Hupp (Framatome GmbH)	Chairs: I. Kljenak (IJS), S. Gupta (Becker Tech. GmbH)
10:55	**Results of the SSM-SICOPS melt tests in the frame of the EU-SAFEST project** G. Langrock (Framatome GmbH)	**AMHYCO Project Overview and First Outcomes** G. Jiménez (UPM)
11:20	**Ex-Vessel stabilization of corium: An analysis of corium-concrete interaction with top flooding for siliceous concrete** F. Fichot (IRSN)	**Outcomes of the experimental and numerical work on the operational behavior of passive autocatalytic recombiners in the late phase of a severe accident in the framework of the AMHYCO project** E.-A. Reinecke (FZJ)

11:45	**Fluids Mixing Modelling with Phase Change for Molten Corium-Concrete Interaction** I. Khurshid (Khalifa University)	**Assessment of Unmitigated Combustion Risk in Late Phase Within the AMHYCO Project** S. Kelm (FZJ)
12:10	**Characterisation of prototypic ex-vessel fuel debris simulating MCCI at Fukushima Daiichi** C. Journeau (CEA)	**Heat Removal to Large Water Pools – Macroscopic Modelling of Microscopic Phenomena in the Simulation Code AC^2/COCOSYS** C. Spengler (GRS)

12:35 Lunch break

PLENARY SESSION: 'REGULATORY PERSPECTIVE AND APPROACHES FOR SEVERE ACCIDENTS IN SMALL MODULAR REACTORS' (Room: Nya Matsalen)

Chairs D. Jacquemain (OECD/NEA), A. Miassoedov (IAEA)

14:00	**U.S. NRC Regulatory Perspective and Approaches for Severe Accidents in Small Modular Reactors** Jason Schaperow, (USNRC Office of Nuclear Reactor Regulation) **ON-LINE**
14:20	**The Canadian Nuclear Regulatory Role, Approaches and Challenges for Severe Accidents in Small Modular Reactors** Samuel Gyepi-Garbrah (Canadian Nuclear Safety Commission)
14:40	**Beyond Design Basis Analysis: Regulatory Perspective on New Reactor Designs** Ali Tehrani (Office for Nuclear Regulation, UK)
15:00	**Applicability of new Swedish regulations to Small Modular Reactors – Opportunities and challenges of a performance based approach** A. O. Mowitz (Swedish Radiation Safety Authority, SSM)

15:30 Poster Session & Coffee break

MELCOR 2.2 analyses of passive systems modeling based on the PANDA facility experiments M. Malicki (PSI)
Model Development for the Simulation of Fission Product Release from Molten Pools F. Krist (PSS-RUB)
Severe Accident Sequence of LOCA for APR1400 using CINEMA Computer Code Rae-Joon Park (KAERI)
Comparative Study of the Hydrogen Distribution Among Different PWR-W Lumped-Parameter and 3D Containment Models with GOTHIC 8.3 (QA) A. García-Herranz (UPM)
Thermophysical property measurement of oxide melts using aerodynamic levitation Y. Gong (CNPE)
PLINIUS – experimental platform for nuclear excellence A. Bachrata (CEA)
Thermodynamic Evaluation of Liquid-Gas Surface Tension for U-O-Zr Mixtures Using the Butler Equation A. Tourneix (CEA)

Analysis of combustible gases distribution with accident management action in a generic PWR-W containment
J. Fontanet (CIEMAT)

Development of 3D view application debrisEye for decommissioning of Fukushima Daiichi Nuclear Power Plant
T. Yamashita (JAEA)

Thermal shock resistant geopolymers as refractory material for core catcher
B. Mészáros (UJV)

Investigation of New Inorganic Materials for Nuclear Industry under Severe Accident Conditions
J. Hrbek (UJV)

Modeling of pool scrubbing and sensitivity analysis using GOTHIC
X. Wang (KTH)

MELCOR analyses of Severe Accident sequences in an integral PWR with passive systems
F. Giannetti (Sapienza University of Rome)

Development of severe accident simulation code for sodium-cooled fast reactors: SIMMER-V (2) Development and verification of detailed fuel pin model
S. Ishida (JAEA)

Analysis of the combustion risk mitigation inside the containment during a postulated severe accident in a PWR using the code package AC2
M. Müer (PSS-RUB)

Uncertainty quantification analysis with radiological consequences for a loss of cooling accident in a spent fuel pool
M. D'Onorio (Sapienza University of Rome)

THS-15 Experimental Facility: Effect of Surface Roughness on CHF Values
D. Batek (UJV)

Cooperative Nuclear Safety Research Activities at the Nuclear Energy Agency in Response to the Fukushima-Daiichi Accident
Y. Kumagai (OECD/NEA)

Filtration efficiency of electrostatic precipitator for iodine particles in different gas atmospheres simulating the severe accident scenarios
S. Basnet (University of Eastern Finland)

Study on AP1000 accident diagnosis and treatment for loss of monitoring and control
Y. Yu (China Nuclear Power Engineering Co.)

Failure modes of the reactor coolant pressure boundary in high-pressure core melt accident scenarios
C. Bläsius (GRS)

Parametric sensitivity studies for RELAP/SCDAPSIM model of QUECH-20 test
N. Elsalamouny (LEI)

Assessment of how Zr-clad oxidation affects the speciation and release of FPs under accidental conditions
C. Riglet-Martial (CEA)

LPM vs. 3D Analysis of an In-Vessel LBLOCA Sequence using the ALMARAZ NPP GOTHIC Containment Model
C. Gabicagogeascoa-Cuesta (UPM)

Some results of the AMICO project
G. Langrock (Framatome GmbH)

Effect of the Stages of the Accumulators on THE Hydrogen Production During LOCA+SBO in BNPP VVER-1000
A. Hosseini (Shahid Beheshti University)

Comparative Study of Two Experimental Configurations with an Internal Compartment in the PANDA Facility
S. Arfinengo-del-Carpio (UPM)

Numerical Simulation of LIVE2D Two-Layer Melt Pool Experiment
P. Guo (Tsinghua University)

Severe Accident R&D in UJV Group
P. Vácha (UJV)

Investigation of IVMR Strategy for BNPP-1 VVER 1000
A. Najafi (Sharif University of Technology)

Assessment of RELAP5-3D condensation models for small modular reactor passive safety
P. K. Bhowmik (INL)

V&V of nuclear fuel oxidation behavior in sleeveless SiC-matrix during air ingress accident
Y. Nishimura (University of Tokyo)

Numerical Analyses on Melt Water Interactions with Super Absorbing Polymers Added to the Cooling Water
M. Buck (University of Stuttgart)

	Room: Nya Matsalen	**Room: Gröten**
	Session 3.2 Ex-vessel corium interactions and coolability	**Session 5.1 Severe accident scenarios: Model development and validation**
	Chairs: S. Bechta (KTH), P. Piluso (CEA)	**Chairs: M. Angelucci (University of Pisa), F. Gabrielli (KIT)**
17:00	**Production of prototypic corium in the VULCANO facility using uranothermite and induction heating** A. Denoix (CEA)	**Formulation of material property formula for calculation of damage in reactor pressure vessel during accident evaluation** K. Shimomura (JAEA)
17:25	**Overview of Ex-Vessel Severe Accident Scenarios Simulations at UJV Rez** J. Komrska (UJV)	**Correlation Development for the Determination of Aerosol Particle Retention in Liquid Pools** J. Rehrmann (PSS-RUB)
17:50	**Simulations of FLOAT debris quenching experiments** M. Uršič (JSI)	**In Vessel Melt Retention 0D model for integral Pressurized Water Reactors** M. Principato (Sapienza University of Rome)

18:15 **Adjourn**

WEDNESDAY, MAY 15th, 2024

PLENARY SESSION: 'LOOKING AHEAD IN SA RESEARCH' (Room: Nya Matsalen)

Chairs: L. Carénini (IRSN), S. Gupta (Becker Technologies GmbH)

09:00	**Incoming Euratom Research funded Projects on Severe Accident and Nuclear Safety** A. Iorizzo (European Commission)
09:10	**Recent IAEA Activities Related to Severe Accidents** A. Miassoedov (IAEA)
09:20	**Status and Perspectives in NEA Joint Nuclear Safety Research Projects in the Severe Accident Area** D. Jacquemain (OECD/NEA)

10:30	**Coffee break**

	Room: Nya Matsalen	Room: Gröten
	Session 7.1 Source term	Session 5.2 Severe accident scenarios: Model development and validation
	Chairs: O. Coindreau (IRSN), A. Bentaib (IRSN)	**Chairs: A. Stakhanova (KIT), T. Hollands (GRS)**
10:45	**The Reduction of Radiological Consequences of design basis and extension Accidents: re-assessment of calculation results and main outcomes of the R2CA project** N. Girault (IRSN)	**Assessment of pH-values in water pools during severe accidents in PWR using the lumped parameter code COCOSYS** L. Anschuetz (Framatome GmbH)
11:10	**Progress in understanding fission products remobilization behaviour and hydrogen risk in water cooled reactors under severe accident conditions: OECD/NEA THAI-3 project** S. Gupta (Becker Technologies GmbH)	**Modeling of oxidation behavior of Accident Tolerant Fuel by using AC²** G. Stahlberg (PSS-RUB)
11:35	**Source Term Assessment for a VVER-1000 Reactor Equipped with Filtered Venting: Sensitivity Study of the Impact of Different Forms of Iodine in the Containment** M. Kotouc (UJV)	**ASTEC validation of SFP Dewatering using Results from the DENOPI project** L. Laborde (IRSN)
12:00	**Source Term Dispersion Analysis and Construction of the Risk Map around the Peach Bottom Unit-2 Plant Using the ASTEC and JRODOS codes** O. Murat (KIT)	**A review of correlations of stainless steel oxidation in steam, and modeling the reaction with MELCOR** T. Sevón (VTT)

12:25	**Lunch break**

	Room: Nya Matsalen	Room: Gröten
	Session 7.2 Source term	Session 6.2 Hydrogen risk and Containment behavior
	Chairs: L. E. Herranz (CIEMAT), S. Gupta (Becker Tech. GmbH)	**Chairs: I. Kljenak (IJS), S. Kelm (FZJ)**
14:00	**Effect of Boric Acid on Fission Product Tellurium and Iodine in Severe Accident-Like Conditions: Analysis with X-Ray Photoelectron Spectroscopy** F. Börjesson Sandén (Chalmers University of Technology)	**Validation of REKO-DIREKT and ContainmentFOAM-9 Code Coupling Using THAI-HR Experiments** L. Serra Lopez (UPM)
14:25	**Local measurements on particle mass transfer in gas-liquid flows** A. Ramos Perez (PSI)	**THAI Experiment on Iodine Absorption Capacity of Pre-Stressed Paint Coatings on Different Surfaces** K. Dieter (Becker Techn. GmbH)
14:50	**New Experimental Devices for Severe Accident Study: a Laser Based Approach** Y. Pontillon (CEA)	**Validation of the PARUPM and GOTHIC 8.3 Code Coupling using THAI Hydrogen Recombination Tests** A. Domínguez-Bugarín (UPM)
15:15	**On the progress made in source terms evaluation and possible open issues relative to advanced technologies** L. Cantrel (IRSN)	

15:40 Coffee break

	Room: Nya Matsalen	Room: Gröten
	Session 7.3 Source term	Session 6.3 Hydrogen risk and Containment behavior
	Chairs: T. Lind (PSI), F. Rocchi (ENEA)	**Chairs: I. Kljenak (IJS), S. Kelm (FZJ)**
16:00	**Gaseous and aerosol formations in the pseudo-binary CsI-MoO3 reaction system** M. Rizaal (JAEA)	**Evaluation of hydrogen risk and its mitigation strategies adopted in isotope manufacturing building** C. Peng (Shanghai University)
16:25	**Experimental study on removal effect of radioactive materials in the course of the leakage through the equipment hatch** K. Nakamura (CRIEPI)	**The Scaling of Turbulent Flame Acceleration and Detonation Transition for Hydrogen-Air mixtures in the RUT Facility** M. Kuznetsov (KIT)

16:50 Adjourn

19:00 Conference Dinner

THURSDAY, MAY 16th, 2024

	Room: Nya Matsalen	Room: Gröten
	Session 9.1 Severe Accident Scenarios: Accident Tolerant Fuels	**Session 8.1 Severe Accident scenarios: Fast Reactors**
	Chairs: S. Gupta (Becker Tech. GmbH), M. E. Cazado (KIT)	**Chairs: A. Stakhanova (KIT), F. Gabrielli (KIT)**
09:00	**Overview of KIT activities on ATF cladding materials** M. Steinbrück (KIT)	**Preliminary Evaluation of Reactivity Insertion during BDBA-LOCA of Super Fast Reactor** K. Matsuoka (Waseda University)
09:25	**Application of AC²/ATHLET-CD and ASTEC for ATF Experiments in the Frame of OECD QUENCH-ATF and IAEA CRP ATF-TS** T. Hollands (GRS)	**Development of severe accident simulation code for sodium-cooled fast reactors: SIMMER-V (1) Overview of the SIMMER-V code development** H. Tagami (JAEA)
09:50	**The CODEX-ATF experiment** R. Farkas (EK-CER)	**Experimentation & simulation of ablated surface by jet impingement for core-catcher safety issue** N. Seiler (CEA)

10:15 Coffee break

CLOSING PLENARY SESSION (Room: Nya Matsalen)

Chairs: L.E. Herranz (CIEMAT), F. Gabrielli (KIT)

10:45	**Summary and conclusions of the plenary sessions** L.E. Herranz (CIEMAT), Coordinator of the SEAKNOT project
11:00	**Summary and conclusions of the technical sessions** NUGENIA TA2 Leaders
11:30	**Concluding Remarks and Introduction of ERMSAR2026** L.E. Herranz (CIEMAT), Coordinator of the SEAKNOT project F. Gabrielli (KIT)

11:40 Closing of the conference

14:30 NUGENIA TA2 Meeting (Restricted) (Room: Gröten)

FRIDAY, MAY 17th, 2024

Technical visit to the Westinghouse Thermal-Hydraulic Testing Laboratory in Västerås, https://youtu.be/Mn4aoww73IE?si=p9hiSovOr6612iPg

Note: max. 40 participants

Schedule

11:00	**Departure from Nymble**
12:45	**Arrival at Westinghouse facility in Västerås**
13:00	**Technical tour**
16:00	**Departure to Stockholm**
17:30	**Arrival at Nymble**

Reflections on a career in rector safety: Severe Accident Code Development

Dr. Randall Gauntt
MELCOR Code Development Manager, Retired Sandia National Laboratories

ABSTRACT

The MELCOR Accident Analysis code took root in the early 1980s as a next-generation Source Term Code Package (STCP) and is still under development today for use in safety studies, probabilistic safety analyses and licensing support activities. After the initial development phase, Dr Gauntt led and managed the MELCOR effort for the next 25 years, throughout a vigorous validation phase and later a code architecture modernization phase and an extension program to address new reactor concepts and designs. Dr Gauntt retired from the still-active program at Sandia National Laboratories in 2019, and the effort the effort continues under David Luxat. In this talk, Dr Gauntt will reflect on his experiences over his thirty five year career at Sandia National Laboratories as a severe accident researcher, safety analyst and code developer. Topics covered will include the importance of code architecture, multi-disciplined code developer teams, the evolution of computational platforms, international user community, international sharing of data and experiences, the code as a repository of severe accident knowledge, and evolving future roles and applications. Personal anecdotes and observations over his career will be shared.

Progress in Predicting Reactor Pressure Vessel Failure

Joy L. Rempe[1]
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In light water reactors (LWRs) and many non-LWRs, reactor pressure vessel (RPV) failure significantly affects subsequent accident progression. Failure of this important barrier to fission product release defines the initial conditions for ex-vessel phenomena. The failure timing and mode determine the fraction of melt available for release to the containment and affect the potential for subsequent phenomena (e.g., direct containment heating, fuel-coolant interactions, and molten core concrete interaction). In addition, this information provides a basis for developing severe accident prevention and mitigation actions.

Motivated by examination data from the Three Mile Island Unit 2 RPV, a 'generic' research approach (applicable for reducing many severe accident phenomena uncertainties) was applied. A key aspect of this approach was to define the range of vessel and penetration materials, geometries, and accident conditions requiring evaluation. Then, vessel failure modes were identified, and models were developed for each failure mode. Model assessments were performed. Assessment results guided what, if any, additional data should be obtained. When sufficient confidence was gained with assessment results, accident management strategies were developed. In discussing this effort, important 'generic' insights are highlighted (with examples illustrating how these insights affected this research).

Keywords: Severe Accident Phenomena, Reactor Pressure Vessel (RPV), RPV Failure

Your Research Maps: Needs and challenges with scaling consideration

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ABSTRACT

Roadmaps (RMs) for nuclear safety research and/or reactor development (R&D) have contributed to visualize the R&D needs with their background for stakeholders: research institutes, industries and regulatory bodies, and to promote human development. It is, however, uneasy to continuously conduct "rolling" to update the RM by gathering feedbacks from watching progresses in every subject in the relevant thermal-hydraulics (TH) and severe accident (SA) fields. RM is typically composed of three items: a whole view, technology maps and chronological maps, for each of various types of reactor designs. Technology maps collect and summarize background, needs and progress for all of the relevant technical subjects, with criterions to judge the extent of research needs to achieve and improve safety by means of accident management measures etc. which may suggest the size of research, thus budget to undergo the required experiments and/or analysis work. The RM may change every day in response to your R&D outcomes. RMs for SA R&D have been discussed at the past ERMSARs, aiming at sharing the research subjects important for the reactors under operation. Now, the technology maps should be greatly amended/updated in order to properly consider new subjects required or expected for emerging reactor designs including small modular reactors (SMRs). Major subjects should address verification and validation (V&V) of computer codes for the TH safety assessment and SA analyses. This involves quantitative evaluation of uncertainty, a large portion of which may arise from phenomena scaling distortions in scaled facility/material experiments. NEA/WGAMA issued a scaling SOAR in 2016. It however does not include SA phenomena, because they are under complicated boundary conditions. High-temperature core melt even destroy/change the shape of reactor structures that enclose/define the phenomena. Experimental phenomena representation is achieved mostly in scaled-down test facilities by using simulant because of difficulties to handle core melt. The extrapolation of results from the scaled facility to prototype requires quantitative evaluation of uncertainty in size and geometry, fluid and material physical properties under reactor conditions at high pressure and temperature. Such a realistic uncertainty evaluation, however, is uneasy even for TH safety assessment because of lacking in the data under prototype conditions. An appropriate R&D planning possibly with experimental campaign and sharing it through the updated RMs is desirable to cope with the safety assessment of emerging reactors. In the Plenary session, RM and phenomena scaling will be discussed to think together how to solve our problems for meaningful reactor safety assessments.

Keywords: Roadmap, Scaling, Thermal-hydraulics, Severe accident, V&V, Uncertainty evaluation, Safety assessment, Prototype conditions

Regulatory Perspective:
Beyond Design Basis Analysis and Expectations on New Reactor Designs

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On 11 March 2011 Japan suffered its worst recorded off-shore earthquake. Reactor Units 1, 2 and 3 on Fukushima-1 site were operating at power before the event and on detection of the earthquake were shut down safely. The tsunami caused by the earthquake inundated the site resulting in the loss of essential supplies. Despite the efforts of the operators under extremely difficult conditions, back-up cooling was eventually lost, resulting in fuel melt, hydrogen generation and its release in Units 1 to 3 containments and their designed venting system. This subsequently led to several hydrogen explosions, leading to major releases of radioactivity to the environment including leakage to water sources.

This in time resulted in a significant impact on the Japanese regulatory approach and philosophy, as well as nuclear development across the globe, particularly in those countries with an active programme of civil nuclear power generation.

The accident renewed public interest in nuclear safety and an added focus on the independence of national nuclear regulatory authorities. This presentation will provide a brief overview of the changes to ONR's standards and guidance since the accident has occurred, and the UK's regulatory approach to the assessment of design basis, beyond design basis and severe accidents.

The presentation will also cover the current reactors being reviewed as part of ONR's Generic Design Assessment (GDA) and a brief overview of ONR's expectations and scope relating to severe accident analysis and severe accident management guidelines.

Key words: Severe Accident, Regulatory Expectations, Guidance and Standards

Incoming Euratom Research funded Projects on Severe Accident and Nuclear Safety

A. Iorizzo
European Commission

ABSTRACT

Dr Angelgiorgio Iorizzo (European Commission, DG RTD, Unit Euratom Research) gave a presentation titled “Incoming Euratom Research funded Projects on Severe Accident and Nuclear Safety” on the current EC Policy and actions on Severe Accident and Nuclear Safety.
Policy points discussed: the role of EC Commissioner for innovation, research, culture, education, and youth in the development of Severe Accident and Nuclear Safety research in the EU, the European Industrial Alliance on Small Modular Reactors lead by EC DG ENER with the support of DGs GROW, RTD and JRC.
The presentation detailed the EC Euratom Reserch actions on Severe Accident and Nuclear Safety: Horizon Europe & Euratom, the Selection procedure for the Euratom Research funded Projects, the Euratom research and training programme RTD, the Euratom research and training programme 2023-2025, the Incoming projects on Severe Accident and Nuclear Safety, the Euratom Research cooperation with International Organizations on Severe Accident and Nuclear Safety.

RECENT IAEA ACTIVITIES RELATED TO SEVERE ACCIDENTS

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ABSTRACT

The International Atomic Energy Agency (IAEA) fosters the efficient and safe use of nuclear power by supporting existing and new nuclear programmes around the world, catalysing innovation and building capacity in energy planning, analysis, and nuclear information and knowledge management. This presentation highlights various IAEA initiatives focused on the analysis and management of severe accidents.

- Technical Meetings to facilitate the development and validation of computer codes and experimental techniques for analysis of severe accidents.
- IAEA Coordinated Research Activities provide opportunities to engineers and scientists in developing and developed Member States to collaborate on research topics of common interest.
- Courses on severe accident phenomenology jointly conducted by the IAEA and the International Centre for Theoretical Physics, ICTP, in Trieste, Italy. The courses offer lectures and exercises on the phenomenology of severe accidents in water-cooled reactors, covering recent R&D findings and research-supported analyses, along with an overview of numerical methods and codes for event prediction.
- Training workshops on the development of SAMGs using the IAEA's SAMG-D Toolkit with the goal of supporting and assisting Member States in understanding, developing and establishing their SAMG programmes to enhance the nuclear safety knowledge, expertise and competence related to SAMGs.
- Education and training programmes on active learning about nuclear technologies through basic principle simulators. The IAEA coordinates the development and dissemination of basic principle simulators, along with accompanying manuals and documentation. Additionally, the agency sponsors education and training courses and workshops covering physics and technology of advanced reactors, as well as methodologies for technology assessment.
- IAEA publications related to severe accident analysis and management, providing valuable resources for the global nuclear community.

Status and perspectives in NEA joint nuclear safety research projects in the severe accident area

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ABSTRACT

For over 65 years, the Nuclear Energy Agency (NEA) has served as a flexible and powerful platform for multinational research co-operation in areas related to nuclear safety, including management of severe accidents. The projects conducted under the NEA auspices have enabled nuclear safety regulators, technical support organisations (TSOs), the nuclear industry and research organisations to share research costs and results and develop expertise. This, in turn, has supported safety regulations and practices and facilitated their harmonisation around the world.

The NEA held a Nuclear Safety Research Joint Projects Week event in January 2023 to review the accomplishments of NEA Joint Nuclear Safety Projects over the last four decades and to discuss future perspectives. Through a series of webinars, leaders in the sector and research project operators discussed the outcomes of close to fifty nuclear safety projects and examined how the established frameworks, research platforms and networks could effectively support future developments in nuclear energy (NEA, 2024).

A key point of discussions was related to the challenges ahead, the effectiveness and efficiency of these activities, but also how they might better serve the development and maintenance of key skills and research infrastructures in the future. In addition, participants explored new ways of encouraging public and private stakeholders to contribute to the funding of future safety research joint projects. Several additional objectives, listed below, were discussed and are being addressed at NEA, including for the severe accident area.

Regarding the joint projects' frameworks:

− Promoting a more integrated approach in safety research around sets of experimental platforms in the main safety technical areas, where appropriate, e.g., in the thermal-hydraulic area;

− Engaging with policy-making bodies, efficiently recruiting stakeholders and growing the funding base;

− Contributing more broadly to education, training and knowledge dissemination;

− Securing the quality assurance (QA), preservation and dissemination of the project results for the benefit of the nuclear safety community and organising, where needed, database framework models;

− Increasing interaction between joint research projects and relevant NEA working groups to broaden as much as possible the insights into the projects as well as the communities supporting them.

Regarding research plans:

− Establishing a nuclear safety research roadmap, prioritising work globally and by safety technical area including the severe accident area, so as to balance the needs of currently operating and future reactors;

− Developing Phenomena Identification and Ranking Tables (PIRTs) for innovative reactor designs and phenomena related to long-term operation (LTO) issues.

The talk will set out the main conclusions and recommendations of the event for projects in the severe accident area and describe the NEA initiatives dedicated to support future research in the severe accident

area (e.g. development of a nuclear safety research roadmap, organization of the FRAME workshop on Future Research for Accident Management Enhancement in operating and future reactors, informed by Fukushima Daiichi insights).

(NEA, 2024), Summary and main outcomes of the NEA Nuclear Safety Research Joint Projects Week, Nuclear Energy Agency (NEA) - Summary and Main Outcomes of the NEA Nuclear Safety Research Joint Projects Week (oecd-nea.org)

IVMR MODELLING WITH TRANSIENT EFFECTS DURING MOLTEN POOL FORMATION AND STABILIZATION – OUTCOMES FROM MODELS' COMPARISON PERFORMED IN THE IAEA CRP J46002

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ABSTRACT

In 2020 the International Atomic Energy Agency (IAEA) has started the 4-year Coordinated Research Project (CRP) on In-Vessel Melt Retention (IVMR) with the main objective to harmonize the international understanding of the scientific and technological bases underpinning crucial parts of the safety demonstration of this Severe Accident (SA) management strategy. IVMR is already implemented in several nuclear power plants of generation II and III (VVER440, AP1000, HPR1000…) and under consideration for implementation in existing VVER-1000 reactors and future Small Modular Reactor (SMR) for examples. This strategy consists in maintaining the degraded reactor core (corium) within the vessel by ensuring its cooling thanks to cavity flooding and power extraction through the vessel wall.

In the scope of this CRP, several analytical benchmarks were performed focusing on different reactor designs: a generic 1000 MWe PWR design, the Russian VVER-1000 design, the Chinese CAP1400 design and the Canadian CANDU design. This paper focuses on work done and achievements obtained within the generic 1000 MWe PWR design benchmark. This benchmark is a follow up and an extension of the one developed in the scope of the European H2020 IVMR project. Compared to the other benchmarks dedicated to a specific reactor design, this benchmark is called “generic” because its objective is not to simulate expected corium configuration for a given scenario and reactor but to allow detailed comparison of models between codes based on prescribed and simplified configurations. It intends to cover generic models to be implemented in capable codes (either integral SA code or dedicated code) which may be used for IVMR simulations. Several cases are defined with increasing complexity to allow step by step comparison of the code models. Such an approach is useful for code developers but also for code users in order to better understand results and code capabilities or limitations. Specific attention is paid to molten pool formation and transient effects until reaching steady state configuration. Eight different benchmarks have been developed covering molten pool formation from solid particles, progressive corium relocation to the lower plenum, progressive molten steel incorporation, vessel wall ablation and possible stratification inversion. State of codes performance and remaining issues are discussed in this paper, focusing on configurations involving progressive molten material arrival in the lower plenum.

KEYWORDS

Severe accident, In-Vessel Retention, code benchmark, transient phase, BEPU

NUMERICAL ANALYSIS OF MELT PENETRATION BEHAVIOR IN THE CONTROL ROD DRIVE HOUSING OF FUKUSHIMA DAIICHI NUCLEAR POWER STATION UNIT-2

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ABSTRACT

The decommissioning of Fukushima Daiichi Nuclear Power Station (1F) Unit-1 to 3 requires information regarding the statuses of the damages of the reactors and distribution of the fuel debris for the purpose of debris removal. For Unit-2, the muon radiography and the Primary Containment Vessel (PCV) internal investigation results indicate that the fuel debris are largely retained inside the Reactor Pressure Vessel (RPV), although multiple breaches are expected at the RPV lower head, some of which must be large enough to let the upper tie plate of the fuel assembly to relocate down to the pedestal floor. Hence, numerical analyses are necessary to deduce comprehensive scenario candidates, which can explain the current observations.

The previous accident scenario analysis suggested that in the 1F Unit-2 the dryout (disappearance of liquid water) of debris bed formed in the lower plenum of RPV following core slumping (relocation to lower plenum) and quenching probably took place at ca. 4:00 on March 15th, 2011. Subsequently, the debris bed in the lower plenum was estimated to gradually heat up due to decay power, and metallic melt could have first formed and caused damages to the Control Rod Drive (CRD) housings made of Stainless Steel (SS). Such damages could occur below the melting point of SS as suggested in the experiment ELSA-1 which revealed the possibility of CRD housing breach by eutectic melting. Once such CRD housing breach occurs, the metallic eutectic melt could penetrate inside the CRD housing followed by fuel pellets which can heat up from the inside of the CRD housing inducing its further potential failure. To investigate such possibility, the current study focuses on the analysis of such metallic melt penetration behavior in the CRD housing. To well capture the melt flow and its interactions with the CRD housing, including the heat transfer and melting / solidification, it is beneficial to apply the Moving Particle Semi-implicit (MPS) method, which is a Lagragian method that excels in capturing the free surface and melt/solid interface. A three-dimensional CRD housing model with simplified inner structures was established. The injection of SS-Zircaloy eutectic melt into the CRD housing was simulated and its downstream penetration and freezing behavior under vertically varying temperature boundary conditions was analyzed. It is found that the melt would start to freeze and form channel blockages soon after it enters the region with a relatively cold boundary in the downstream.

KEYWORDS
Fukushima Daiichi (1F), BWR penetration, severe accident, MPS method

REACTOR PRESSURE VESSEL INTEGRITY DURING SEVERE ACCIDENT WITH CORE MELTDOWN: CHARACTERIZATION OF MATERIAL PARAMETERS, STRUCTURE INTEGRITY ASSESSMENT AND THERMAL-HYDRAULIC ASSESSMENT

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ABSTRACT

For high power reactors, such as VVER 1000, is the assessment of reactor pressure vessel (RPV) during several accident (SA) with application of In-Vessel Melt Retention (IVMR) still an open issue. There is no international consensus or document that defines the exact procedure how to assess the RPV during SA as in "conventional" assessment such as fatigue. This paper is focused on the basic steps of structural and thermal RPV assessment and how to compute deformations on RPV wall. This paper describes material inputs, severe accident progression calculations to obtain thermal boundary conditions and finally the structural assessment of RPV for the reference plant being Czech Temelin nuclear power plant (NPP), equipped with 2 VVER-1000/320 units.

KEYWORDS
IVMR, FEM, RPV integrity, TH calculations, Creep, Ablation

Improvement of thermochemical corium stratification accounting for uranium and zirconium composition difference in metallic and oxide phases

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ABSTRACT

The in-vessel melt retention (IVR) is a mitigation strategy aiming at keeping the corium pool inside the reactor pressure vessel (RPV). The success of this mitigation strategy requires that the RPV keeps its integrity. It must withstand the heat transfers delivered by the corium pool, *i.e.* it shall not be ablated beyond its rupture point. To ascertain this last point, the heat fluxes coming from the corium pool must be calculated as precisely as possible. The main threats for RPV integrity are, the focusing effect and the rising of an overheated heavy metal layer to the upper part of the corium bath, due to the large heat fluxes these events impose on the RPV. The kinetics of the focusing effect layer height, as well as the heavy metal layer rising, are dictated by the inter-layer thermochemical transfers at play. These are often modelled as 0D integral models in severe accident codes where only inter-layer mass transfers are explicitly taken into account *i.e.* the transitory relocation of metal droplets through the oxide phase is considered as infinitely fast. While this dispersed phase relocation kinetics is fast in comparison with the inter-layer mass transfer, it can have a significant impact, as droplets keep on exchanging mass as they pass through the oxide layer. This was shown in a previous model prototyping work in which the impact of the steel relocation either on top of the pool or as droplets in the oxide was studied. However, this last work is limited by the assumption that the Uranium over Zirconium ratio ($R_{U/Zr}$) is kept uniform among the phases (which is contrary to thermodynamic equilibrium data) and constant in time (preventing the analysis of scenarios with multiple corium relocations from the core). It is important to lift the latter hypothesis as $R_{U/Zr}$ may change with the course of the accident and thermodynamic predictions show that a difference of 33 % between the $R_{U/Zr}$ in the oxide phase and in the metallic phase. It is also typically greater than 1 in the oxide and lower than 1 in the metal phase. In this paper, a model taking into account this difference in $R_{U/Zr}$ between the oxide phase and the metallic phase is derived and tested. It is shown, through numerical simulations, that this difference in $R_{U/Zr}$ can change the stratification transient duration.

KEYWORDS
In-vessel melt Retention, Corium pool Stratification, Multicomponent diffusion, Lumped parameter model, Droplets

STAINLESS STEEL OXIDATION AT BOTH SOLID AND LIQUID STATE UNDER AR-H_2O GAS MIXTURE IN SEVERE ACCIDENT CONDITIONS

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ABSTRACT

Extensive failures in nuclear reactor cooling system could lead to different severe accident scenarios. At high temperature, highly corrosive magma called corium is formed as a mixture of molten materials constituting the nuclear fuel, the cladding and the vessel. Consequently, corium is likely to interact with the oxidizing gaseous atmosphere of water steam. This phenomenon is responsible for oxidation process, heat release and hydrogen production, which may result to explosion. The available literature on corium oxidation kinetics and mechanisms appears to be lacking and questionable, as it often relies on integral experiments assuming a specific configuration of corium pool layout that could occur during a severe nuclear accident. The multiphasic aspect inherent to this kind of experiment limits access to knowledge on the kinetics of corium oxidation, as well as on the evolution of its chemical composition when the temperature increases. In order to contribute to fill the lack of information in this area, several high temperature oxidation tests have been conducted recently on VITI facility of the PLINIUS Severe Accident Platform at CEA Cadarache. The VITI separate-effect set-up allows testing at high temperatures using induction heating, up to 3000 K. It also enables to measure hydrogen production in-situ using online gas analysis during sample oxidation. For this matter, a new optical gas analyzer, calibrated to the hydrogen absorption wavelength has been used and qualified, providing reliable quantitative data on in-situ hydrogen measurement. The oxidation of the vessel's 304L stainless steel was studied, under an oxidizing atmosphere of steam by a two-gas mixture: 90% mol. Ar - 10% mol. H_2O. The focus will first be on the mechanisms and kinetics of oxidation just below the solidus point, as appropriate, in the solid state. Subsequently, attention shifts to higher temperature levels, studying oxidation in the liquid state. Finally, a new dihydrogen production law for 304L stainless steel is proposed.

KEYWORDS
stainless steel, steam oxidation, hydrogen in-situ, solid state, liquid state

A THERMODYNAMIC STUDY OF MOLTEN POOL STRATIFICATION MORPHOLOGY

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ABSTRACT

In the event of severe accident in nuclear power plant, the stratification morphology of the molten pool within the pressure vessel's lower head has a profound impact on the effectiveness of In-Vessel Retention strategies. The analysis of the melt pool stratification requires liquid density data of nuclear materials, which are rarely included in general thermodynamic databases. This study presents a novel thermodynamic analysis method by integrating a dedicated volumetric database with NUCLEA15 to calculate densities of stratified liquid phase at high temperatures, using the Thermo-Calc software. Various stratification patterns under changing conditions of U/Zr ratios, Zr oxidation, and steel content are investigated to explore their influence on element distribution and layer density. It is found that a decrease in Zr oxidation or steel content leads to an increase in metal phase density, making it more likely to be situated beneath the oxide layer. The calculation results agree with data from MASCA-1 and CESEF experiments, confirming its effectiveness and reliability.

KEYWORDS
stratification morphology; thermodynamics; liquid density; severe accident

IN-VESSEL CORIUM THERMOCHEMISTRY BENCHMARK BASED ON MASCA EXPERIMENTAL DATA

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ABSTRACT

This paper presents the first two steps of a benchmark on in-vessel corium thermochemistry, which is part of the IAEA Coordinated Research Project on In-Vessel Melt Retention. The benchmark focuses on the stratification of corium in the lower plenum of a reactor vessel during a severe accident. The experimental data from the MASCA test facilities are used to validate/calibrate the models of the different codes used by the participants. The paper gives an overview of the different models, including their underlying assumptions. The results show that the models are generally in good agreement with the experimental data, but further experimental data is needed for model validation. Besides the need for further work, this paper highlights the fact that integral models (published in the open literature for most of them) are available to account for the first-order phenomena associated with in-vessel corium. Accordingly, this paper may be useful for code developers that would like to upgrade their models in order to take into account transient stratification kinetics, a prerequisite for a state-of-the-art analysis of in-vessel melt retention.

KEYWORDS
corium, thermochemistry, stratification, benchmark

COOLABILITY OF A CORIUM POOL IN A DEBRIS BED – CALCULATION OF CRITICAL HEAT FLUX (CHF) TESTS WITH THE ASTEC CODE

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ABSTRACT

In case of severe accident in a light water reactor, a debris bed may form in the core and possibly melt, as it happened in TMI-2. Knowledge about the coolability of such molten pool surrounded by debris is crucial to investigate the possibility of stabilizing a part of the fuel inside the vessel. Based on experimental tests conducted at IRSN investigating the effect of debris size, steam and liquid flowrate, inclination and pressure, some numerical simulations have been conducted with the severe accident system code ASTEC. Some global conclusions could be drawn: the surface temperature evolution is well reproduced and the pool boiling CHF value is correctly calculated except in presence of debris. Pressure and liquid flowrate are satisfactorily evaluated but differences are shown as concerns the debris diameter impact, the inclination impact and the gas flowrate impact. The observed discrepancies are analyzed here and prospects for improvements of CHF and void fraction modelling are provided.

KEYWORDS
Severe accident, Corium coolability, Critical Heat Flux, ASTEC code

SEAKNOT: LOOKING AHEAD OF SEVERE ACCIDENT RESEARCH

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Abstract

Severe Accidents (SAs) dominate the risk associated to the commercial production of nuclear energy. A vast amount of research has been done for decades to minimize consequences of these events by optimizing their management. However, further research is still needed to continuously improve the outstanding safety record, to deal with upcoming new technologies (ATFs; SMRs), more stringent safety requirements (i.e., no need of evacuation under any condition), more challenging operating conditions (i.e., higher burnups; load-follow operation mode), and further enlargement of the research scope to long-term processes affecting accident sites control. Under the present circumstances, though, an efficient research roadmap should be outlined to achieve: the preservation of both previously obtained knowledge and know-how and infrastructures (analytical and experimental); and, more importantly, the development of a skillful workforce capable of making the most of the current research tools based on a deep understanding of fundamentals. This is the final objective of the SEAKNOT project (Severe Accident research and KNOwledge managemenT for LWRs).

This paper introduces the project (, launched in October 2022 under the frame of EURATOM researche and training program in HORIZON EUROPE, coordinated by CIEMAT and participated by 17 European organizations. The SA roadmap is to be built on a critical assessment of the current state-of-the-art through a Phenomena Identification and Ranking Table (PIRT), a directory of the code validation database (VADD) and a European mapping of experimental infrastructures. The project gives an outstanding relevance to the knowledge and knowhow transfer to young generations of nuclear scientists and engineer, generation who will be responsible for conducting the necessary SA research and analysis in the coming decades. Furthermore, activities like the Severe Accident Phenomenology course (SAP) and the European Review Meeting on Severe Accident Research (ERMSAR) are in the project portfolio for dissemination and communication activities.

Besides shortly describing the SEAKNOT project, the paper will outline the progress made one year after its beginning.

Keywords: R&D priorities; Database directory; Severe Accident; knowledge and know-how transfer.

ASTEC core degradation calculations in support of Level-2 Probabilistic Safety Assessment for 1300MWe French reactors: methodology and preliminary results

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ABSTRACT

The Institute for Radiation Protection and Nuclear Safety (IRSN) in France has updated its Level-2 Probabilistic Safety Assessment (L2 PSA) for the French 1300MWe Pressurized Water Reactors (PWRs) as part of the decennial safety reevaluation for these specific reactor units. This study was particularly underpinned by computations performed using the IRSN ASTEC V2.2 code. ASTEC, which stands for Accident Source Term Evaluation Code, is the reference integral code employed by IRSN for modeling and predicting the progression of severe accidental sequences. Within this framework, IRSN has conducted a total of 554 simulations of accidental sequences, for both conditions of 100% Nominal Power and of reactor shutdown. These accidental sequences have been defined based on the IRSN results of Level-1 PSA. They encompass the entire spectrum of events starting from initiating event to the point of vessel rupture. Furthermore, these simulations implement state-oriented Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs). This paper presents the different calculations carried out, outlines the methodology used to define them and the primary outcomes derived.

KEYWORDS
Severe Accident, L2 PSA, PWR, ASTEC, SAMG

Log Number: 012

MELCOR ANALYSES FOR INVESTIGATION ON HYDROGEN MANAGEMENT DURING BWR SEVERE ACCIDENTS WITH THE FILTERED CONTAINMENT VENTING SYSTEM

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ABSTRACT

In the unlikely event of a severe accident, the basic hydrogen gas (H_2) management of a BWR differs from that of a Pressurized Water Reactor (PWR) as it cannot be condensed by the Suppression Chamber (S/C) water pool and directly contributes to pressurization of the small PCV. Moreover, unlike the PWR, the risk of H_2 explosion does not exist inside the PCV but outside as H_2 leaks out from the N_2 filled containment to the O_2 rich atmosphere of the Reactor Building (R/B). In particular, the risk of H_2 explosion in the Reactor Well (R/W) may be of worth investigating as it is a closed small volume and is not suited for installing the Passive Autocatalytic Recombiner (PAR) to keep H_2 concentration low. Normally, such risk is mitigated by injecting water into the R/W so that gas leakage from the PCV top head flange is prevented. Nevertheless, as anything may happen during the severe accident, this study focuses on the scenario that the planned R/W water injection fails and significant H_2 leaks out of the PCV to the R/W. In particular, this study investigates the possible side effect of the filter containment venting system (FCVS), which may in principle, further enhance possibility of such H_2 leak from the PCV in post-containment venting phase.

MELCOR-2.2 was utilized to identify a possible accident scenario with the standard "intact PCV gas leakage model", which is commonly assumed for evaluation of a BWR. By numerous parametric scenario studies with different venting combinations at different venting timings, the analysis results could identify a scenario, in which the PCV venting accompanied with high pressure loss (i.e., venting through S/C and FCVS) led to continuous hydrogen leakage from PCV to R/W after the venting due to the residual pressure (slightly higher than the atmospheric pressure) in the PCV and led to ex-containment H_2 explosion.

KEYWORDS

Severe accident management (SAM), boiling water reactor (BWR), filtered containment venting system (FCVS), MELCOR

ON THE PROGRESS MADE IN SAFETY ASSESSMENT AND SEVERE ACCIDENT MANAGEMENT AS PART OF THE FRENCH FUKUSHIMA POST-ACCIDENT RESEARCH PROGRAMME

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ABSTRACT

The Fukushima-Daiichi accidents in 2011 underlined the importance of severe accident management (SAM), including external events, in nuclear power plants (NPP) and the need of deepen safety research implementing efficient mitigation strategies. For these reasons, the French General Commissariat for Investment has released significant additional resources to strengthen research in this area. Within this framework, several projects have been launched on the highest priority issues, in order to fill the knowledge gap and reduce the uncertainties still hanging over the main phenomena governing severe accidents and their management.

Among them, the PERFROI, DENOPI, ICE, MITHYGENE and MIRE projects, led by IRSN, addressed remaining issues related respectively to fuel degradation and reflooding, spent fuel loss of cooling, fuel-coolant interaction, hydrogen risk and source term mitigation. For each of the projects mentioned, experimental and analytical activities were carried out to help understand possible accident scenarios and related phenomena, and to contribute to improving the safety of existing and future reactors.

This article gives an overview of the main results of these projects. The achievements in terms of modelling severe accidents and improving severe accident management are presented and their applicability to innovative reactors, including SMRs, is discussed.

KEYWORDS
Reflooding, spent fuel, ICE, hydrogen, source term

HORIZON EURATOM ASSAS PROJECT: CAN MACHINE-LEARNING MAKE FAST AND ACCURATE SEVERE ACCIDENT SIMULATORS A REALITY?

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ABSTRACT

Interfacing reference severe accident (SA) codes with commercial nuclear simulators can help spread SA knowledge to a larger audience and build more accurate plant models. However, this requires improving the performance of calculation codes. The Horizon Euratom ASSAS project (Artificial intelligence for the Simulation of Severe AccidentS) addresses this issue and evaluates, among other strategies, the possible added-value of machine-learning (ML) to speed-up calculations.

Efficient programming will be applied to ASTEC (Accident Source Term Evaluation Code, developed by IRSN) [1], with negligible impact on accuracy. This might be sufficient to achieve a real time execution of ASTEC if combined with simplifications of some physical models and of the nodalization used in the input decks. Nevertheless, such modifications are hard to implement, and high acceleration factors are out of reach for complex legacy multiphysics codes like ASTEC.

ML could be a game-changer to reach significantly higher acceleration factors. The first step is the definition of the parts of the SA code that can be replaced by a data-driven surrogate model. Global models, replacing the code integrally, can be more efficient and do not require access to the source code. They will be tested on ASTEC for the ex-vessel phase of the accident. On the other hand, they are expected to be data-greedy since they need to capture a large diversity of physical phenomena. Therefore, ASSAS also explores hybrid approaches, for which surrogate models will replace only selected functions of ASTEC and will be interfaced with other modules.

The specifications of the basic-principles simulator to be developed during ASSAS must be selected at the beginning of the project, to ensure the consistency between the plant model and the training data for ML models. Exercise scenarios have been chosen to determine the systems to be modelled, their scope and the data to be displayed on the Human-Machine Interface.

The present paper aims at providing a description of the ASSAS project and its goals. The adopted strategy and the related challenges will be discussed.

KEYWORDS

Simulator, hybrid machine-learning, ASTEC, nuclear energy, severe accident, training, efficient programming

PRELIMINARY STRATEGIES FOR TRAINING DATASET GENERATION AND SURROGATE MODELLING OF STATION BLACKOUT MANAGEMENT MEASURES IN PWR

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ABSTRACT

This study is conducted in the frame of the EU-funded project ASSAS (Artificial intelligence for Simulation of Severe AccidentS), which aims at developing a basic-principles severe accident simulator for a generic PWR-1300 MW and at evaluating the possibility to replace some ASTEC models with machine-learning surrogate models. This paper presents the preliminary stage of the surrogate model development, focusing mostly on the generation of training datasets necessary for the machine learning process, and addresses essentially the CESAR module, which handles thermal-hydraulics in the primary and secondary circuits.
To achieve this, a base case scenario involving a Station Blackout (SBO) without any safety measures until Reactor Pressure Vessel (RPV) failure is considered. From this base case, various calculations are performed by introducing safety measures (e.g. low-pressure injections) at different times. This safety measures sampling is applied within the time range between the onset of the primary pressure drop (enabling low-pressure injections) and the RPV failure initially predicted in the base case.

Two approaches are employed. The first involves conducting a batch of 22 runs, varying one or two criteria in each run, and saving most relevant ASTEC variables at each time step (especially state variables from CESAR cells, but also system signals such as pump actuation). This should generate a suitable dataset for the development of time-stepping surrogate models, that are expected to compute each time-step, like ASTEC does, and also account for user's decisions during the accident calculation interactively.

Alternatively, another approach would consist of developing a meta-model that predicts key outcomes of the SBO - such as the delayed RPV failure time or final hydrogen production - with respect to one criteria (to simplify the problem): the safety injection time. However, due to the intricate mechanisms leading to RPV failure, training the neural network requires a dataset with a larger amount of samples, i.e. a finer variation of the injection time; here, a batch of 110 runs. Unlike the previous approach, preserving results at every time step is not as necessary. This last strategy is quite independent of the ASSAS project, as this global approach cannot suit the requirements of an online simulator; the purpose here is more exploratory.
The last part of the paper proposes machine-learning strategies and discusses the optimal utilization of the generated training datasets.

KEYWORDS
ASTEC, SBO, machine-learning, training

Analyses of accident progression behavior and simulation of FP release into the environment using MELCOR code for the Fukushima Daiichi Nuclear Power Station Units 1 to 3

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ABSTRACT

Plant data from the accident at Fukushima Daiichi Nuclear Power Station Units 1 to 3 are not enough to understand the overall picture of the accident. It was identified which unit's each event was responsible for the peak of the dose rate measurements at the on-site monitoring post. Although it was not possible to confirm the time of core meltdown from the measured pressures of the reactor pressure vessel (RPV) and the primary containment vessel (PCV), it was possible to estimate the time of the RPV failure. For the purpose of assisting the detailed explication of the accident, we conducted MELCOR2.2 code analysis applying plant data from Units 1 to 3 and safety function operating conditions during the accident to reproduce the time of the RPV failure and the hydrogen explosion, and also to predict the stabilization of the accident. It was predicted that the accident had not stabilized at any of the units by 24:00 on 3/31 2011 as the end of calculations, which was consistent with the measured plant data.
The amount of leakage from the PCV that can be read from plant data at the accident was estimated on condition that the decay heat might be reduced by the amount of water injected to cool the reactor core, its amount of evaporation and the amount of leakage. Also, the amount of leakage was estimated so that the measured pressure of the PCV could be simulated. Inside the RPV, the fuel debris collapsed into fragments with the temperature not rising to its melting point, and the control rod drive mechanism remained in its original shape to some extent due to the injection of cooling water. The other side, MELCOR2.2/CORCON handles the debris like molten material and hastens the molten core-concrete interaction after it falls onto the pedestal, so that MELCOR analysis results in a conservative assessment. The temperature of the debris was below the melting point of iron as around 1500 K after falling onto the pedestal and dissipating heat, so that it was considered the debris eroded not the steel wall but the concrete floor. Therefore, we conducted the evaluation that only caused erosion to the floor. The most plausible analyses cases for the Units 1 to 3 accident were prepared based on the above idea. The analyses results of FP release timings into the environment agreed well with the measured plant data.

KEYWORDS
Fukushima Daiichi Nuclear Power Station accident, MELCOR, severe accident, source term, on-site dose rate, environmental FP release

LONG-TERM SEVERE ACCIDENT MANAGEMENT AT LOVIISA NPP

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ABSTRACT

Loviisa nuclear power plant is located in Finland and has two VVER-440 units. The implementation of severe accident management systems was finalized in 2003, but the accident management strategy did not consider fully long-term issues. For long-term severe accident management it is essential that core melt can be solidified and pressure and temperature inside containment can be brought down. In Finland new regulations to achieve these objectives have been presented in 2019.

Equipment survivability in high radiation conditions is a fundamental issue in long term severe accident management. For Loviisa nuclear power plant a long term severe accident management strategy has been developed and justified by a set of analyses. Source term and containment pressure were key parameters to evaluate the acceptability of the strategy as the goal of the new regulations is that the radioactive releases are low even if containment building loses its leak tightness at some point.

The strategy is to use low pressure emergency core cooling system, that will be taken in service after the early phase of an accident. At this stage the core has already melted and radioactive material has spread into the containment thus the pumped water will be highly radioactive. Equipment survivability will be assessed based on the need to operate the components, dose estimates and materials used in the components. This work is still ongoing and possible actions are considered based on the results.

KEYWORDS
MELCOR, Apros, MCNP, SAMG, dose

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ABSTRACT

Maturity of severe accident (SA) codes, progress in the computational methods, and computer infrastructures were considered a sound platform to conduct, for the first time in SA modelling, a systematic and broad application of Uncertainty and Sensitivity Analysis (UaSA) in this domain. The overall objective of the HORIZON-2020 project on "Management and Uncertainties of Severe Accidents (MUSA)" was to quantify the uncertainties of SA integral codes when modelling reactor and spent fuel pool (SFP) accident scenarios of Gen II and Gen III reactor designs for the prediction of the radiological source term. This paper presents the major achievements accomplished by MUSA. To name a few: a database on input parameter uncertainties has been assembled; SA codes and statistical tools, either in-house built or existing ones, have been properly coupled by scripting or interfaces; major specific challenges have been identified and different solutions have been worked out; and, applications to several reactor types and SFP scenarios have shown interesting insights of this simulation approach, particularly when dealing with Source Term variables as figures of merit. No less important, major challenges were found and are here discussed as needs to be addressed before achieving an effective implementation of UaSA in the SA domain.

KEYWORDS
Severe accident analysis; Uncertainty and Sensitivity Analysis, UaSA; Source Term

SOURCE TERM UNCERTAINTY ANALYSIS OF FILTERED CONTAINMENT VENTING SCENARIOS IN NORDIC BWR

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ABSTRACT

Nordic Boiling Water Reactors employ filtered containment venting and ex-vessel debris coolability in the deep pool located under the reactor pressure vessel as a severe accident management strategy.

This paper focuses on the uncertainty analysis of the source term in accident sequences that result in filtered containment venting to the environment using the MELCOR code. The impact of uncertain MELCOR modeling parameters and modeling options on the timing and magnitude of the source term released to the environment has been evaluated in accident sequences initiated by a large break LOCA and SBO.

The performed simulations illustrate the effect of MELCOR modeling parameters and options on the code's predictions of severe accident progression, event timing, and the magnitude of the source term released to the environment in different accident scenarios. Furthermore, the results highlight the importance of various retention mechanisms that limit the release of fission products into the environment.

KEYWORDS
MELCOR Uncertainty MVSS Severe Accident

Uncertainty and Sensitivity Analysis of the ASTEC Source Term Results of a MBLOCA Scenario with the Activation of Severe Accident Management Actions in a Generic KONVOI Plant

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Abstract

The results of uncertainty and sensitivity analysis (U&Sa) presented in this work were performed at Karlsruhe Institute of Technology (KIT) in cooperation with Framatome GmbH, in the frame of their joint contribution to the HORIZON 2020 Management and Uncertainties of Severe Accidents (MUSA) project (2019-2023). The overall objective of MUSA was to assess the capability of severe accidents (SA) codes when modelling reactor and spent fuel pool accident scenarios of Gen II and III. In particular, the project aimed at consolidating a harmonized approach for U&Sa of the source term (ST) associated with SA analyses. In addition to fulfilling the main goals of MUSA, KIT's and Framatome's contribution also aimed at improving the existing approaches of ST calculations in case of SA as well as quantifying the corresponding uncertainty to support the emergency team with making decisions. In this framework, the uncertainty quantification of the ST in case of activation of Severe Accident Management (SAM) actions during a hypothetical SA scenario is addressed.

Having this in mind, a calculation platform has been developed at KIT and employed in MUSA to perform the U&Sa of the results of SA simulations by means of integral codes. The European reference Accident Source Term Evaluation Code (ASTEC), developed by Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and co-developed by KIT since 2019, is employed for performing the SA analyses. Furthermore, the KArlsruhe Tool for Uncertainty and Sensitivity Analysis (KATUSA), developed by KIT, is used to assess a database of SA results, by propagating the uncertainty of selected input parameters, and to perform the U&Sa.

In the current paper, attention is focused on the application of the KATUSA tool to the ASTEC results of Medium Break Loss of Coolant accident (MBLOCA) simulations with the activation of the Containment Filtering Venting System (CFVS) in a generic KONVOI NPP. A database of about 300 ASTEC simulations has been assessed to evaluate the effect of the uncertainty of input parameters on the ST governing the ASTEC modelling related to the fission product release, the core degradation, the aerosol behaviour, the containment leakage, the fuel inventory, and the filters' efficiency.

Furthermore, a detailed analysis of the influence of the different uncertain parameters at different stages of the accident progression and in particular after the activation of the CFVS is discussed. The results show that the KATUSA tool is able to perform the U&Sa of the ASTEC results. Moreover, the selected input parameters have been ranked based on their correlation with the ST and with the CFVS efficiency by employing Spearman and Pearson correlations. Finally, the uncertainty of the amount of key ST elements and isotopes transported to the containment and released to the atmosphere is quantified. The present work, therefore, provides a solid basis for understanding and identifying the uncertainties of the ASTEC modelling of SA sequences and for quantifying their effects on the ST to support the improvement of our current capabilities to predict the radiological impact during hypothetical SAs.

Keywords: ASTEC, SAM, KATUSA, MBLOCA, Uncertainty and Sensitivity Analyses

SOURCE TERM UNCERTAINTIES IN UNMITIGATED SBO SEQUENCES IN A PWR-1000: INSIGHTS FROM THE EU-MUSA PROJECT

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ABSTRACT

The Management and Uncertainties of Severe Accidents (MUSA) project focused on the testing, adapting, and employing Uncertainty Quantification (UQ) and sensitivity methods for Severe Accident (SA) analysis. In collaboration with the University of Pisa, CIEMAT conducted an Uncertainty and Sensitivity Analysis (UaSA) of Source Term (ST) estimates in an unmitigated Station Black-Out (SBO) scenario involving a 3-loop Pressurized Water Reactor with Westinghouse design (PWR-W). The study employed the MELCOR 2.2 code (version 2.2.21402) in conjunction with DAKOTA 6.7 as the UQ tool. To propagate uncertainties, the Wilks' approach was employed, with random sampling of 24 selected uncertain parameters. The Figures Of Merit (FOMs) chosen for the analysis were iodine, cesium, and noble gases releases to the environment. A sensitivity analysis, based on Pearson's and Spearman's Correlation Coefficients (CCs), was also conducted. In relative terms, the magnitude of the ST displayed a narrow uncertainty band. The early containment failure led to most iodine released from fuel to reach the environment, with a narrow variability. Conversely, a visible uncertainty surrounded the onset time of release to the environment (approximately 5 hours), potentially influencing the implementation of Accident Management (AM) measures that might be adopted in the scenario. Although sensitivity analyses did not identify dominant variables, emphasis was placed on the relationship between the release onset time and fuel failure time (temperature controlled), as well as the subsequent diffusional release of radionuclides.

KEYWORDS
Source Term, Severe Accident, UaSA, MELCOR.

SYNTHESIS OF SOURCE TERM ASSESSMENTs FOR A LOSS-OF-COOLING ACCIDENT IN A SPENT FUEL POOL: INSIGHTS FROM UNCERTAINTY AND SENSITIVITY ANALYSES AND POTENTIAL BENEFIT OF WATER INJECTION BY SPRAY SYSTEM

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ABSTRACT

The recently completed Horizon-2020 project on "Management and Uncertainties of Severe Accidents (MUSA)" has applied uncertainty and sensitivity analysis in the assessment of Severe Accidents (SA), particularly in predicting the radiological source term of reactor and Spent Fuel Pool (SFP) accident scenarios. For SFP, the two main objectives were: 1/ to determine the main uncertainties affecting the accident progression and fission product release during an accident in a SFP and 2/ to assess the possible benefits, in terms of source term reduction, of water injection by spray systems. The target of this paper is to describe the main achievements at the end of the 4 years' project, regarding SFP accident scenario. The uncertainty and sensitivity analyses were performed by eleven participants thanks to the propagation of the uncertainties of the input parameters to the output uncertainties of severe accident codes. A SFP accidental scenario was selected, uncertainty sources in models and input parameters were identified and key target variables were chosen as Figure of Merits (FOMs). It was found that spreading in SA code responses coming from the uncertainty quantification done by each participant was relatively small when compared to the variability between reference computations of all participants. It reveals that important uncertainty sources, such as the lack of modeling or the nodalization, have not been taken into account in the uncertainty quantification. The sensitivity analysis, based on statistical correlation coefficients and supported by the analysis of scatter plots, has put in light that uncertain parameters linked to aerosol phenomena in the building significantly affect the source term. It has also revealed that the relationship between the source term and uncertain parameters linked to degradation is not straightforward, due to the non-linear nature of the phenomena in play and due to cliff-edge effects. Concerning the benefits of spray systems, the mass flow rate and the injection time that can prevent cladding failure, and thus Fission Product (FP) release were determined by seven participants, thanks to ASTEC, RELAP/SCDAPSIM and MELCOR calculations. The results obtained are very preliminary and should not be considered as demonstrative, since SA codes are generally not validated to compute the cooling of an assembly by spray injection. It is thus mandatory to carry out a validation work of the thermal hydraulics and the POLCA OECD project would be an opportunity for this.

KEYWORDS
SFP, source term, uncertainty quantification, sensitivity analysis, spray system, SA codes

UNCERTAINTIES ON FISSION PRODUCT RELEASE IN A LOSS-OF-COOLING SCENARIO IN A SPENT FUEL POOL WITH MELCOR 2.2

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ABSTRACT

After the events occurred in Fukushima in 2011, the vulnerability of the Spent Nuclear Fuel (SNF) stored in Spent Fuel Pools (SFPs) was highlighted. Subsequent studies involving accident scenario simulations in SFPs questioned the reliability of the Severe Accident (SA) codes predictions for this purpose. The present study synthesizes the main insights gained by quantifying the uncertainties and assessing outcomes' sensitivities of a SFP analysis. Inspired by the Unit 4 SFP of Fukushima-Daiichi, a Loss-Of-Cooling Accident (LOCA) was assumed with the focus on Fission Products (FPs) release. Calculations were performed with the MELCOR 2.2 code, and the uncertainty propagation was articulated through the DAKOTA statistical tool. Fifteen input uncertain parameters related to fuel rod failures, their degradation and FPs release from fuel rods were selected. By using Correlation Coefficients (CCs), some correlations between input parameters and Figures Of Merit (FOMs) were noted. As expected, Xe and Cs uncertainties were qualitatively similar, while Ru release showed its own footprints. Two outcomes of the study worth highlighting here: the relevance of a consistent choice of the input uncertainty parameters set, both number and variables, was shown to be crucial in the analysis; a thorough characterization of the FOM sensitivity will likely require more than just an individual technique.

KEYWORDS
Source Term, Severe Accident, UaSA, Spent Fuel Pool

UNCERTAINTY AND SENSITIVITY ANALYSES OF SEVERE ACCIDENT CODES USING THE ACE ALGORITHM-BASED SURROGATE MODEL

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ABSTRACT

This paper explores the alternating conditional expectation (ACE) algorithm-based surrogate model to advance the state-of-practice in uncertainty and sensitivity methodologies for severe accident analysis. For engineering purposes, the ACE algorithm has been used as an alternative means to find the optimal functional forms of the multiple input variables and response variables of interest. Analysis results here demonstrate that compared with the reference cases the proposed surrogate model provides much higher performance in terms of the coefficient of determination (R^2) and normalized root mean square error (NRMSE), thus giving more robust insights into the relationship and correlation between the input parameters and figures of merit (FOMs) of interest. Relevant results and insights are summarized in terms of points of interest.

KEYWORDS
Severe Accident; Uncertainty and Sensitivity; Surrogate Model; ACE Method

SYSTEM IDENTIFICATION AND RANKING TABLE (SIRT) FOR CHEMICAL THERMODYNAMICS OF SEVERE ACCIDENTS

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ABSTRACT

Within the second phase of the OECD/NEA project on Thermodynamic Characterisation of Fuel Debris and Fission Products Based on Scenario Analysis of Severe Accident Progression (TCOFF-2), a Systems Identification and Ranking Table (SIRT), has been derived from the usual PIRT and adapted to chemical thermodynamics and material science. Firstly, a draft list of systems of interests has been distributed to TCOFF-2 partners. After review, a final list of 154 systems has been considered. Then system ranking has been carried out. Two series of figures of merit have been considered: importance for safety including the contribution to severe accident phenomena and to fission product behaviour and source term as well as needs for further R&D, based on the lack of existing data and the needs to improve relevant existing thermodynamic databases (mostly NUCLEA and TAF-ID). Each participating organization provided ranking of the systems for these Figure of Merits. Relevance to safety has been organized in 10 columns: Classical LWR / High BU, High enrichment fuel/ Fukushima Daiichi, TMI2, Chornobyl/Near-term ATF cladding (Cr-coated Zry)/ Near-term ATF cladding (FeCrAl) /Long-term ATF cladding (SiC) /Advanced fuels (UN, U-Si) /Advanced Modular Reactors / Post-Accident Leaching). Received rankings have then been agglomerated and averaged. A dedicated finalization meeting has been held in June 2023 in which the systems having the highest relevance and the largest needs for improvement have been considered. It must be noted that some systems for which there is a rather good knowledge, like (U,O, Zr) have not been selected, nor systems in which thermodynamic results are very hard to attain due to kinetics effect such as (U,H,O). After some system grouping, consensus have been reached on a list of 20 systems (13 linked to current or near-term fuels and 7 linked to longer-term fuels) having the highest priority. They are Fe, Pu, O; PuO_2, UO_2, ZrO_2; Fe, Pu, U, Zr; CaO, SiO_2, UO_2; FeO_x, Cr_2O_3, SiO_2, UO_2, ZrO_2; Cr, Fe, Zr, O; Al_2O_3, Cr_2O_3, FeO, UO_2; Al, Cr, Fe, ; B, O, Zr; Cr, Cs, O; Cs, Si, O; B, Cs, O; Cs, O, Sr; SiC, UO_2; H_2O, SiC; PuO_2, SiC; Si, Pu, U; Cr, Si, O; N,O,U; (Eu,Sm),Zr,O.

KEYWORDS
Corium, Thermodynamic, Phase Diagrams, Ranking

Results of the SSM-SICOPS melt tests of the EU-SAFEST project

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ABSTRACT

In the frame of the EU-SAFEST project, the Swedish Radiation Safety Authority (SSM) requested AREVA/Framatome to perform tests on molten corium-concrete interaction (MCCI) with basaltic concrete and BWR-specific corium, characterized by a higher Zr/U ratio than PWR corium.
These tests (denoted SSM-1a/2b/2c/3) were carried out in Framatome's SICOPS facility in Erlangen. Once the melt, which initially contained about 75 wt% UO2 and 17 wt% ZrO2, was generated and its interaction with the concrete had started, metallic Zr was subsequently added to the melt from the top.
All performed tests showed a fast concrete ablation, much faster than in previous experiments in the same facility and with the same concrete composition, but without Zr addition. A likely explanation for the higher erosion velocities is the now higher temperature of the melt caused by the Zr oxidation.
In addition, the absence of crusts and the even erosion profile at the bottom seen during post-test examination support the assumption of a homogeneously mixed melt pool during MCCI, consistent with the isotropic heat flux distribution observed in earlier BALI experiments with simulant fluids.
In the presence of high melt temperatures and the vivid mixing by rising concrete decomposition gases, local crusts between melt and concrete should not be stable and no evidence for them has been found in the SSM-SICOPS tests. As the formation of such crusts is seen as the main potential mechanism for anisotropic concrete ablation in the early stages of an MCCI, the investigated conditions with Zr being present in the melt can be considered capable of suppressing this effect. This is important since the core melt, after its release from the RPV, typically contains a high fraction of non-oxidized Zr.
While no crusts were found at the interface, the sampling in the late phase of the SSM-3 test revealed evidence for the formation of a kind of transition zone between melt and concrete. This conclusion was drawn based on tactile feedback during sampling from the bottom of the melt pool where a soft/viscous zone could be felt and from the appearance of the taken samples including small solid aggregates from the transition zone.
Mass spectrometric measurements during the SSM-3 test showed a significant hydrogen production, due to the reaction of water released from the concrete with the metallic zirconium. The peaks in the hydrogen production rate correlate well with the times when the measured concrete erosion velocities were highest, i.e. even at high erosion rate, the Zr in the bulk is still capable to reduce the percolating steam.

KEYWORDS
MCCI, concrete, zirconium, hydrogen production

Ex-Vessel stabilization of corium: An analysis of corium-concrete interaction with top flooding for siliceous concrete.

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ABSTRACT

In case of severe accident without possibility of in-vessel retention, the corium must be stabilized outside the vessel, either in a designed core-catcher or directly on the concrete basemat of the reactor building, after spreading. In this last case, the stabilization strategy must be efficient enough to avoid the progression of corium through the concrete basemat and to the environment.

In order to ensure corium stabilization, the physical processes leading to quench some part of the liquid corium must be fast enough, compared to the erosion of concrete. One of the simplest strategies consists in spreading the corium over a rather large area (i.e. the reactor pit plus one or two adjacent rooms) and flood it with water on top. Two processes have been identified experimentally for a potential quenching of corium: melt eruption and water ingression. Melt eruption is a process that is driven by the flow of gases (CO_2 and H_2O) generated by concrete erosion: the gas flow is likely to entrain droplets of corium into the overlying water, quenching the droplets quickly. Water ingression is a process where cracks are generated by the contraction of the top crust due to cooling: this process enables the flow of water into the cracks and the propagation of the cracking front deep into the corium crust. When the basemat is made of siliceous concrete, stabilization depends mostly on water ingression because there is little gas content in such concrete and melt eruption may only have a limited effect.

This paper presents the analysis of a situation where corium is spread over a rather large area (around 80 m^2) and flooded with water on top. First, a sensitivity analysis is made to determine the parameters which have the largest impact on the time of stabilization. The maximum heat flux extracted by water ingression (CHF) appears as one of the most important parameters. Then, a theoretical determination of the CHF is made, as a function of the concrete content in the corium, based on the existing Lister-Epstein model and on data obtained at Argonne National Laboratory. In addition, a simple modelling of transient heat conduction through the concrete basemat is proposed, to determine a criterion of stop of erosion that can be used in a lumped-parameter code. As a conclusion, it is shown that the quenching of corium may be divided in three main stages. The first stage is characterized by a fast erosion of concrete and a significant ejection and quenching of debris. This stage lasts from half a day to a few days, depending on the type of concrete and on the model parameters. The second stage is characterized by a slower erosion of concrete and the start of water ingression. This stage may last from half a day up to a few weeks depending on the type of concrete and on the model parameters. The last stage is characterized by a very slow erosion up to the complete quenching of corium. There is a non-trivial dependence of water ingression efficiency on the potential occurrence of melt eruption at the very beginning of corium concrete interaction. Melt eruption, even if limited, is likely to significantly enhance the triggering of water ingression and favor a fast stabilization of corium.

KEYWORDS

Corium, concrete, water ingression, ex-vessel stabilization

FLUIDS MIXING MODELLING WITH PHASE CHANGE FOR MOLTEN CORIUM-CONCRETE INTERACTION

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ABSTRACT

A severe accident in a nuclear power plant involving a reactor core meltdown could result in origination of molten corium, which is a mixture of nuclear fuel, claddings and structural components. In this paper, a new numerical mixing model is developed and implemented in OpenFOAM to comprehensively analyze the ablation of concrete during the Molten Corium-Concrete Interaction (MCCI). The model is capable of predicting the heat transfer, concrete ablation, phase change, mass transfer, mixing of concrete and corium, and natural convection. The model was tested and benchmarked against the well documented CCI-2 experiment. The imposed constant temperature boundary conditions as per the experimental setup of CCI-2, instigates ablation of concrete which progresses along both the axial and radial directions of the test configuration. Experimental results reveal that the axial and radial ablation of concrete is not same. Our model captured this effect because our model is considering the density of corium and concrete and its variation. Moreover, our results reveal that the melted concrete and corium, even though, having different densities, form a single phase/mixer which under natural convection further enhances the ablation.

KEYWORDS
Corium ablation; fluids mixing; heat transfer; melting; phase change

CHARACTERISATION OF PROTOTYPIC EX-VESSEL FUEL DEBRIS INGOTS SIMULATING MCCI AT FUKUSHIMA DAIICHI

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ABSTRACT

Two experiments have been carried out in VULCANO facility to fabricate fuel debris representing MCCI products expected to be found in Fukushima Daiichi pedestals. VF-U1 test was an inductively-heated corium-concrete interaction experiment in which 46 kg of corium melt ablated a basaltic concrete representative of 1F concrete. In VF-U3 test, 40 kg of powders representative of the MCCI composition - estimated from unit 1 calculations and including prototypes of major fission product elements - have been mixed then melted by induction. Samples from both fuel debris prototypic experiments have been taken for analyses. Scanning Electron Micrography of samples from these fuel debris prototypes has been carried out in conjunction with Energy Dispersive Spectroscopy and X-Ray Diffraction. Quantitative phase analysis was carried out with Rietveld method. This provides insights on the composition of these debris and the large number of phases (silicate glass, uranium-zirconium oxides of various compositions, uranium-containing zirconium silicate, aluminosilicates, ferrosilicates, chromites, metallic droplets and blocks) present in them. Comparisons between these two experiments are discussed.

KEYWORDS
Fukushima Daiichi, Fuel Debris, MCCI, SEM, XRD

Production of prototypic corium in the VULCANO facility using uranothermite and induction heating.

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ABSTRACT

VULCANO is the advanced facility of the PLINIUS severe accident platform of the CEA, enabling the melting of up to 100 kg of UO2-containing prototypic corium. The facility is equipped with several standard cameras, a thermal camera, pyrometers, weighing scales and the possibility to have up to 150 type-K thermocouples and 20 type-C thermocouples. In VULCANO, two melting technics can be used separately or combined: uranothermite and induction heating.

Since the end of 2014, the team installed a new induction generator on the PLINIUS platform. The nominal power is 400 kW and the frequency can be set between 80 and 300 kHz. The device was already used to directly melt the corium in several tests.

Since 2016, the team installed a thermite workshop inside the PLINIUS platform. It allows the fabrication of up to 100 kg of uranothermite, in 2 kg batches, under a controlled atmosphere. A large variety of composition has been fabricated and used inside VULCANO and our others facilities.

Both techniques have advantages and drawbacks. However, combining the two technics one may be able to pull the advantages of both. It means that the quick increase of the temperature of the corium with the uranothermite reaction is possible followed, heating with the induction heating (to simulate the decay heat or continue to increase the temperature for example)

After production of the corium, several actions can be done in VULCANO facility. For example: let it cool to fabricate a fuel debris block, maintain the induction heating to simulate a molten core concrete interaction or spread the molten corium on a surface.

In this paper, the measurement results of temperature, dismantling observation of the solidified ingot for experiments with uranothermitic heating and uranothermitic and induction heating that have been done in the VULCANO facility will be highlighted.

KEYWORDS
Fuel Debris, Corium, uranothermite, induction

Overview of Ex-Vessel Severe Accident Scenarios Simulations at UJV Rez

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ABSTRACT

Severe accidents (SA) at nuclear power reactors may be mitigated in an in-vessel phase of their progression (e.g., by means of in-vessel melt retention by external reactor vessel cooling, IVMR-ERVC), or after vessel failure by ex-vessel corium localization and cooling. A SA in the ex-vessel phase demands huge attention, as such a scenario may lead to breaching the final barrier for fission products (FP) hold-up – the containment (CTMT). Some NPPs rely on the so-called wet cavity strategy, which consists of flooding the reactor cavity prior reactor pressure vessel breach; the ejected corium is thus fragmented, which promotes its cool ability. On the other hand, the risk of steam explosion, which could jeopardize CTMT integrity, exists. The wet cavity strategy is also applicable only for reactors with very deep cavities. Therefore, some NPPs opt for a dry cavity configuration, which allows for corium spreading and its consequent cooling by water top- flooding. Such strategy is foreseen also for the Czech Temelin NPP, equipped with 2 VVER-1000 reactors. As breaching of the CTMT basemat would lead to large radioactivity release into the environment, which shall be practically eliminated, investigations of ex-vessel corium spreading and cooling phenomena are in the spotlight of analytical and experimental research of UJV Rez. Numerical simulations using a variety of mechanistic SA analysis codes - e.g., MELTSPREAD, CORQUENCH) were employed to assess corium spread-ability and cool-ability in the reactor cavity and adjacent rooms. Moreover, CFD-grade codes were used to perform simulations to further investigate corium spreading in more detail. The initial conditions for corium spreading, namely acquisitions of corium mass, composition, temperature, and decay power, were assessed using various integral SA codes (ASTEC, MELCOR, MAAP). Besides analytical and numerical investigations, several experimental programs, partially funded by the government grant, were launched to further support analytical and numerical work. These projects are aimed at studying corium properties at elevated temperatures, interaction of corium with RPV internals or layered steel structure at the reactor cavity floor or investigating of the possibility of using innovative materials (such as geopolymers) as a suitable reactor cavity lining, replacing concrete basemat and promoting thus ex-vessel SA mitigation. These national experimental projects are in conjunction with the broader international OECD project ROSAU, which aims to experimentally investigate large-scale corium spreading and cooling. A comprehensive overview of UJV Rez's most recent findings on ex-vessel corium behavior is presented in the proposed article, considering all the progress that has been made both in analytical activities and experimental programs, in which UJV Rez actively participates. The final goal of those investigations is to support long-term safe operation of Czech nuclear fleet.

KEYWORDS
Ex-Vessel Retention, Corium, Severe Accident, UJV, Numerical Simulations

SIMULATIONS OF FLOAT DEBRIS QUENCHING EXPERIMENTS

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ABSTRACT

Recently, a series of tests have been performed at the FLOAT experimental facility (IKE, Germany). The purpose of the facility is to investigate the debris bed coolability during top-flooding or bottom-flooding conditions, at different initial debris bed temperatures, and at the presence of air inflow from the bottom. The aim of the air injection at the bottom of the debris bed is to simulate the possibility of the release of non-condensable gases during the molten corium concrete interaction. In the paper, two top-flooding FLOAT tests are presented, analyzed, discussed, and simulated with the MC3D computer code (IRSN, France). Selected tests were performed at similar temperatures of particles in the debris bed, similar water flow injection rates, and with or without air inflow. A 2D computational domain is developed and it is considered reasonable because of the rotational symmetricity of the FLOAT facility and because at the selected tests the preferential penetrating channel of water was in the central region of the debris bed. The simulation and test results are in good qualitative agreement. The observed discrepancy could be attributed to the establishment of other penetrating channel(s) at the edge of the debris bed that could not be covered with 2D simulations.

KEYWORDS
severe accidents, debris bed coolability, computer simulation

SASPAM-SA: Assessment of the relevance and applicability of existing experimental databases to iPWR

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ABSTRACT

SASPAM-SA project, funded by Euratom, aims to investigate the applicability and transfer of the operating large light-water reactor knowledge and know-how to the iPWR, taking into account European licensing analysis needs for the severe accidents and Emergency Planning Zone. Project outcomes will help accelerate the licensing and siting process of iPWRs in Europe. One of the objectives of SASPAM-SA, in particular Work Package 3 (Applicability and Transfer of the Existing SA experimental database for iPWR Assessment), is to evaluate the relevance and applicability of the existing experimental database to iPWR. Based on postulated plant severe accident scenarios, identified and investigated in the Work Package 2 of the SASPAM-SA project, the main boundary conditions and the specific features of iPWR are determined and compared to those of large LWR. Based on this comparison, the applicability of the existing experimental data to iPWR is assessed for in-vessel, containment, source term and ex-vessel coolability (if applicable) phenomena. The specific data to be evaluated include, e.g., natural circulation, debris bed formation, liquid melt spreading, steam explosion, re-flooding of an overheated core, in-vessel melt pool formation, corium cooling under water, hydrogen distribution, combustion and mitigation, aerosol characteristics, transport and hygroscopic growth, iodine speciation and mitigation, as well as pool scrubbing. If needed, different ways to extend the applicability of existing data to iPWRs will be explored. Finally, the potential need for new experiments is determined. In this paper, a summary of the work done during the first project year is given.

KEYWORDS
iPWR, experimental data, SASPAM-SA

NUMERICAL INVESTIGATION OF NATURAL CIRCULATION INSIDE A SCALED-DOWN PRISMATIC MODULAR REACTOR

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ABSTRACT

This study investigates codes that simulate the dynamic and thermal-hydraulic behavior that are present in the natural circulation processes inside prismatic modular reactors. The RHYS coupling ASYST VER 4 package, a crucial part of the RELAP 5 framework, is deployed to enable this emulation. The setup of the research system is a plenum-to-plenum facility with two vertical pipes that are intended to imitate the radial temperature gradients found in prismatic modular reactors (PMR) cores. By implementing a range of uniform iso-flux (200 to 1400W/m^2) and distinct outer surface temperatures (278 to 308K), axial temperature distributions were analyzed to provide a comprehensive assessment of the heat transfer in the heated and cooled channels. As a result of spontaneous convection, temperature and velocity reversals were seen following maximum spots. The flux of 1400W/m^2 records a temperature increase from the entrance by 63.6 and 51% for the air and the wall, respectively, while the flux of 200W/m^2 gives a smaller increase of 14.6 and 14%. Also, the air and inner wall temperatures were impacted by 85 and 81%, respectively, as the temperature rose 30 degrees from 278 to 308K. There was a small effect of elevating the cooling temperature on the air velocity, although the high temperature of 308K was the highest one. The Nusselt number is calculated using heat transfer coefficients, which are a representation of convective heat transmission. Furthermore, data from additional literature sources shows significant agreement with the RHYS model.

KEYWORDS
RELAP 5; RHY; ASYST; Natural Circulation; Prismatic Modular Reactor

Development of a LW-SMR dry containment model with containmentFOAM

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ABSTRACT

In line with the growing interest in Europe for the deployment of Small Modular Reactors (SMRs), the Horizon Euratom SASPAM-SA Project launched in 2022 aims to investigate the applicability of the know-how for large-LWRs to water-cooled SMR (LW-SMR). More specifically, the project investigates postulated Severe Accident (SA) sequences for two different “generic” SMRs, Design-I with a submerged containment and Design-II with a dry containment and several passive safety features, based on information available in the open literature. On this basis, the ability of different codes widely used in Europe to analyse the identified SA sequences are evaluated.

This paper presents the up-to-date status of the ‘Design-II’ LW-SMR dry containment model development using containmentFOAM, a CFD package tailored for containment safety analyses based on OpenFOAM-9, which is developed at Forschungszentrum Jülich (FZJ). The use of the code in SASPAM-SA aims to complement the calculations performed with lumped parameter codes, such as MELCOR, ASTEC, or AC2. Due to the close coupling of the containment and the reactor cooling phenomena, the additional insights of containmentFOAM for the buoyancy-driven transport processes and the condensation in presence of non-condensable gases are relevant for the system response. However, the new safety concepts that are feasible due to the reduced size of SMR containments come with novel challenges for the code.

The first priority of the modelling process is to ensure a consistent definition of the model geometry, as it is based on a database conceived for lumped parameter models. The evaluation of the thermal-hydraulic conditions predicted by the lumped parameter codes motivates a review of the applicability of the code to the expected higher pressures and steam concentrations (fluid properties, condensation regimes, etc.). Besides, the liquid released from the reactor vessel occupies a significantly larger fraction of the smaller containment volume. Therefore, two-phase models are currently being tested to consider the more prominent role of the liquid for the containment behaviour. Last, an OpenModelica-containmentFOAM coupling scheme is under implementation to represent the specific safety features of Design-II, such as the pressure suppression system.

KEYWORDS
SMR, CFD, containmentFOAM, two-phase flow, system code coupling

COMPARISON OF A DBA SEQUENCE IN A GENERIC iPWR BETWEEN MELCOR AND ASTEC CODES

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ABSTRACT

In view of the possible licensing needs of Small Modular Reactors (SMRs), due to the envisage short-term deployment, the international scientific community has focused his interest in characterizing the SA integral codes capabilities in simulating SMR behavior during postulated Design Basis Accident (DBA), Beyond Design Basis Accident (BDBA) and Severe Accident (SA) scenarios. One type of SMR is the integral Pressurised Water Reactor (iPWR), which is ready to be licensed as a new build because they start from the well-proven and established large-LWR technology including moderate evolutionary design modifications to increase the inherent safety of the plant. However, despite the inherent increase of the safety of the plant, with the adoption e.g. of passive safety systems and integral configuration, a sound demonstration of iPWR ability to address SA should be carried out along a safety review process. Therefore in order to speed-up iPWR European licensing process, the Horizon Euratom project "Safety Analysis of SMR with Passive Mitigation strategies – Severe Accident" (SASPAM – SA), coordinated by ENEA (Italy), investigates the applicability and transfers of the operating large LWR knowledge and know-how to iPWRs in view of SA and Emergency Planning Zone (EPZ) European licensing analyses needs. In this framework, a code-to-code benchmark exercise between MELCOR and ASTEC integral codes has been conducted by ENEA, considering a generic 300 MWe iPWR. Due to the integral configuration a SBLOCA event has been

considered as Postulated Initiating Event (PIE) of the DBA sequence. Main goal of the work is to evaluate the capability of the codes in simulating the dominat thermal-hydraulic phenomena in a generic iPWR design and assess the possible encountered discrepancies.

Analysis of Postulated Severe Accidents in Generic Integral PWR Small Modular Reactors in the frame of the Horizon Euratom SASPAM-SA Project

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Abstract

Currently, there is a growing interest in Europe in the deployment of Small Modular Reactors (SMRs). One family of SMR, the Integral Pressurised Water Reactor (iPWR), is ready to be licensed as a new build. Despite the reinforcement of the first three levels of the Defence-in-Depth (DiD), a sound demonstration of iPWR ability to address Severe Accidents (SA) and handle offsite protecting countermeasures in case of radiological releases should be carried out (DiD Levels 4 and 5). In this context, the HORIZON-EUROPE project ‘Safety Analysis of SMR with PAssive Mitigation strategies - Severe Accidents’ (SASPAM-SA), coordinated by ENEA and launched in 2022, aims at at investigating the applicability and transfer of the current reactor safety knowledge and know-how for large Light Water Reactors (LWR) to iPWRs, in the view of the European licencing needs related to SA and Emergency Planning Zone (EPZ) analyses.

Elements not addressed in other on-going iPWR initiatives are the pillars of the project: a) identification and analysis of postulated SA scenarios; b) identifying the RPV and containment conditions characterizing iPWR SA scenarios that might differ from large-LWRs; c) investigating existing experimental database for iPWRs and additional needs; c) capability of European and non-European (but widely used in Europe) codes for SA and radiological impact on-/off- site analyses in iPWRs.

For maximizing knowledge transferability and project impact, two generic iPWR concepts are considered: ‘Design-1’ (60 MWe) with a submerged containment; ‘Design-2’ (300 MWe) employing several passive systems and a dry containment. These two iPWR designs incorporate the main design features of the most promising designs ready for deployment in the European market.

This paper focuses on the activities of the ‘Input deck development and hypothetical SA scenarios assessment’ Working Package (WP) 2, coordinated by KIT. It aims at assessing generic, but representative, SA and CFD codes’ input decks, analyzing the iPWRs’ behaviour in hypothetical SA conditions, and investigating the codes’ capability to simulate the dominant phenomena driving the scenarios. A set of Design Basis Accident (DBA) and SA scenarios are postulated and analysed. It should be noted that since no Probability Safety Assessment (PS) considerations are done in the project (generic designs are considered), the SA scenarios are analysed in terms of severity and not of probability to occur. Major outcomes will be the boundary conditions influencing accident management strategies, like In-Vessel Melt Retention and Filtered Containment Venting, and quantitative bases to estimate EPZs.

In the paper, the results of the analyses of postulated DBA and SA scenarios in both iPWR designs performed by means of SA

(AC2, ASTEC, MELCOR, MAAP, EDF-MAAP) and CFD (containmentFOAM, Ansys CFX) codes are discussed. The preliminary results have shown that SA codes (AC2, ASTEC, MELCOR, MAAP) can reproduce the main thermal-hydraulics and in-vessel degradation phenomena during the scenarios, showing consistent values and trends in the primary variables of the sequences analysed, i.e. main events timing, hydrogen mass production, and corium mass.

Keywords: SMR, Severe Accident, Design Basis Accidents, EU project

UPDATE ON SEVERE ACCIDENT ANALYSIS RESEARCH AT CNL FOR SMALL MODULAR AND ADVANCED REACTOR DESIGNS

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ABSTRACT

The drive for carbon-free energy to combat climate change has generated increased interest in nuclear energy, specifically small modular reactor (SMR) and advanced reactor (AR) technologies. Research is conducted at Canadian Nuclear Laboratories (CNL) to support the deployment of these technologies, including modelling and experiments related to reactor safety. These efforts are part of a robust technical program supporting research, development, and licensing activities for SMR/ARs for remote (off-grid) locations and grid-level applications. Severe accident analysis and the prediction of radionuclide source terms are a major portion of this work informing design, regulatory interest, and emergency response planning. CNL has refined its technology-agnostic approach to focus on technologies nearest to deployment but continues to follow a progression of i) determining knowledge gaps through phenomena identification and ranking table (PIRT) exercises; ii) tailoring research projects to address these gaps, and iii) evaluating postulated accident scenarios, source terms and impacts on emergency response. Through these activities, CNL builds capabilities to support understanding of severe accidents in SMR/AR designs to service both government and industry stakeholders. The extensive experience gained in CANDU analysis, broad experimental capabilities, and innovative new test apparatus enable CNL to build this competence in SMR/AR safety analysis and severe accident assessment. Technologies under investigation include water-cooled designs (integral pressurized water reactor, iPWR; and small boiling water reactor, BWR), high temperature gas reactors (HTGR; including micro-reactor designs), molten salt reactors, sodium fast reactors (SFRs), and heat-pipe designs. Recently completed and in-progress experimental investigations include the behaviour of fission products in small water-cooled reactor containments, air ingress into HTGR cores, passive cooling systems in water-cooled and other designs, heat pipe behaviour in operational and accident conditions, the behaviour of fission products in molten salt fuels under accident conditions (i.e., release/retention), and UO_2 corium behaviour and molten corium-concrete interaction. Modelling includes design analysis to support experiments, benchmarking activities, and integrated severe accident analysis of SMR/AR designs to determine accident progression, source terms and dose consequences. Efforts have also been made to inform criteria and procedures for emergency planning and response to potential accidents, including consideration of remote and northern operations, and the proximity to populations and/or industrial processes. Guidance on the technologies of focus and the depth and urgency of research comes from the evolution of the Canadian SMR/AR deployment landscape, such as proposed new builds and vendor design reviews by the Canadian Nuclear Safety Commission. As Canada's national nuclear laboratory, CNL's research supports industry (utilities, vendors, suppliers) and government stakeholders in deploying SMR/AR technologies in Canada.

KEYWORDS
Small Modular Reactors, source term, molten salt, high temperature gas cooled reactors (HTGR), integral PWR, small BWR

APPLICATION OF THE PROBABILISTIC METHOD TO PROPAGATE INPUT UNCERTAINTY ON A DBA SEQUENCE IN A GENERIC iPWR

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ABSTRACT

In view of the possible licensing needs of Small Modular Reactors (SMRs), the international scientific community is focusing on investigating the applicability of Severe Accident (SA) integral codes in simulating SMR response during postulated transients. Among the SMRs designs, the integral Pressurized Water Reactor (iPWR) concept, starting from the well-proven and established large Light Water Reactor (LWR) technology, is characterized by the adoption of passive safety systems and integral configuration. Within this framework, the Horizon Project "Safety Analysis of SMR with Passive Mitigation strategies – Severe Accident" (SASPAM – SA), coordinated by ENEA (Italy), is focused on investigating the applicability and transfer of operating large LWR knowledge and know-how to iPWRs in view of SA and Emergency Planning Zone (EPZ) European licensing needs. In this framework, the present study aims to provide initial insights on the application of Best Estimate Plus Uncertainty (BEPU) approach to SMR and about MECOR code uncertainties on the main thermal-hydraulic phenomena occurring in a generic iPWR..

KEYWORDS
SASPAM-SA; SMR; MELCOR; BEPU

Comparison between EDF MAAP5.04 and ASTECv3 codes on an hypothetical Severe Accident on the ELSMOR project NUWARD-Like SMR Design

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Abstract

This paper presents a comparison between EDF MAAP 5.04 and ASTECv3 codes for a hypothetical Severe Accident leading to core degradation on the ELSMOR project proposed NUWARD-Like SMR Design.

The ELSMOR (towards European Licensing of Small Modular Reactors) project is a Horizon 2020 Euratom project. The consortium includes 15 partners from 8 European countries, involving research institutes, major European nuclear companies and technical support organizations. The 3.5-year project, launched in September 2019, investigates selected safety features of Light-Water (LW) SMRs with focus on licensing aspects.

The Modular Accident Analysis Program (MAAP) is a deterministic code owned and licensed by Electric Power Research Institute (EPRI) that can simulate the response of light water moderated nuclear power plants during accidental transients for Probabilistic Risk Analysis (PRA) applications. It can also simulate severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

EPRI MAAP 5.04 does not enable to model SMRs: this code has been adapted by EDF (EDF MAAP 5.04) to make it compatible with the simulation of Severe Accidents transients for the ELSMOR project SMR design.

ASTEC v3 is a severe accident system code developed by IRSN aiming at source term evaluation. The code is used to evaluate major nuclear accidents for different nuclear installations with a main focus on western light water reactor designs. A wide range of phenomena are represented, in particular thermal-hydraulics and core degradation while modules dealing with fission products behavior and molten corium concrete interaction are not yet used in this study.

The comparison performed between EDF MAAP5.04 and ASTECv3 includes the transient evolution from the initiating event (a Station Black out), the core degradation and hydrogen generation, the corium relocation to the Lower Plenum and the In-Vessel Melt retention. The physical phenomena in the containment are also compared (steam condensation on the walls…). Potentiality of H_2 combustion related to the specific assumptions of the selected transient is evaluated through plotting flammability diagram and a sensitivity analysis to N_2 injection for inerting the containment is assessed.

Keywords: MAAP, ASTEC, SMR, H2, IVR

ASTEC VALIDATION OF SFP DEWATERING USING RESULTS FROM THE DENOPI PROJECT

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ABSTRACT

Loss of cooling in a Spent Fuel Pool (SFP) of a nuclear power plant can lead to the melting of fuel assemblies and to strong radiological consequences to the environment. In order to study the first phases of such accidents, up to the fuel assemblies uncovery, the DENOPI project was launched by the French Institute for Radiation Protection and Nuclear Safety (IRSN) supported and funded by the French Government and partners. Among the different facilities developed in the project, the MIDI facility aims at studying the complex thermal-hydraulics phenomena occurring in a large water pool heated from the bottom by electrical rods arranged in dedicated racks. MIDI is scaled by homothety to a typical French SFP. Different assembly arrangements (loading patterns) have been tested at different power levels, with either uniform power repartition, or hot and cold cells. In each test, the water level and temperatures at different elevations are followed, as well as mass flow rate entering each fuel rack. These experimental results also provide relevant data for the analysis and understanding of large natural convection loops that are expected in immersed passive heat removal systems of Small Modular Reactors. The forthcoming OECD/NEA POLCA project aims to extend such results database, in particular to assess the capability of thermal-hydraulics codes to reproduce the main tendencies of these experimental results.
The ASTEC code developed by IRSN is a system code dedicated to the simulation of major accidents in nuclear facilities that may lead to the release of radiological material. Recent works within the MUSA European project have shown the importance of reducing models uncertainties in the first phases of the accident, during the pool dewatering. In this paper, first simulations of MIDI tests are performed with ASTEC in order to assess and improve the capability of ASTEC to simulate the dewatering of a large water pool such as a SFP during a loss-of-cooling accident. Simulations are performed for a selection of MIDI tests with different heating patterns and power levels. Different models of subcooled boiling models from the literature are tested in ASTEC, stressing the key role of these models for an accurate prediction of the experimental flow.

KEYWORDS
Spent Fuel Pool, Loss-of-Cooling, ASTEC, subcooled boiling, natural convection.

Correlation Development for the Determination of Aerosol Particle Retention in Liquid Pools

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ABSTRACT

For the estimation of the particle retention in a liquid pool, a correlation is under development that is suitable to be implemented in the software package AC² developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH. Within AC², the Containment Code System COCOSYS is used for the simulation of ex-vessel phenomena in the frame of postulated severe accident scenarios in light water reactor containments. As a lumped parameter code COCOSYS is characterized by a low simulation time compared to other simulation methods such as CFD. Validation work shows a significant underestimation of the retention efficiency by the pool scrubbing model implemented in COCOSYS compared to experimental data. Therefore a new model is being developed.
Considering that the quantification of the contribution of individual deposition mechanisms to the particle retention in pool scrubbing is not concluding solved due to a lack of measurement options, analytical methods and evaluation procedures were conducted to determine the dependence of retention efficiency on different boundary conditions. The work indicates a high dependence of the retention efficiency on particle diameter and submergence of the aerosol injection in the liquid pool. Essential boundary conditions can be reproduced via dimensionless numbers, so that interacting effects can be taken into account. The input parameters of the new correlative approach are provided by COCOSYS and a calculation of the particle retention is possible without significant user influence. The correlation was developed on experiments with different types of test facilities, injection devices and boundary conditions, so that the approach is not limited to any specific configuration. First calculation results show high agreement with experimental data.

KEYWORDS
Pool Scrubbing, Particle Retention, COCOSYS, Correlation, Validation

IN VESSEL MELT RETENTION 0D MODEL FOR INTEGRAL PRESSURIZED WATER REACTORS

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ABSTRACT
The In Vessel Melt Retention (IVMR) strategy developed for Light Water Reactors (LWR) is an appealing solution for the mitigation of many Severe Accident (SA) scenarios. Allowing the corium to remain in the Reactor Pressure Vessel (RPV), significantly reduces the stresses on the containment. This approach is typical of innovative reactors like the AP1000 and the Hualong One, both GW-size Pressurized Water Reactors (PWR). In this paper, the IVMR application for integral PWR (iPWR) is investigated. These kinds of reactors are characterized by an electrical power output generally lower than 300 MWe, an integral layout and the use of passive safety systems. During a postulated accidental sequence that leads to fuel melting, the molten corium relocates down to the lower plenum of the RPV. By flooding the reactor cavity where the RPV is immersed, the external face of the vessel is cooled, and corium is maintained inside the lower head. The overall phenomenology occurring during the IVMR in an iPWR is similar to the phenomenology in a high-power LWR, but a few differences can be highlighted due to the different design features. Among the differences, it is interesting to notice that the molten pool is rather shallow and does not occupy a full hemisphere. This has an impact on the heat flux profile along the vessel wall. Another difference is the rather thick oxide crust, for which the modelling approach is adapted with respect to what is usually done for the analysis of high-power reactors. The purpose of this paper is to develop a 0D model for the steady-state analysis of idealized cases in an iPWR. It has been developed in the framework of the SASPAM-SA Horizon Euratom project. It is part of the preliminary activities of Work Package 4, which examines the evaluation of the safety margin for several existing iPWR designs, considering available data and plausible assumptions for the unknown parameters. The aim of this model is to include the specific features of iPWRs and provide first estimates of safety margins. Heat transfer correlations depending on the aspect ratio H/R are used to consider the shallow geometry of the pool. The absence of a designed channel or baffle around the vessel requires the use of pool boiling correlations for the external cooling. This work will also prepare the next steps of SASPAM-SA, which is the identification of models in SA codes needing some improvements. Integral reactor calculations focused on the IVMR are also planned: they will be compared to the results of the present 0D approach. Finally, we mention that the developed model is not design specific: it may be easily adapted to various designs by changing some geometrical or material parameters. The 0D model was implemented in Python. This tool will be made available for other partners for applications to other designs of interest.

KEYWORDS
SMR, SA, IVMR, Integral PWR, SASPAM-SA

Assessment of pH-values in water pools during severe accidents in PWR using the lumped parameter code COCOSYS

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ABSTRACT

The pH-value in water pools during a severe accident has an important influence on the release and consequential distribution of iodine in the containment. Additionally, its correct determination in severe accident analyses improves iodine distribution assessments, which facilitates equipment qualification, development of operating procedures and quantification of potential releases. Additionally, the pH-value plays a major role in the degradation of materials in contact with water.

The paper delves into the analysis of pH-values in large water pools and puddles within the containment during severe accidents. Focusing on the utilization of the lumped parameter code COCOSYS, that is part of the integral code system AC² developed at Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) [1], the paper presents its capabilities in calculating pH-values by integrating detailed distributions of pH-relevant species for a generic GenIII+ PWR. Most of these species are released from the primary circuit and corium, such as boron, lithium, cesium. Other species are created during the severe accident such as CO_2 or HNO_3. A primary emphasis is placed on the role of HNO_3, generated through radiolysis throughout the duration of the accident. The creation of HNO_3 contributes to lower pH-values in water pools for a long duration, thereby becoming a major species influencing long-term pH within the containment.

The paper demonstrates the influence of several key events in the accident sequence (e.g., start of mass and energy release, reactor pressure vessel failure, corium quenching, boiling of the water pool above the corium and containment spraying), on local pH-values. Furthermore, it illustrates the capability of COCOSYS to assess pH-values on compartment level, providing an understanding of the pH dynamics induced by the presence of different pH-relevant species, without delving into the specific calculation equations of the pH in water pools or the formation and deposition of the species.

The paper shows the nature of pH variations in water pools during severe accidents as a result of the distribution of different species, highlighting the role of HNO_3 and demonstrating the capability of the lumped parameter code COCOSYS in predicting pH-values in severe accidents in the short and long term.

KEYWORDS
pH, COCOSYS, Lumped Parameter

Modeling of Oxidation Behavior of Accident Tolerant Fuel by using AC²

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ABSTRACT

Postulated and unmitigated accidents in a nuclear power plant could endanger the facilities integrity due to escalating temperatures, rising pressures and released combustible gases. Regarding the barriers of the defense-in-depth concept, the fuel cladding could be a major contributor affecting the accident progression. Although the commonly used alloys based on Zirconium (Zr) provide excellent performance under long-term operational conditions, its limited beneficial behavior at higher temperatures under accident conditions may lead to a fast accident progression along with high hydrogen production. Several studies have shown that other materials could partially improve the claddings behavior under accident conditions. Currently, there are several accident-tolerant concepts that are leading the development of Accident Tolerant Fuels (ATF), like e.g. Iron-Chromium-Aluminum alloys (FeCrAl), claddings based on silicon carbide (SiC-SiC composites) or the coating of existing and widely used Zr-alloys.

In this paper modeling the oxidation resistance is the focus of the extension of AC², a system code package for simulating various phenomena as well as complete accident sequences of nuclear power plants, developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH. The major outcome is a first newly developed in-house ATF model at the PSS Group at RUB regarding the oxidation of Chromium (Cr)-coated claddings. The main motivation is its increased international use in reactors to qualify market readiness as well as new experimental investigations at Karlsruhe Institute of Technology (KIT) during the QUENCH-ATF project. Within a nationally funded research project regarding model development for system codes by PSS, the scope of ATF, among others, is therefore of particular interest with regard to the potential benefit by upgrading existing plants or projected reactors.

There are currently two modeling options for FeCrAl available in AC². One for the calculation of a fixed composition and the other with flexible input for own FeCrAl related correlations. The corrosion resistance of both concepts depends highly on thermodynamically stable oxides, mainly Al_2O_3 for FeCrAl and Chromium oxide (Cr_2O_3) for Cr-coated Zr-alloy, which are formed during oxidation. New findings resulted in further developed FeCrAl correlations, with different dominant kinetics in various temperature regions. This correlation by KIT is implemented in an PSS in-house version of AC².

Regarding Cr-coating, sensitivity studies and the comparison with experimental data resulted in various suggestions for improvements of the in-house Cr-coating model. The oxidation process relies on different influential parameters like coating quality, thickness and degradation phenomena. The diffusion of oxygen into the binary system of Cr/Zr as well as diffusion processes between the coating and substrate result in a loss of protectiveness of the coating, eventually enhancing the oxidation kinetics. Furthermore, mechanical stresses arising from external factors, such as thermal expansion and mechanical loading, play a crucial role in coating failure. Pressure induced ballooning can lead to cracking, ultimately resulting in local coating failure at e.g. burst regions.

KEYWORDS
AC², ATHLET-CD, ATF, Cr-coating, Oxidation, In-Vessel

FORMULATION OF MATERIAL PROPERTY FORMULA FOR CALCULATION OF DAMAGE IN REACTOR PRESSURE VESSEL DURING ACCIDENT EVALUATION

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ABSTRACT

From the results of the internal investigation of Fukushima Daiichi Nuclear Power Station Unit 2, it was confirmed that part of the fuel assembly (upper tie plate) had fallen to the bottom of the pedestal periphery. From this result, it could be presumed that RPV (Reactor Pressure Vessel) has a hole large enough for the upper tie plate to drop. One of failure mode of the RPV lower head would be assumed to be “mechanical failure”. In the mechanical failure, it is assumed that the RPV lower head will be damaged due to the accumulation of creep damage caused by core material above the creep temperature of the RPV substructure materials falling into the lower plenum. Such damage evaluation is performed by thermohydraulic-structure coupled analysis. Such analysis requires RPV material properties from the creep temperature range to near the melting point. In this study, we obtained the strength data of RPV material including welded joints form the creep temperature range to near the melting point and formulated the material property formulas (elastoplastic stress-strain formula, creep strain formula, creep rupture formula) necessary for mechanical failure evaluation.

KEYWORDS
Fukushima Daiichi Nuclear Power Station, Reactor Pressure Vessel, Material Property, Tensile, Creep,

A REVIEW OF CORRELATIONS OF STAINLESS STEEL OXIDATION IN STEAM, AND MODELING THE REACTION WITH MELCOR

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ABSTRACT

Metals oxidize in contact with steam in severe accident conditions. A lot of research has been conducted on the oxidation of zirconium, but less attention has been paid to the oxidation of stainless steel. However, it has been observed in many MELCOR simulations that steel oxidation can generate significant amounts of hydrogen. The steel oxidation correlation in the MELCOR code is from the year 1966.

Five sets of separate effect tests on stainless steel oxidation in steam above 900 °C were found in the literature. Each of the published oxidation correlations is based on a single set of experiments. In this work, a new correlation was developed by fitting the Arrhenius equation to 14 experiments from four different test facilities. The constants of the new correlation are $A = 4.3\times10^7$ kg_{steel}^2/m^4s and $B = 37\ 800$ K. These can be entered into MELCOR by changing the sensitivity coefficient 1002. The correlations were tested by modeling the separate effect tests and the Fukushima unit 1 accident with MELCOR.

None of the correlations is clearly better than the others. Generally, each correlation can reproduce well those experiments, based on which it has been developed. The MELCOR default correlation is conservative. It gives the highest oxidation rate among all the correlations, and it overestimates the oxidation in all tests, except the Bittel test, based on which it has been developed. The new correlation, developed in this study, can be considered as the best fit to all the published data, overestimating the oxidation in some experiments and underestimating it in others. In the Fukushima calculation, the new correlation gave 24 % less steel oxidation and 6 % less total in-vessel hydrogen generation than the default correlation. These are average values of seven calculations with each correlation, perturbed by small changes to see the effect of the numerical noise.

KEYWORDS
Stainless steel, oxidation, MELCOR

AMHYCO PROJECT OVERVIEW AND FIRST OUTCOMES

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ABSTRACT

A severe accident in a nuclear power plant represents a potential risk to both, humans, and the environment. To prevent and/or mitigate the consequences of such an event, it is paramount to have adequate accident management measures in place to ensure the containment integrity. During a severe accident, one of the potential threats to the containment is the release of combustible gases — especially hydrogen and carbon monoxide — leading to combustion risk in the air and steam-filled nuclear containment building. In modern plants, Severe Accident Management Guidelines (SAMG) as well as dedicated mitigation hardware are in place to minimize/mitigate this combustion risk and, thus, protect the containment integrity to avoid an unacceptable release of radioactive material into the environment.

Advancements in SAMGs are in the focus of AMHYCO, an EU-funded Horizon 2020 project (Euratom 2019/2020, GA No 945057) officially launched on October 1st, 2020. The project consortium consists of 12 organizations (from six European countries and one from Canada) and is coordinated by the Universidad Politécnica de Madrid (UPM). AMHYCO benefits from worldwide experts in combustion science, accident management and nuclear safety in its Advisory Board. Thereby AMHYCO covers the most used PWR reactor designs in continental Europe: Westinghouse-type, Siemens-KWU-type, and VVER-type.

In this paper, the progress of the first three years of the AMHYCO project is presented. A comprehensive bibliographic review has been conducted, providing a common foundation to build the knowledge gained during the project. An extensive set of accident transients was simulated, covering in-vessel as well as ex-vessel phenomena. Based on these simulations, a set of challenging sequences from the combustion risk perspective for the different power plant types was under consideration. At the same time, three generic containment models for the three considered reactor designs have been created to enable full containment analysis simulations with lumped parameter models, 3-dimensional containment codes and CFD codes. To further consolidate the modelling base, combustion experiments and performance tests on passive auto-catalytic recombiners under explosion prone H_2/CO atmospheres were performed respectively at CNRS (France) and FZJ (Germany).

KEYWORDS
AMHYCO, combustion risk, in-vessel phase, ex-vessel phase, Passive auto-catalytic recombiners, containment response, SAMGs

OUTCOMES OF THE EXPERIMENTAL AND NUMERICAL WORK ON THE OPERATIONAL BEHAVIOR OF PASSIVE AUTOCATALYTIC RECOMBINERS IN THE LATE PHASE OF A SEVERE ACCIDENT IN THE FRAMEWORK OF THE AMHYCO PROJECT

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ABSTRACT

The molten core-concrete interaction in the ex-vessel phase of a severe accident in a light water reactor is a source of significant amounts of gaseous products including carbon monoxide. The European project AMHYCO addresses open issues related to the understanding of the impact of carbon monoxide on safety-relevant accident phenomena and on hydrogen mitigation measures. In order to support the simulation of accident sequences and the assessment of safety measures, Task 3.2 aimed to provide realistic models considering the influence of carbon monoxide on the operating behavior of passive auto- catalytic recombiners. The work program included conducting new experiments, analyzing existing experimental data, and further developing numerical models.

The experimental program performed within Task 3.2 involved the REKO facilities at FZJ to study the impact of predicted accident atmospheres including oxygen-lean mixtures and the presence of carbon

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monoxide on the hydrogen recombination efficiency of different generic catalysts. More specifically, criteria were derived to predict the atmospheric conditions leading to catalyst poisoning. At the same time, a unified database including selected datasets from the various OECD/NEA-THAI projects was developed. The available experimental data were used to further advance numerical models such as the manufacturer's engineering correlation for Framatome PARs as well as the scientific models SPARK (IRSN), PARUPM (UPM), and REKO-DIREKT (FZJ).

KEYWORDS
Severe Accident, Hydrogen Mitigation, Passive Auto-catalytic Recombiners, Carbon Monoxide

ASSESSMENT OF UNMITIGATED COMBUSTION RISK IN THE LATE PHASE WITHIN THE AMHYCO PROJECT

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ABSTRACT

The European AMHYCO project (Euratom 20192020, GA No 945057) aims at enhancing the understanding of H_2/CO combustion risk within the containment of a nuclear power plant in the late phase of a severe accident. The goal is to incorporate this knowledge into severe accident management guidelines and recommendations for long-term operation upgrades. Based on a critical review of existing methodologies and practices related to gas combustion risk, as well as the identification of accident sequences in which the H_2/CO combustion risk might challenge containment integrity, experimental investigations are conducted to close knowledge gaps related to combustion characteristics and the operation of passive autocatalytic recombiner under late phase conditions. To prepare the basis for the further assessment and refinement of existing SAMGs, systematic detailed analyses of the most challenging scenarios and possible mitigative measures were conducted for three generic European PWR containment designs, namely KWU, Westinghouse, and VVER. For each reactor type, one Loss of Coolant Accident and one Station Blackout scenario were selected for detailed analyses. Both scenarios cover a range of in-containment atmospheric conditions from potentially flammable at medium pressure up to a steam-inertized atmosphere at high pressure, including the late phase with an active filtered containment venting system.

This paper outlines the employed methodology using a consecutive analysis chain consisting of three levels with increasing detail (system codes, 3D GOTHIC™ and CFD) to assess containment pressurization, efficiency and/or options of individual mitigation measures with respect to H_2/CO combustion risk and equipment and instrumentation survivability. As a common basis, the system code nodalization schemes and 3D models are developed from detailed CADs and a shared geometry database. The paper discusses the status of the work with a focus on the comparative assessment of unmitigated reference cases that will be used in the future to assess the impact and effectiveness of mitigative measures (PARs, sprays, FCVS) on the combustion risk in the late phase. Concluding, challenges and lessons-learned are summarized.

KEYWORDS

AMHYCO, combustion risk, in-vessel phase, ex-vessel phase, Passive auto-catalytic recombiners, containment response, SAMG

HEAT REMOVAL TO LARGE WATER POOLS – MACROSCOPIC MODELLING OF MICROSCOPIC PHENOMENA IN THE SIMULATION CODE AC²/COCOSYS

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ABSTRACT

The work presented here describes the latest progress in modelling devoted to the simulation of heated water pools with AC²/COCOSYS and discusses the benefits and limitations of the pursued macroscopic approach. In the safety concepts of advanced nuclear power plants (NPPs), the (temporary) decay heat removal from the reactor core to large water pools, which may be located inside or outside the containment, is provided for the prevention of postulated incidents and accidents and mitigation of their consequences. Examples include passive primary circuit cooling by heat exchangers immersed in a containment water pool, external containment cooling of small modular reactors (SMR), the in-vessel melt retention (IVMR) concept, and the cooling of core melt in an ex-vessel core catcher as included in reactor designs such as the EPR or VVER-1200. When analysing accident scenarios in plants equipped with the abovementioned safety systems, the code must deal with significant heat-up of large water pools. Temperature stratifications may form and have an impact on containment thermal hydraulics and other processes such as fission product transport. The software package AC² is being developed by GRS for the simulation of all relevant phenomena during normal operation, anticipated operational occurrences, design basis accidents, and design extension conditions (including core melting) in nuclear power plants. The objective of the AC² lumped-parameter code COCOSYS is to simulate all relevant phenomena and processes in the containment using a numerically fast macroscopic modelling approach. In the paper at hand, three major modelling concepts are addressed: First, the modelling of combined liquid/gas-flow junctions to consider a vertical subdivision of the pool into several COCOSYS zones is discussed. In addition to that, an extended model range for heat transfer between hot structures and water, including now nucleate boiling, film boiling and corresponding transition regimes is described. Finally, a simplified 'plume' approach to model an efficient upward heat transfer due to boiling is given. The newly implemented models are validated against experiments focused on heating of water in large pools (THAI SMR and PKL SACO experiments); preliminary post-test calculation results are outlined. Moreover, the new models for heat removal into water-filled containment zones are applied in test calculations using the core catcher model in COCOSYS for the simulation of core melt retention and long-term melt cooling in the event of design extension conditions in NPP with EPR. Based on the experience made with the macroscopic modelling approach, conclusions are drawn for further potential or needs of improvements. These conclusions could be beneficial also for the developers and users of other simulation codes with a similar approach to containment thermal hydraulics, like, e. g., ASTEC.

KEYWORDS
water pool, decay heat removal, boiling, stratification, COCOSYS

VALIDATION OF REKO-DIREKT AND CONTAINMENTFOAM-9 CODE COUPLING USING THAI-HR EXPERIMENTS

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ABSTRACT

During a severe accident in a pressurized water reactor, the release of combustible gases can lead to a potential combustion hazard and an associated pressure increase that may put the containment integrity at risk. To study the H_2/CO transport dynamics at reduced scales, experimental campaigns such as the OECD/NEA THAI or the THEMIS projects, have been conducted in the last decades. Specially, THAI-HR experiments explored the mitigation efficiency of several types of Passive Autocatalytic Recombiners (PARs). In the framework of AMHYCO project (Euratom 2019-2020), one of the modeling approaches to assess combustion risk management will make use of the tailored CFD package containmentFOAM, developed at Forschungszentrum Jülich GmbH (FZJ) and coupled with the detailed PAR model REKO-DIREKT. This work presents an iteration on the verification and validation methodology of that code coupling by simulating a selection of THAI-HR tests (between HR-3 and HR-12) in a simplified test-case environment. Thus, a representative range of severe accident conditions, namely vessel pressure, temperature, and gas concentrations, are evaluated from the inlet to the outlet computational domains of the PAR model. The objective is to prove the capability of REKO-DIREKT to reproduce the experimental measurements and to deliver the figures of merit of the recombination process to the containmentFOAM model, this being a preparatory step towards validating the simulations in a complete 3D THAI vessel model in containmentFOAM.

In general, the simulations catch the tendencies of the thermophysical properties of the vessel atmosphere and the general PAR performance characteristics. Nonetheless, the simulations show some deviations with respect to the measurement in the PAR flow rate when replicating the THAI conditions for low pressures and at oxygen starvation conditions. This occurs during the fast gas injection phases and affects the recombination rate and PAR temperatures. Finally, some refinements might be needed in the REKO-DIREKT modules to better capture the mass flow rates entering the PAR, thus replicating PAR efficiency more accurately.

KEYWORDS
AMHYCO, containmentFOAM, CFD, Passive Autocatalytic Recombiner, Severe Accident

THAI EXPERIMENT ON THE IODINE ABSORPTION CAPACITY OF PRE-STRESSED PAINT COATINGS ON DIFFERENT SURFACES

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ABSTRACT

In the aftermath of severe nuclear accidents, the presence of fission products, particularly iodine, within the containment as gases or aerosols poses significant challenges to radiological safety. This has a strong influence on the radiological source term, including the dose rate distribution, corresponding decay heat, and formation of volatile compounds. This study investigates the impact of pre-stressed decontamination coatings on the efficiency of iodine depletion from the gas phase under extreme conditions. The experiment was conducted at the THAI facility in a closed and superheated containment configuration. Twenty stainless steel panels coated with GEHOPON epoxy-based decontamination paint were placed inside the THAI test vessel. Two cylindrical steel coupons were used for on-line monitoring of the iodine deposition process. One coupon was painted with Ripolin epoxy paint, while the other was unpainted. Additionally, a concrete block painted with GEHOPON was used.

The experiment involved short-term, high thermal loads on the paint surfaces, which were introduced by a hydrogen deflagration. Various measurement techniques were used to continuously monitor aerosol and iodine concentrations, including gas scrubbers, Maypacks, bulk filters, impactor measurements, and optical systems such as the light absorption system for iodine (LASI-SPEC) and laser extinction.

The hydrogen deflagration caused mechanical and thermal damage to the paint surfaces, resulting in the release of volatile organic compounds and aerosols. Airborne concentrations were closely monitored for 76 hours to measure the deposition behavior, including 25 hours of radiolabeled molecular iodine (I-123) deposition on painted surfaces and inner walls of the THAI test vessel. Surprisingly, the formation of aerosol-bound iodine was found to be significantly stronger than anticipated.

The results contribute to a deeper understanding of thermal-hydraulic conditions, damaged paint behavior, and airborne fission product concentrations in the late phase of an accident. These findings have implications for the fundamental understanding of severe accident scenarios, as the experimental data provide valuable insights for the development and validation of detailed reaction models in severe accident codes. The study underscores the importance of considering pre-stressed decontamination coatings in enhancing safety measures and mitigating the impact of fission products in the aftermath of severe nuclear accidents.

KEYWORDS
Severe Accident, Containment, Iodine, THAI, Decontamination Paint

VALIDATION OF THE PARUPM AND GOTHIC 8.3 CODE COUPLING USING THAI HYDROGEN RECOMBINATION TESTS

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ABSTRACT

In case of a severe accident in a nuclear power plant (NPP), large amounts of H_2 and CO could be generated, potentially leading to uncontrolled combustion if concentrations within the flammability thresholds are reached. To mitigate this hazard, many NPPs equipped their containment buildings with passive autocatalytic recombiners (PARs). PARs are capable of recombining combustible gases such as H_2 or CO with oxygen, resulting in the production of steam and/or CO_2, even when gas concentrations fall below the flammability threshold.

The AMHYCO project (Euratom 2019-2020, GA Nº 945057) aims to advance experimental understanding and simulation capabilities for H_2/CO combustion risk management in severe accidents. Within the framework of this project, the computational model PARUPM, which uses a physicochemical approach to simulate PARs, has been validated as a standalone tool using experimental data obtained from both the REKO-3 flow channel at FZJ and the OECD/NEA THAI projects executed at Becker Technologies GmbH. In a next phase, PARUPM has been integrated as an add-on program within the thermohydraulic simulation code GOTHIC. This code coupling facilitates the evaluation of the interaction between the operation of PARs and the surrounding atmosphere in which they are located.

The present paper provides an overview of the capabilities of the joint simulation with PARUPM – GOTHIC 8.3. The coupling to GOTHIC has been validated based on a sequence of experiments on H_2 recombination by PARs performed at the THAI experimental facility. The relevant feedback from the THAI vessel to the recombiners performance involving gas stratification and change of boundary conditions is studied by a 3D model which reproduces the complex behaviour of gases inside the THAI vessel during the experiments.

KEYWORDS
AMHYCO, Passive Autocatalytic Recombiners, Hydrogen, GOTHIC, THAI, REKO, PARUPM

EVALUATION OF HYDROGEN RISK AND ITS MITIGATION STRATEGIES ADOPTED IN ISOTOPE MANUFACTURING BUILDING

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ABSTRACT

Nuclear power plants produce isotopes as a support function for the growth of the nuclear technology field and to aid in industrial detection, such as the generation of ^{14}C and ^{60}Co. After the generation process, the rod bundles equipped with such isotopes should be put into a water pool to inhibit the radioactive release, lowering the dosage of the building where such isotopes were manufactured. However, the radioactive source of those isotopes can result in hydrogen produced by the irradiation of source product pool's cooling water, which could increase the risk of hydrogen combustion and explosion in a particular area of the building. The Spent Fuel Pool (SFP) may also have such hidden danger. The three-dimensional isotope manufacturing building and its primary internal structure (source product pool with hydrogen overflow channels, forklift room, venting pathways, etc.) were finely modeled in this study by using the Computational Fluid Dynamics (CFD) approach. By reviewing the progress of numerical research for hydrogen flow and distribution in severe accidents and hydrogen mitigation measures in other hydrogen energy industries all over the world, appropriate boundary conditions were selected to simulate the time-space distribution of hydrogen within the 3-D configuration, which shows hydrogen stratification in vertical direction and local gathering phenomena near the surface of the pool reflecting pronounced non-uniformity within the building. In order to quantify the effects of various hydrogen risk mitigation strategies, the introductions of the ventilation, air ingress, and emergency ventilation system were taken into consideration. It was discovered that while turning on the building's emergency venting system at the roof may greatly reduce the hydrogen risk, and the upper space of the water pool is a crucial place where hydrogen explosion may occur. Last but not least, the hydrogen risk inside the building can be reduced and alleviated by suitably adjusting the air permeability of the canvas that covers the pool, which could serve as a technical reference for the design of an isotope manufacturing building, and, moreover, for the development of new types of hydrogen mitigation strategies for SFPs and other industrial production.

KEYWORDS
Hydrogen distribution, source products pool, hydrogen risk, mitigation, isotope manufacturing building

The Scaling of Turbulent Flame Acceleration and Detonation Transition for Hydrogen-Air mixtures in the RUT Facility

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Abstract

To model the Loss of Coolant Accident (LOCA) in a containment of nuclear power plant (NPP) a series of large scale bench-mark experiments have been conducted in the RUT facility. Flame propagation regimes for very lean hydrogen-air mixtures from 10 to 14% H2 have been investigated in the RUT set-up. The facility consists of three parts: the first channel (34.4 x 2.5 x 2.2 meters), the canyon (10.5 x 2.5 x 2.2 m) and the second channel (20.1 x 2.5 x 2.2 m) Total volume of mixture was about 480 m^3. 30 and 60% of the channel cross-section was blocked by concrete blocks. Slow and sonic deflagrations in the channel have been established for hydrogen compositions up to 12.5%H2 as well as a detonation transition at 14% of hydrogen was registered. In a canyon of a bigger dimension, the detonation was observed even at 12.5%H2 due to shock reflection at the far corner of the canyon. Such detonable concentrations for channel and canyon geometries significantly extend the well-known conventional detonability limits of 18-59% H2 (NASA STD 8719.16 - Safety Standards) and demonstrate the very high danger of the detonation even for very lean hydrogen-air mixtures.

Since the critical conditions for DDT are usually analyzed in terms of the dimensionless ratio of the characteristic size of the system to detonation cell size l of the mixture (as a measure of mixture sensitivity to detonation initiation), the scaling down of flame acceleration and detonation transition was investigated for similar geometry in a MINIRUT facility of 50 times smaller size (1:50) than the original RUT facility. The advantage of the small-scale facility was that it allowed a high-speed video monitoring of flame acceleration and detonation transition. The detonation in the channel and in the canyon of 50 times smaller scale has occurred for hydrogen-air mixtures with a detonation cell size of 50-57 times smaller than for a large-scale RUT facility. The known criterion 7l for detonation onset in an obstructed channel was experimentally confirmed. The video monitoring confirmed the shock reflection and shock-flame interaction mechanisms for deflagration to detonation transition (DDT). Beyond the detonation cases for leaner hydrogen-air mixtures, the flame can propagate as subsonic or sonic deflagration. The criterion for sonic deflagration will be the critical expansion ratio s = 3.75 which corresponds to 10.5-11%H2. The experiments also confirmed that the flame dynamic for deflagration mode does not scale down as the DDT process. Even more, below s = 3.75 the flame can propagate faster in narrower channels at the same hydrogen concentration.

Keywords: hydrogen, LOCA accident, severe accident, flame acceleration, detonation transition

The Reduction of Radiological Consequences of design basis and extension Accidents: re-assessment of calculations and main outcomes of the R2CA project

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ABSTRACT

The 4-year H2020 R2CA (Reduction of Radiological Consequences of design basis and extension Accidents) project was intended to propose guidelines to harmonize the safety analysis methods through the development of generic methodologies for best estimate evaluations of the radiological consequences (RC). It was dedicated to Design Basis Accidents (DBA) and accidents in the Design Extension Condition domain without significant fuel melting (DEC-A). The project addresses a broad scope of LWR designs (BWR, EPR, PWR, VVER) from Gen II, III and III+ focusing on bounding scenarios of Loss Of Coolant Accidents (LOCA) and Steam Generator Tube Rupture (SGTR) transients.

The work performed within the project intends to reduce some of the conservatisms currently used in safety studies and/or licensing calculations, which are necessary to help a better quantification and consideration of the potential changes to come in the operation conditions (e.g. increase of fuel burn-up, usage of ATF). In doing so, some potential higher risks exhibited by knowledge improvements (e.g. clad embrittlement through secondary hydriding, increased fission product releases from high-burn-up or MOx fuels, etc.) were also highlighted. Finally, it was beneficial to the development of innovative measures or tools to prevent these accidents (i.e. algorithms, AI based tools, etc.) as well as to strengthen their management.

This article summarizes the main outcomes of the R2CA project presenting the most relevant improvements in simulation tools and calculation methodologies therein. LOCA and SGTR calculation results in DBA and DEC-A domains are discussed. In addition, some insights are given on tools/methods that could be further used to increase the NPP safety based on exploratory innovative work performed within the project. Finally, in conclusion, preliminary guidelines are given for a better estimation of the RC of the considered scenarios.

KEYWORDS
LOCA, SGTR, DBA, DEC-A, Radiological Consequences

PROGRESS IN UNDERSTANDING FISSION PRODUCT REMOBILIZATION AND HYDROGEN RISK IN WATER-COOLED REACTORS: OECD/NEA THAI-3 PROJECT

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ABSTRACT

OECD/NEA THAI-3 project aimed to investigate hydrogen risk and source term related issues with specific emphasis on representative boundary conditions as expected during a severe accident in light water reactors. The project was conducted between 2016 and 2019, hosted by Germany and supported by signatories from 16 countries. The experimental program was conducted in the technical-scale THAI$^+$ facility, comprising two interconnected vessels with a total volume of 80 m^3.

The hydrogen combustion tests investigated the impact of varying initial flow conditions, gas composition, and burn direction on pressure buildup, flame front propagation, and jet-ignition effects during H_2-deflagration. The test HD-44 served as a benchmark for validating combustion models. Hydrogen recombiner tests provided data on recombination onset, recombination rate, and hydrogen depletion efficiency under counter-current flow conditions, with the test HR-49 used for benchmark code calculations. The source term related experiments examined the re-entrainment of fission products (CsI, I_2) from water pools, showing CsI re-entrainment increased tenfold in the case of water pool with reduced surface tension. The results of fission product resuspension tests revealed significant resuspension of fission products from paint and steel surfaces during hydrogen deflagration with the potential to remain gasborne over a long time ("fine particles"), with notable release of organic iodine and decomposition of CsI to I_2 in the high-temperature environment.

This paper summarizes the key findings of the project and emphasizes their significance for mitigating hydrogen risk and source term issues. It also highlights the use of project results for code validation and reactor analyses to manage and mitigate severe accidents in LWRs.

SOURCE TERM ASSESSMENT FOR A VVER-1000 REACTOR EQUIPPED WITH FILTERED VENTING: SENSITIVITY STUDY OF THE IMPACT OF DIFFERENT FORMS OF IODINE IN THE CONTAINMENT

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ABSTRACT

After the Fukushima accident, both Czech NPPs Dukovany and Temelin have been enhancing their safety with regard to severe accidents (SA). As a measure of preventing containment (CTMT) over-pressurization, filtered containment venting system (FCVS) is to be implemented at the Temelin NPP (equipped with 2 VVER-1000 reactors). In order to specify its design requirements, SA analyses were conducted to assess the amount of fission products (FP) to be drawn onto the FCVS filters while in operation. Results of such analyses are extremely sensitive to the radionuclides' (RN) speciation within the CTMT, especially iodine. The integral code MELCOR considers iodine in the form of CsI by default. However, certain fraction of iodine remains in the molecular (and thus gaseous) form (I_2). To best-estimate such fraction, a CTMT analysis with the dedicated COCOSYS code was performed and the integral MELCOR simulation was afterwards refined by setting a deduced fixed percentage of I_2 among iodine forms in the CTMT. A sensitivity study was finally performed to assess the effect of the form of iodine on the source term (ST) onto the FCVS filters.

KEYWORDS
Iodine form, MELCOR code, COCOSYS code, filtered containment venting system, source term

Source Term Dispersion Analysis and Construction of the Risk Map around the Peach Bottom Unit-2 Plant Using the ASTEC and JRODOS codes

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ABSTRACT

National regulators and Emergency and Preparedness teams require accurate source term predictions to protect the public from abnormal radiological emergency events like severe accidents (SA) in Nuclear Power Plants (NPPs). Therefore, reliable evaluations of the radiological consequences of hypothetical SAs in a NPP are mandatory. With this goal, a robust first-of-its-kind platform of codes has been developed at Karlsruhe Institute of Technology (KIT) in order to evaluate best estimate source term data and the corresponding radiological consequences following a SA in a NPP. In the platform, the fuel inventory loaded in the reactor core is computed by means of the CASMO5 code and the European reference Accident Source Term Evaluation Code (ASTEC), developed by IRSN, is employed to analyze the SA sequence from the initiation up to the fission product release to the environment. Finally, the source term results computed by ASTEC are employed by the JRODOS code, developed by KIT, to evaluate the radiological consequences of the dispersion of the fission product in the environment.
In this paper, the abovementioned calculation platform is employed to analyze the radiological consequences of an anticipated station blackout scenario (SBO) in a generic Peach Bottom Unit-2 NPP. Fuel inventory has been computed based on the real plant operation history and the ASTEC generic model of the plant has been assessed. The isotope-wise activity released to the environment during the investigated SBO scenario computed by ASTEC is then provided to JRODOS for performing the corresponding dispersion analyses. The 'usual' approach for JRODOS analyses consists of selecting 'representative' dates to evaluate the radiological effects of the source term dispersion. In principle, for a precise evaluation of the radiological risk, one should perform such analyses for each part of each day of the year because of the detailed meteorological data available nowadays, which would take a nonrealistic amount of calculation time.
In order to solve this issue, a methodology was developed to evaluate the radiological consequences of the SA in each geographical mesh around the NPP. A series of analysis, coming from about 60000 JRODOS calculations, have been carried out in each season by assuming that the SA occurs in randomly sampled days and hours. As a result, sample groups of JRODOS results in each mesh grid are obtained. For each mesh grid, the probability distribution of the radiological consequences is computed by employing the average values of each sampling group, the central limit theorem being employed. Finally, a risk map for each season of a full year is constructed to visualize, i.e., the average total effective gamma dose rate for each mesh as a consequence of the SBO SA scenario computed by ASTEC in the generic Peach Bottom Unit-2 NPP. In this way scope of the considered accident over the domain pictured in detail to able to address proper planning for early and intermediate phases of the accident. Therefore, the proposed methodology has the potential to provide meaningful and detailed information for the assessment of the Emergency and Preparedness actions in response to a SA event in an NPP. In the paper, the results for the SBO scenario in a generic Peach Bottom Unit-2 NPP will be shown and discussed.

KEYWORDS
ASTEC, JRODOS, BWR, Source term, Radiological Analysis

EFFECT OF BORIC ACID ON FISSION PRODUCT TELLURIUM AND IODINE IN SEVERE ACCIDENT-LIKE CONDITIONS: ANALYSIS WITH X-RAY PHOTOELECTRON SPECTROSCOPY

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ABSTRACT
Tellurium is a volatile fission product with a rich and complex chemistry. In a severe nuclear accident it will be released alongside a variety of other chemical compounds such as fission products and structural material, and be exposed to a variety of atmospheres. By virtue of its chemistry, tellurium can potentially take the forms of aerosols, organic tellurides or other forms during the accident scenario, and understanding its interaction with common materials becomes important from an accident management perspective. In this work, the interactions between tellurium and boric acid, used as a chemical shim to control fission reaction in pressurized water reactors are investigated. Furthermore, the impact of iodine vapor on this system is investigated. The method involves heating tellurium to its melting point and transporting the vapor in a flow of oxidizing, inert or reducing gas to a reaction furnace where droplets of boric acid solution and/or fission products are added. The gas stream is then directed through a filter. Analysis focuses on XPS (X-ray photoelectron spectroscopy) analysis of this filter in an attempt to determine the aerosolized chemical species that may be present in an accident.
The results indicate that there is little direct interaction between tellurium and boric acid, but the degree of oxidation appears to change with the presence of boric acid. Furthermore, tellurium appears to interact with hydrogen, forming H_2Te, and with iodine to form tellurium iodides.

KEYWORDS
Volatile Fission Products, Containment Chemistry, Boric Acid

LOCAL MEASUREMENTS ON PARTICLE MASS TRANSFER IN GAS-LIQUID FLOWS

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ABSTRACT

Particle mass transfer processes in multiphase flows are widely present in chemical engineering and energy conversion processes. During a severe accident event with core degradation in a nuclear power plant, aerosol pool scrubbing can be used to capture radioactive particles from the venting of a containment. Despite its fundamental nature and the significant effort made by numerous studies in characterizing such processes, accurately validated computational multiphase fluid dynamics predictions of the local particle mass transfer at the bubble scale are still unavailable.

In this study, we developed a new experimental technique based on high-speed tomographic conductivity measurements to visualize hydrodynamics and particle mass transfer in a particle-loaded gas-liquid bubbly flow in a vertical channel. The conductivity measurements are performed with a 1 mm resolution Wire-mesh sensor at 3200 Hz. The injected aerosols are NaCl micrometer diameter particles. The passage of the bubbles is characterized by a signal of lower relative conductivity than the liquid water. From the measured data, we can directly obtain the local value of the void fraction. The size and shape of every single bubble can be measured. Once the aerosol particles are transferred to the liquid phase, they dissolve, and the conductivity signal is proportional to the electrolyte concentration. The mass transfer is visualized in a concentration wake that follows downstream from the bubble. We performed experiments at different aerosol concentrations and particle sizes to assess the effects of the aerosol characteristics on the particle mass transfer.

We obtain first-of-its-kind data of cap ellipsoid bubbles and associated mass transfer wakes. These CFD-grade data results can be used to develop and validate high-fidelity aerosol pool scrubbing numerical codes. The resolution is such that the bubble's geometry can be characterized in detail. The geometry of the wake corresponds to the expected momentum wake behind a blunt body in the subcritical turbulent regime. The method is sensitive enough to capture the changes in the aerosol concentration for micrometer-size particles. Future work will extend the analysis and the measurements to obtain the particle mass transfer-related coefficients.

KEYWORDS
Aerosol, Mass transfer, Multiphase gas-liquid flow, Wire-mesh sensors, Local measurement

NEW EXPERIMENTAL DEVICES FOR SEVERE ACCIDENT STUDY: A LASER BASED APPROACH

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ABSTRACT. Three complementary laser-based platforms, named CHAUCOLASE (French acronym for laser-controlled heating) for concepts validating, CHARTREUSE (French acronym for laser heating for experimental treatments on nuclear samples in extreme conditions) for qualification involving non irradiated nuclear fuel before implementation in hot-lab facilities and finally MALAGA for application on irradiated nuclear fuels are currently developed by the CEA-Cadarache (IRESNE/Fuel study department). This paper highlights how the use of laser technologies makes it possible to increase the experimental capacities of nuclear fuel studies for: (1) the quantification of thermophysical properties as a function of temperature, (2) applying complex temperature transients representative of the RIA and LOCA type accidents. In each case, recent results obtained thanks to the different methodologies developed in close collaboration between the CEA and the Fresnel Institute are presented as proof of concept.

KEYWORDS
Laser based platforms, thermophysical properties, RIA, LOCA

On the progress made in source terms evaluation and possible open issues relative to advanced technologies

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ABSTRACT

Since 2000's, many progresses have been made on the knowledge regarding the source term through different R&D programmes. The outside releases in severe accident (SA) conditions result from different phenomena: releases from degraded fuel, transport and chemistry in the RCS and aerosol physics and chemistry in the containment. Each phenomenon has been studied experimentally to provide the knowledge and data for developing and validating models.
The physical models gained were implemented, for IRSN side, in ASTEC software dedicated to SA scenario simulations for nuclear power plants providing an understanding of the main phenomena involved and finally a more accurate prediction of source term. The source term assessment contributes to enhance the prevention and mitigation of severe accident consequences by evaluating the effectiveness of mitigation systems and the Severe Accident Management Guidelines (SAMGs) effect. In addition, source term evaluations enable informed decision-making during crisis situations by helping to prepare emergency plans.
This paper gives an overview of the progress made over the last decade on ST issues, with new impacting phenomena, as well as a special focus on ongoing IRSN activities, mainly relative to chemical speciation of main fission products in the primary circuit under LOCA or Severe Accident conditions, and chemical reactivity of iodine in the containment.
Impact of new models dedicated to iodine behaviour in the containment coming from last R&D data, is presented using the results of ASTEC calculation of a LOCA scenario for French PWR 1300 MWe.
Finally, some possible open issues linked to new situations (LTO) and/or the development of a set of advanced technologies like Accident Tolerant Fuels (ATF) and Small Modular Reactors (SMR) will be discussed.

KEYWORDS
Source term; R&D, ASTEC, iodine, iodine aerosols

GASEOUS AND AEROSOL FORMATIONS IN THE PSEUDO-BINARY CsI-MoO_3 REACTION SYSTEM

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ABSTRACT

The pseudo-binary CsI-MoO_3 reaction system during transportation in the reactor under severe accident conditions has been investigated by using the JAEA-TeRRa semi-integral test facility. An oxygen gas with approximately 10 mbar of partial pressure was given to argon-steam upstream gas flow during the reaction to study its effect on downstream chemistry between CsI and MoO_3 (Mo/Cs molar ratio of 2.13). The product of CsI-MoO_3 reactions both gas and aerosols were analyzed upon their condensation on type-304L stainless steel sampling coupons at respective temperatures (T= 1150-450 K). Post-test analyses revealed that at a low oxygen potential of -192 kJ/mol (i.e. reference case with argon-steam only), most of the CsI reached downstream (T < 400 K) without any reaction with MoO_3. On the other hand, when oxygen potential was slightly increased to about -144 kJ/mol, the CsI vapor could react with MoO_3 to form cesium polymolybdates ($Cs_2Mo_3O_{10}$ and $Cs_2Mo_4O_{13}$) and gaseous iodine which predominated the aerosol and gas that reached the downstream region. Particle size at this location was found to be less than 2.7 µm in contrast to the former case having an irregularly large size. The gaseous iodine in the latter case, based on the thermodynamic analyses, was estimated to be hypoiodous acid (HIO) or molecular iodine (I_2). The results in this study indicated that speciation of both Cs and I with the Mo chemistry during a severe accident could be variedly formed depending on the prevailing oxygen potential.

KEYWORDS
Cesium molybdates, oxygen potential, gaseous reaction, aerosol attenuation

EXPERIMENTAL STUDY ON REMOVAL EFFECT OF RADIOACTIVE MATERIALS IN THE COURSE OF THE LEAKAGE THROUGH THE EQUIPMENT HATCH

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ABSTRACT

The equipment hatch in the containment vessel of the nuclear power plant is a potential part from which fission products leak into the environment during a severe accident. If a gap height in the leak pathway is small, some fission products in the form of aerosol can be trapped inside the pathway during the leakage. The objective of this study is to clarify aerosols removal effects in the leak pathway of the equipment hatch quantitatively with an experimental approach. The study especially focuses on understanding the dependency of the removal effects on a gap height or a flow rate. The test section represents a partial arc length of the whole circumference of the equipment hatch. It enables to reproduce a leakage with a perpendicular flow to the gasket rings. Silica is used as an aerosol material of which particle diameter is equivalent with that observed in a severe accident. The decontamination factors, DF, which is defined as a ratio of aerosol concentration in the inlet to that in the outlet of the test section, was measured in each test. A wide range of flow rate is considered for experimental conditions not only to cover assumed leak conditions under actual accidents, but also to understand basis phenomenal mechanism. The results showed that DF decreases as flow rate increases. Massive aerosols deposition on the floor of the test section were observed after experiments with a low flow rate while little depositions on the ceiling were observed. The observation supports the consideration that sedimentation mainly contributes to higher DF under a lower flow rate. The results also showed the possibility that tests with a smaller gap height are linked with higher DF. It can be said that a smaller gap height causes a smaller falling distance of a sedimentation and a larger deposition. A series of experiments indicated that DF exceeds approximately several tens under the Reynolds number less than approximately 100 with any gap heights of 0.1 mm to 1 mm. It is noted that the condition of degraded gaskets in the accidents is unknown so far and hypothetical configuration of gasket is assumed in this study. The data obtained in the experiments can be useful to discuss basic mechanism of aerosol removal in the equipment hatch.

KEYWORDS
Severe accident, Fission product, Aerosol, Decontamination factor, Equipment hatch

Preliminary Evaluation of Reactivity Insertion during BDBA-LOCA of Super Fast Reactor

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ABSTRACT
The fast reactor core is not in its neutronically most reactive configuration, and the common challenge is to suppress the compaction reactivity to prevent excessive mechanical energy generation due to recriticality during the core degradation in an anticipated severe accident. The Super Fast Reactor (Super FR) is the fast reactor design concept of the Supercritical Light Water-cooled Reactor (SCWR) being developed at Waseda University. The core is loaded with enriched Pu MOX fuel and cooled by the supercritical water at 25 MPa during normal operation. The severe accident of the Super FR with the system depressurization leads to the core degradation together with the decrease of the coolant inventory, because of the low boiling point of the coolant. Consequently, positive reactivity insertion into the core is expected not only by the core compaction (due to degradation and melting) as expected for any other fast reactors, but also due to the water level decrease, because of the large neutron scattering and moderation characteristics of water. This study aims to reveal the combined effects of the core exposure and compaction on the reactivity insertion during anticipated severe accidents of the Super FR.

The 285 MWe (650 MWth) Small Modular Reactor (SMR) class Super FR with the active core height of 2000 mm and equivalent diameter of 1780 mm has been tentatively selected for the evaluations. A Beyond Design Basis Loss-Of-Coolant Accident (BDB-LOCA) without water injection is considered as an example of the severe accident scenario to give history of the water level decrease during the core degradation process. A 100% hot-leg break was considered as an initiating event. The neutron transport calculations based on the Monte Carlo method (MVP) were carried out with the JENDL-4.0 nuclear data library.

The reactivity changes from the normal operation condition to the core under the above assumptions were analyzed. The water level reduction was the dominant cause of the reactivity insertion until the water level decreased to about 2/3 of the active core. Then, the reactivity insertion due to the core compaction became non-negligible as the water level further decreased from about 2/3 to 1/3 of the active core. Eventually, as the compaction volume expanded, the additional reactivity insertion due to core compaction became less significant until the water level further decreased and the damaged core was fully uncovered.

KEYWORDS
Supercritical Water-Cooled Reactor (SCWR), Super FR, water cooled fast reactor, severe accident, re-criticality

Development of severe accident simulation code for sodium-cooled fast reactors: SIMMER-V

(1) Overview of the SIMMER-V code development

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ABSTRACT

JAEA has been developing the SIMMER-V code in collaboration with CEA to perform severe accident (SA) simulations of future sodium-cooled fast reactors (SFRs) including a unique core design with large-scale heterogeneous cores. An SA sequence in SFRs has been analyzed by: the SAS4A code for the Initiation Phase (IP), in which fuel pin disruption and vertical fuel dispersion occur in individual fuel subassemblies; and the two-dimensional SIMMER-III or three-dimensional SIMMER-IV code for the Transition Phase (TP), in which core disruption extends to the whole core. The joint development of SIMMER-V is of limited scope but aims at significantly expanding the code applicability by providing flexible interfaces to couple a SIMMER-V calculation with other computational domains or other codes, and by adding new advanced physical models such as a detailed fuel pin model and a model of flexible treatment of fuel isotopic composition. The former tasks are conducted by CEA the latter tasks by JAEA. In parallel to the code development, verification and validation of the new models and methods have been performed. This paper describes the objectives and overall framework of SIMMER-V code development program, representative new elements, and recent development progress.

KEYWORDS
Sodium cooled Fast Reactor, Severe accident, SIMMER, Code development

EXPERIMENTATION & SIMULATION OF SURFACE ABLATION BY A JET FOR CORE-CATHER SAFETY ISSUE

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ABSTRACT

These coupled experimental and numerical studies support the safety demonstration of Sodium Fast Reactors SFR severe accident mitigation features. In order to limit potential power excursion in SFR core in the case of hypothetical severe accident, the implementation of the mitigation devices called transfer tubes was planned into the ASTRID core design. More specifically, they connect the core to the core-catcher and enable the early discharge of the corium towards the core-catcher. However, these devices, which effectively reduce the probability of reactivity excursions in the core, accentuate the issue of core-catcher resistance to corium inflow. Differing from past reactor core designs, the presence of these tubes could lead to coherent corium jets impinging on the core-catcher surface. In the past, numerous studies dedicated to thermal ablation of a solid by a jet have been carried out to characterize the maximum ablated depth. Although these experiments are very valuable at a macroscopic scale, they do not give information about the local physical phenomena governing heat transfer and melting. Keeping this final goal in mind, this study followed a R&D methodology based on an analytical situation using simulant materials enabling direct visualization. In this paper, the ablation of an ice block by an immersed water jet and a free surface jet are investigated in conditions shown to be close to the reactor case. Associated to the experimental part, CFD simulations of these tests are performed to validate the calculation methodology (mesh, turbulence models and interface melting…).
In this paper, the physical analysis of the heat transfer at jet impingement and in its vicinity is characterized for immersed jet conditions.

KEYWORDS
RNR Na, core-catcher, jet ablation, experiment, simulation, CFD

OVERVIEW OF KIT ACTIVITIES ON ATF CLADDING MATERIALS

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ABSTRACT

Research on accident-tolerant fuel (ATF) cladding materials is being conducted worldwide to reduce the risk of hydrogen explosions and to extend the response time to severe nuclear accidents. This paper reviews experimental activities at KIT on the behavior of promising ATF cladding materials at high and very high temperatures. These are largely embedded in bilateral cooperations as well as international research projects under the umbrella of the European Commission (IL TROVATORE, SCORPION), IAEA (ATF-TS) and OECD-NEA (QUENCH-ATF).

First large-scale experiments with FeCrAl (QUENCH-19) and Cr-coated Opt. ZIRLO (QUENCH-ATF1) were performed at the QUENCH facility. The main results of these experiments are discussed to the extent permitted by the publication guidelines of the corresponding projects. The results of these experiments have been used for international benchmark tests, which are presented in a separate paper.

The bundle tests are accompanied by an extensive separate-effects test program that provides data on the oxidation kinetics and degradation mechanisms of ATF cladding materials up to very high temperatures. From these experiments, it is found that the maximum temperatures for protective cladding are 1300°C, 1400°C, and 1700°C for Cr-coated Zry, FeCrAl alloys, and SiC-based composites, respectively. Illustrative examples for the three ATF coating systems are shown.

KEYWORDS
ATF cladding, degradation, HT oxidation, bundle tests, separate-effects tests

APPLICATION OF AC²/ATHLET-CD AND ASTEC FOR ATF EXPERIMENTS IN THE FRAME OF ONGOING INTERNATIONAL PROJECTS

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ABSTRACT

Zirconium based claddings have been widely used for decades due to their good mechanical properties, corrosion resistance and low neutron absorption at operational conditions. At elevated temperatures, these claddings have some negative characteristics such as the occurrence of exothermic oxidation reactions in steam or air environments. This has motivated research and development of Accident Tolerant Fuel (ATF) and Cladding (ATC) materials, which began prior to the Fukushima Daiichi nuclear power plant core melt events, though these events have highlighted the importance of developing such new materials. Potential ATF concepts should be more resistant to oxidation at least for design basis conditions (DBC), exhibiting significantly slower oxidation kinetics compared to the Zr-based alloys used in typical light water reactors (LWR). This should reduce the hydrogen generation along with the rate and magnitude of energy generation due to oxidation during and after uncovering of the reactor core. This can result in an increased time to introduce accident mitigation measures. Currently, two cladding concepts are under development with a relatively high technical readiness: FeCrAl claddings and Cr-coated Zr-based claddings. Both cladding concepts are topic of investigation in the frame of e.g. OECD QUENCH-ATF and IAEA CRP ATF-TS. In both projects bundle tests are performed or investigated on the one hand from material science point of view and on the other hand as evaluation tests for severe accident codes. In the frame of OECD QUENCH-ATF three experiments with Cr-coated claddings are foreseen, up to now one experiment under extended DBC were performed. In addition, three different bundle test facilities are considered in IAEA ATF-TS: QUENCH-19 with FeCrAl, CODEX with a mixed bundle considering Cr-coated and uncoatedZr-claddings as well as DEGREE tests with Cr-coated claddings. Selected experiments of these test series (pre- and post-test simulations) are calculated with the severe accident codes AC²/ATHLET-CD and ASTEC. Currently, both codes were extended for the prediction of FeCrAl and Cr-coated claddings by enhanced models or explicitly by input. The results of the simulations show that both codes are able to predict the physical behaviour of the experiments and different ATF concept correctly. The results of selected cases will be discussed in the paper in detail. Finally, the models will be evaluated possible model developments will be identified.

KEYWORDS
Accident Tolerant Fuel, core degradation, AC²/ATHLET-CD, ASTEC

THE CODEX-ATF TEST

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ABSTRACT

An integral bundle test was carried out in the framework of the IAEA Testing and Simulation for Advanced Technology and Accident Tolerant Fuels (ATF-TS) project in the CODEX (COre Degradation Experiment) facility. The test section included an electrically heated seven-rod bundle composed of 3 pieces uncoated and 4 pieces Cr coated optZIRLO cladding tubes. The main parameters of the scenario were selected on the basis of pre-test calculations.

Electrical heating with two tungsten heaters in each rod was applied. The heated length was 650 mm. The rods were pressurized during the test in order to reach ballooning and burst in the early phase of the experiment. The rods in the bundle were fixed by two spacer grids made of Zr1%Nb alloy. The bundle was placed into a hexagonal shroud. The shroud material was Zr2.5%Nb alloy, the total length of the shroud was 1000 mm. The bundle was heated by direct current power supply units. The steam generator provided hot steam to the test section. The water injection into the steam generator was performed with precision pump at constant flow rate. For heating up, argon gas was also injected into the steam generator.

In the preparatory phase the facility was heated up to 600 °C in 0.2 g/s steam and 0.2 g/s argon flow rates using both external heaters and fuel rod heaters. The heat-up phase continued with the same flow rates and with 1000 W heating power on the rods and 800 W power of external heaters. The rods were pressurised and cladding burst took place at ≈900 °C on most of the rods. The temperature increase was very smooth. At the initiation of water quench, the cladding temperature in the top of the bundle was above 1600 °C. In the upper part of the fuel rods 1400 °C was reached. It is expected that intense Zr-Cr eutectic formation took place at these temperatures. During the quench phase, room temperature water was injected to the bottom of the test section. The total hydrogen production was about 3 g, which indicated significant oxidation of the Zr components.

KEYWORDS
Accident tolerant fuel, Zr-Cr interactions, station blackout event, ballooning, burst

COMPARISON BETWEEN EDF MAAP5.04 AND ASTECV3 CODES ON A HYPOTHETICAL SEVERE ACCIDENT ON THE ELSMOR PROJECT E-SMR DESIGN.

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ABSTRACT

This paper presents a comparison between EDF MAAP 5.04 and ASTECv3 codes for a hypothetical Severe Accident leading to core degradation on the ELSMOR project proposed E-SMR (Small Modular Reactor) Design. The ELSMOR (towards European Licensing of Small Modular Reactors) project is a Horizon 2020 Euratom project. The consortium includes 15 partners from 8 European countries, involving research institutes, major European nuclear companies and technical support organizations. The 3.5-year project, launched in September 2019, investigates selected safety features of Light-Water (LW) SMRs with focus on licensing aspects. The Modular Accident Analysis Program (MAAP) is a deterministic code owned and licensed by Electric Power Research Institute (EPRI) that can simulate the response of light water moderated nuclear power plants during accidental transients for Probabilistic Risk Analysis (PRA) applications. It can also simulate severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs). EPRI MAAP5.04 has limited capability to model SMRs: this code has been adapted by EDF (EDF MAAP 5.04) to make it compatible with the simulation of Severe Accidents transients for the ELSMOR project SMR design. ASTEC v3 is a severe accident system code developed by IRSN aiming at source term evaluation. The code is used to evaluate major nuclear accidents for different nuclear installations with a main focus on western light water reactor designs. A wide range of phenomena are represented, in particular thermal-hydraulics and core degradation while modules dealing with fission products behavior and molten corium concrete interaction are not yet used in this study. The comparison performed between EDF MAAP5.04 and ASTECv3 includes the transient evolution from the initiating event (a Station Black Out), the core degradation and hydrogen generation, the corium relocation to the Lower Plenum and the In-Vessel Melt Retention. The physical phenomena in the containment are also compared (steam condensation on the walls…). Potentiality of H_2 combustion related to the specific assumptions of the selected transient is evaluated through plotting flammability diagram and a sensitivity analysis to N_2 injection for inerting the containment is assessed.

KEYWORDS
EDF MAAP5.04, ASTECv3, SMR, Accidental transients

Model Development for the Simulation of Fission Product Release from Molten Pools

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Abstract

The fission product (FP) release into the primary circuit during a postulated reactor accident represents the potential primary radioactive source term for the containment. An adequate simulation of the release is therefore necessary to provide reliable initial conditions for further source term analyses. During the early accident phase, mainly volatile FP are released. With progression of the accident and liquefaction of the fuel, the transport mechanisms in the materials change. The FP move faster in the liquid phases than in the crystal structures of the solid fuel, which also favours the release of medium and low volatile FP. Further, volatile species can be formed by certain low volatile FP in oxidising conditions with significant impact on their release rates.

While established models and databases exist for the representation of the release of volatile FP from a fuel rod geometry and the release is predominantly adequately represented in current code systems, the available data on the release behaviour of less volatile FP from molten core material is very limited. Thus, there is need for improvement in the AC² code package developed by the German Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH regarding the source term contribution of a molten pool in the lower reactor plenum. The existing deficits and the necessary further developments therefore focus in this work on the release of medium and low volatile FP in the late in-vessel accident phase.

For the representation of the FP release from a molten pool in the lower plenum, a new in-house model based on the diffusion equation is being developed, which is intended to be integrated into the AC² model basis. The variables of the multicomponent system are described in an idealised way applying the Antoine equation. In order to take the influence of volatile species in an oxidising atmosphere into account new Antoine parameters are investigated. Convective mass transfer is considered by the calculation of Sherwood-Numbers. Preliminary verification analyses conducted by the simulation of experiments of the Late Phase Phenomena programme show that the release behaviour of relevant FP is plausibly represented by the new model.

This work was funded by the German Federal Ministry for the Environment, Nature Conservation, Nuclear Safety and Consumer Protection under grant number 1501629 on the basis of a decision by the German Bundestag.

The simulations were performed with the code AC², developed by GRS.

Keywords

AC², ATHLET-CD, Fission Product Release, Model Development, Molten Pools

SEVERE ACCIDENT SEQUENCE ANALYSIS OF LOCA FOR APR1400 USING CINEMA COMPUTER CODE

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As an integrated severe accident computer code development in Korea, CINEMA (Code for INtegrated severe accidEnt Management Analysis) has been developing for a severe accident sequence analysis from an initiation event to a containment failure. The basic goal of this code development is to design a severe accident analysis code package by exploiting the existing domestic DBA (Design Basis Analysis) code system for the severe accident analysis. The CINEMA computer code are composed of CSPACE, SACAP (Severe Accident Containment Analysis Package), and SIRIUS (SImulation of Radioactive nuclide Interaction Under Severe accident), which are capable of core melt progression with thermal hydraulic analysis of the RCS (Reactor Coolant System), severe accident analysis of the containment, and fission product analysis, respectively. The CSPACE is the result of merging the COMPASS (COre Meltdown Progression Accident Simulation Software) and SPACE (Safety and Performance Analysis CodE for nuclear power plants) models, which is designed to calculate the severe accident situations of an overall RCS thermal-hydraulic response in SPACE modules and a core damage progression in COMPASS modules. For the purpose of CINEMA verification for a real power plant, severe accident sequences of LOCA (Loss Of Coolant Accident) without SI (Safety Injection) for the APR1400 has been analyzed in this study. This analysis has been performed to estimate the efficiency of the CINEMA computer code and the predictive qualities of its models from an initiating event to a containment performance during severe accidents. Best estimate calculations from the initiating events of the SBLOCA (Small Break LOCA) of 2 inch diameter and LBLOCA (Large Break LOCA) of 9.6 inch without SI (Safety Injection) have been performed by using the CINEMA computer code.

The accident was initiated by producing 2inch of the SBLOCA and 9.6 inch of the LBLOCA equivalent diameter breaks in the cold leg. The reactor and the RCP (Reactor Coolant Pump) s were assumed to be tripped at an accident initiation time. The RCS water inventory rapidly decreased and a boiling started in the core because the safety injection pumps were not actuated. The fuel began to heat up when the core was uncovered. Oxidation of the fuel cladding began when the cladding surface temperature reached 1,000K and produced an oxidation heat. The fuel cladding was failed by a sausage-type ballooning. When the cladding surface temperature reached 1,700K, oxidation of the zircaloy was accelerated as the steam was supplied from the bottom of the reactor vessel. This resulted in a rapid increase of the cladding surface temperature. At about 2,129K of the cladding surface temperature, the zircaloy inside the oxide shell began to liquefy and the outer portion of the fuel pellets was dissolved. The relatively thin ZrO_2 shell ruptured at about 2,390K because the shell strength decreased with the temperature increase. The bottom of the core dried out because a hot mixture of liquefied fuel and cladding had relocated downward. The debris formed at the bottom of the fuel rods, where the liquefied mixture had resolidified. The melting temperature of the zirconium dioxide is 2,390K, and that of the uranium dioxide is 2,400K in these calculations. The flow blockage in the lower part of the core region occurred because of a fuel melting and a cohesive debris formation. The melted core material had relocated to the lower plenum of the reactor vessel. Finally, the reactor vessel was failed by a creep through a melt thermal attack.

COMPARATIVE STUDY OF THE HYDROGEN DISTRIBUTION AMONG DIFFERENT PWR-W LUMPED-PARAMETER AND 3-D CONTAINMENT MODELS WITH GOTHIC 8.3 (QA)

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EXTENDED ABSTRACT

The containment building is a key safety element in a nuclear power plant (NPP), due to its role as the last barrier against potential radioactive material releases in case of a severe accident. As a lesson learned after TMI-II and Fukushima accidents, H_2 may be accumulated in the containment building during an accident, leading to a combustion, which could increase the temperature and pressure locally above containment design limits. For this reason, a deep understanding of the thermal-hydraulic conditions inside the containment is crucial to assess its integrity against accidental scenarios.

In that line, GOTHIC thermal-hydraulic code allows, by means of lumped-parameter (LP) or 3-D Control Volumes (CVs), to simulate a variety of transients. In the framework of research projects such as GO-MERES (CSN) and AMHYCO (European Commission H2020 programme, GA nº 945057), several models of a generic PWR Westinghouse design (PWR-W) containment with GOTHIC have been developed by the ETSII-UPM group. These models have been used as the basis for this study, performed in the framework of the INTERCON3D project (ref. SN-080-2022), led by the UPM and funded by the Spanish regulator (CSN).

The present work analyzes the capabilities of different generic PWR-W LP and 3-D containment models to simulate the distribution of H_2 in the containment. The first step of the ETSII-UPM group methodology, developed throughout the last ten years to build containment models in GOTHIC, involves the creation of a detailed CAD model of the containment from public information and layouts [1]. From less to high complexity, the complex LP models of 11, 17 and 81 CVs were created following the criteria followed in [2]. The nodalization attends to: (1) compartmentalization, (2) simulation of convection loops in the containment dome, and (3) simulation of the plumes from the Steam Generator compartments, typical of a cold or hot leg LBLOCA type accident. For the 3-D PWR-W model, the "Preventive Methodology", developed by the ETSII-UPM group during the last years, was used [3]. In this methodology, the detailed geometry is adapted to a 3-D Cartesian grid using simple shapes (wedges and cubes) to finally block cells and face surfaces in GOTHIC, ensuring the tightness of different areas of the containment. The ETSII-UPM group generated several 3-D models previously, for German PWR design [4], AP1000 design [5] and PWR-W design [1], using previous methodologies. However, the PWR-W model used for the comparison, represented in GOTHIC by a single subdivided CV with a 1x1x1.5 m grid and 70.400 cells, was adapted to the latest methodology [3].

To perform the analysis a Double-Ended Guillotine Break (DEGB) Loss of Coolant Accident (LOCA) input was implemented in GOTHIC, coming from the MELCOR UPM PWR-W model [6]. This scenario, combined with a single failure of the low-pressure injection of the Emergency Core Colling System (ECCS), led to an In-vessel Severe Accident (SA) situation. The containment initial conditions were 49 ºC, 101.35 kPa, 20 % of humidity. Computational time, performed by 4 cores of Intel Core i7-6700, shows an increasing in time with the complexity of models. For LP models, computational time threshold for a 10,000 s of transient is minutes, in particular: 1' 48" for 11 CVs, 15' 28" for 17 CVs and 29'34" for 81 CVs. In contrast, the computational time recorded for the 3-D model is 5111' 28", much longer than for the LP models.

Pressure trends show similarities between models, but the temperature distributions show relevant differences. First, about average containment pressure, 81 CV LP model and the 3-D model slightly deviates, which may be caused by the pressure difference between near rupture cells, where the gas is entering, and cells farther form the rupture. On the other hand, the average containment temperature depends on the released gas temperature and its contact with the Heat Structures (HS). For 11 and 17 CV LP models, the gas is homogenized in large cells and is instantly in contact with HS. However, 81 CV LP and 3-D models contains small cells near to the break and not in contact with HS, causing an increase on the temperature.

Towards a deep analysis of the impact of the nodalization of LP models on hydrogen distribution in a PWR-W containment, and its comparison with the 3-D approach, a visual representation was conducted. This representation of H_2 volume fraction was generated by ParaView and ProTON, a post-processing tool developed by ETSII-UPM group. Through this representation, relevant differences are observed. First, 17 and 81 CV LP models simulate the ascendant movement of H_2 throughout the containment. However, just the 81 CV volume reach similar H_2 volume fractions than 3-D model, due to the configuration of the plumes. Similarities between complex 81 CV LP and the 3-D models verify its use for test cases. Despite the similarities between 81 CV LP model and 3-D model, the finer nodalization and the limitations of the LP approach (non-conservation of fluid momentum and homogenization of properties) cause differences which justify the use of 3-D models. Those differences are relevant to identify H_2 clouds with combustion risk.

The following conclusions have been extracted from the comparison: (1) The increase of complexity in GOTHIC models implies an increase in computational time, making LP models useful for test cases; (2) Pressure trends show similarities between models, but the temperature distributions show relevant differences, caused by the introduction of hot gases into small cells; (3) The visual representation of the H_2 distribution allows to observe the similarities, as well as limitations and benefits of LP and 3-D approaches.

KEYWORDS
Containment Analysis, GOTHIC, Lumped-Parameter, 3-D Models, H_2 Combustion Risk

THERMOPHYSICAL PROPERTY MEASUREMENT OF OXIDE MELTS USING AERODYNAMIC LEVITATION

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ABSTRACT

Following the Fukushima incident, there has been a global focus on the safety of nuclear power plants, emphasizing the need for comprehensive research on severe accident phenomenology. A key aspect of understanding corium behavior during severe accidents is the elucidation of the thermophysical properties associated with corium components. This study is specifically dedicated to the measurement of density, surface tension, and viscosity of oxide melts by aerodynamic levitation. Aerodynamic levitation is an effective method for conducting thermophysical property measurements at elevated temperatures. This technique involves stably levitating small sample spheres through a conical nozzle, thereby eliminating interactions between the sample and crucible. By integrating laser heating technology, samples can easily reach temperatures of up to 3000K. The oscillating drop method is used to determine the resonant frequency and damping constant. Density is calculated as the ratio of mass to volume, while surface tension and viscosity are derived from the resonant frequency and damping constant using the Rayleigh and Lamb equations, respectively. The present study contributes valuable thermophysical property measurements for alumina and zirconia, demonstrating the effectiveness of the approach for the measurement of corium components.

KEWORDS
Thermophysical property, aerodynamic levitation, alumina, zirconia

THERMODYNAMIC EVALUATION OF LIQUID-GAS SURFACE TENSION FOR $U-O-ZR$ MIXTURES USING THE BUTLER EQUATION

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ABSTRACT

In this work, the Butler equation is assessed for evaluating the liquid/gas surface tension for three compositions in the ternary $U-Zr-O$ system over a wide temperature range 2600−3100 K. In this approach, the liquid phase Gibbs energy is obtained from two CALPHAD-type databases, NUCLEA and TAF-ID. Two formulations of the Butler equation make use of an ideal phase mixing hypothesis (φ being the weighting factor) with and without surface tension laws for pure components. A third formulation corresponds to the classical form of the Butler equation found in the literature where the interfacial Gibbs excess energy is assumed to be proportional to the liquid one (the multiplicative factor k_Γ being related to the ratio of broken bounds). Computed surface tensions are compared to experimental values from the literature. Three different compositions corresponding to different Zirconium oxydation degree, C0, C30 and C50 are considered. It is shown through a limited parametric analysis that the choice of the CALPHAD database affects both the slope w.r.t temperature and the level of the interfacial energy while parameters φ and k_Γ mainly affects the level and only slightly the slope. Considering pure component surface tension data has a significant impact on both and, considering the C0 composition measurements, it is shown that parameters φ and k_Γ can be successfully calibrated for reproducing these data. However, the steep variation of the interfacial energy w.r.t temperature exhibited by experimental results for the C30 composition is, in any case, not correctly captured. This shows the need for further improvement of the liquid thermodynamic modelling and a path forward is proposed.

KEYWORDS
Surface tension, corium, Butler equation, CALPHAD

ANALYSIS OF COMBUSTIBLE GASES DISTRIBUTION WITH ACCIDENT MANAGEMENT ACTION IN A GENERIC PWR-W CONTAINMENT

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EXTENDED ABSTRACT

In a Severe Accident (SA) in a Light Water Reactor (LWR), hydrogen can be mainly generated by oxidation reactions inside the Reactor Pressure Vessel (RPV) or by the Molten Core-Concrete Interaction (MCCI) in the cavity pit, where, in addition, carbon monoxide can be produced. Both gases are released towards the containment and, depending on containment thermal-hydraulics, they can accumulate and eventually reach flammable concentrations. Aimed to improve severe accident management concerning the minimization of the in-containment combustion risk, the AMHYCO project has been launched in the frame of the Euratom program.

In the present work, within AMHYCO, a generic Western PWR containment is modelled with MELCOR v2.2 to analyze the effect of the safety systems and Passive Autocatalytic Recombiners (PARs) in the evolution of combustible gases concentrations with a special focus on the ex-vessel phase of the accident. Two scenarios have been evaluated. The first one is a Large Break LOCA (LBLOCA), representing a double-ended guillotine break of a hot leg pipe. The containment sprays are available in injection mode until the Refueling Water Storage Tank (RWST) is depleted. Then, the spray system switches to recirculation mode. The second scenario is a Station Black-Out (SBO) with late recovery of sprays at different times into the accident.

Only the containment has been modelled following the specifications discussed and defined within the project. This containment (free volume of 61,000 m^3) gets divided into 11 control volumes connected by 32 flow paths. Forty Passive Autocatalytic Recombiners (PARs) of two types (FRAMATOME FR-1500 and FR-960) are considered within the containment. Correlations for H_2 and CO recombination rates have been specifically developed within the project and are implemented through so-called control functions in the MELCOR code. The release of fluids into the containment is modelled as boundary conditions (i.e., external sources) taken from full plant simulations. The LBLOCA is characterized by a fast release of water and steam into the steam generator chamber from both sides of the guillotine break followed by the hydrogen generated in the vessel. After RPV failure (~3 h), H_2, CO, steam and CO_2 are released by the MCCI directly into the cavity atmosphere. The SBO has a delayed release of gases into the annulus compartment from the pressurizer relive tank. The RPV failure, and thus, the beginning of the MCCI, is predicted at ~11 h. Additionally, a heat source is defined into the cavity to model the corium-to-cavity atmosphere heat transfer of both scenarios after the failure of the RPV.

For the LBLOCA, PARs significantly reduce the molar fraction of combustible gases (H_2 and CO) and avoid conditions of uncontrolled combustion in the in-vessel phase but not for some hours during the ex-vessel phase (Fig. 1), mainly due to the large amount of CO generated by the MCCI combined with the lower PARs efficiency to remove CO compared to H_2. Another parametric case has been studied in which the cooling of the containment is provided by fan coolers and sprays do not have a heat exchanger in the recirculation

mode. In this case, the pressure is 0.3 bar higher in the ex-vessel phase since fan coolers have a milder heat removal capacity than the spray system. The impact on the combustible gases concentration is however low, and therefore, flammability conditions can also be reached in the ex-vessel phase.

For the SBO, PARs are able to prevent the mixture from reaching flammable conditions along the whole sequence, even if a PAR efficiency reduction of 50% is postulated. PARs have a moderate effect on increasing the pressure and temperature. Regarding spray recovery, as expected, a strong effect in reducing the steam molar fraction, and therefore in the pressure, is observed. In all cases, PARs prevent uncontrolled combustion in the dome, but in the annulus (the release compartment), the mixture reaches the flammability limits for short periods in the in-vessel phase if sprays are activated at the earliest times considered, 4.5 hours (Fig. 2).

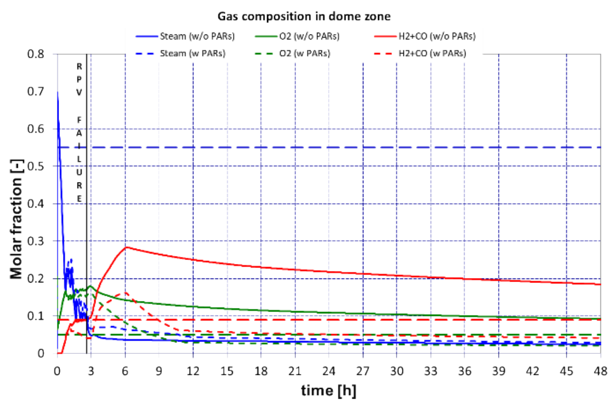

Fig. 1. Gas molar fraction in the dome for the LBLOCA. Comparison of simulations with PARs and without PARs

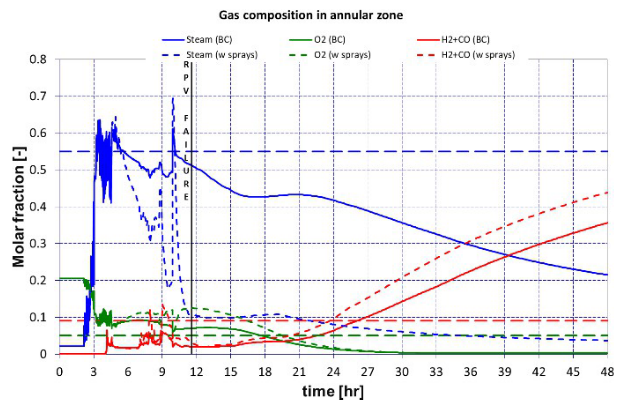

Fig. 2. Gas molar fraction in the annulus for the SBO. Comparison of the base case simulation and with sprays recovered at 4.5 h.

The two scenarios analysed have very different conditions in pressure and steam concentration mainly due to the operation of the sprays in the LBLOCA unlike the SBO scenario. In both sequences, PARs strongly reduce the risk of combustion by decreasing the concentration of combustible gases and depleting O_2 in the containment atmosphere. Nonetheless, attention has to be paid to the early activation of sprays since, if the PARs do not have time to sufficiently remove the combustible gases or the O_2, the sprays can greatly increase the risk of flammability. In addition, the CO generation in the ex-vessel phase can represent a significant risk if the steam concentration is low, even if PARs are used as a passive mitigation measure.

The AMHYCO project, coordinated by the UPM (Universidad Politécnica de Madrid) has received funding from the Euratom research and training programme 2019-2020 under Grant Agreement n°945057. The content of this paper reflects only the authors' view. The European Commission is not responsible for any use that may be made of the information it contains.

Development of 3D view application debrisEye for decommissioning of Fukushima Daiichi Nuclear Power Plant

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ABSTRACT

Internal investigations of the Fukushima Daiichi Nuclear Power Plant (1F) have been conducted, and the internal situation is gradually becoming clearer. In addition, trial debris removal has been conducted and much information is being obtained. The information obtained from the trial debris removal is managed in the decommissioning fundamental research database (debrisWiki), which was established by JAEA and TEPCO. However, it is difficult to understand the entire accident progress only from individual data. Therefore, we developed a 3D view application (debrisEye) for 1F decommissioning. debrisEye was created by Unity. For the CG displayed in debrisEye, pre- and post-accident conditions were constructed. The pre-

accident status was created using design information and point cloud data from periodic inspections. The post-accident status was created mainly from the results of the internal investigation. For areas where internal investigations have not yet been obtained, the information in the estimation diagram was reflected. CG displayed on debrisEye can be viewed from any viewpoint and angle using the functionality contained in debrisEye. It is also possible to clipping at any cross section and to show or hide each part. debrisEye can be linked to and used with debrisWiki to write information in any location, thus displaying the analysis results and location of the debris collected. Visual linking of debris analysis results with on-site information is expected to facilitate understanding of accident progress and improve efficiency of decommissioning work.

KEYWORDS
Fukushima Daiichi, Severe accident, Decommissioning, debrisEye

Thermal shock resistant geopolymers as refractory material for core catcher

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ABSTRACT

Geopolymers have been intensively studied for immobilization of radioactive waste, but they can be beneficial in other applications. These materials show strong resistance to radiation, low moisture content and possibility of incorporation of various additives to modify their properties. This universality makes them a potential sacrificial, structural, or refractory component in passive or active safety systems in nuclear power plants (NPPs). [1] This may enhance the safety of NPPs and help to mitigate the severe accident consequences.

This paper presents a scoping interaction test of a geopolymer refractory material with prototypic corium melt. The corium mixture was achieved by induction skull melting technology and poured on a geopolymer refractory plate with high MgO content. The results suggest a good resistance of geopolymers to corium impact and minimal interaction observed through post-test analysis.
The good thermal shock resistance and low thermal conductivity make this material interesting for application in first contact layers, refractory sacrificial materials etc.

KEYWORDS
corium, severe accident, geopolymer, refractory ceramics, core catcher

ACKNOWLEDGMENTS

The presented results have been financially supported by the Technology Agency of the Czech Republic (TA CR) - the project TK01030130, and they were obtained using the CICRR infrastructure, which is financially supported by the Ministry of Education, Youth and Sports - project LM2023041.

INVESTIGATION OF NEW INORGANIC MATERIALS FOR NUCLEAR INDUSTRY UNDER SEVERE ACCIDENT CONDITIONS

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ABSTRACT

Geopolymers can show superior properties than concrete in nuclear reactor severe accident applications, helping to increase safety and mitigate consequences. Scientists study geopolymers primarily for usage in radioactive waste immobilization due to their polymeric structures and advantageous properties. With these properties, such as high gamma radiation resistance, low water content, and the ability to incorporate various elements to enhance physicochemical properties, geopolymers may be sacrificial materials in GEN IV reactors [1]. In present study we conducted a geopolymer interaction experiments with molten corium and studied the physicochemical properties at extreme temperatures to contribute to their qualification and application. Based on our research, geopolymers with Gd_2O_3 additive as a neutron absorber and Fe_2O_3 additive as a liquidus lowering agent of the mixture indicated as functional sacrificial materials for GEN IV reactors, exhibiting advantageous physicochemical properties for use as sacrificial material and commonly used concrete.

KEYWORDS
Geopolymer, corium, sacrificial material, severe accident

ACKNOWLEDGMENTS

The presented results have been financially supported by the Technology Agency of the Czech Republic (TA CR) - the project TK01030130, and they were obtained using the CICRR infrastructure, which is financially supported by the Ministry of Education, Youth and Sports - project LM2023041.

Modeling of pool scrubbing and sensitivity analysis using GOTHIC

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Abstract

Pool scrubbing referred as the process of the retention of fission products in a water pool, is a mitigation measure to avoid leaking of radioactive fission products into the environment during severe accidents. Aerosol particles accompanied by gaseous mixtures (e.g., steam and non-condensable gas) can be injected through multi-hole spargers or vents into a large water pool during different scenarios. The amount of aerosols captured by the pool is governed by the transport processes of aerosols inside the bubbles and the dynamic characteristics of the gas-liquid interface. Decontainment factor (DF) defined by the ratio of injected aerosols over escaped aerosols is used to evaluate the efficiency of the pool scrubbing. Several codes (e.g., BUSCA, SUPRA, SPARC-90, SPARC-B/98) have been developed to evaluate the DF during pool scrubbing. In the latest update of the containment system code GOTHIC (v8.4), the pool scrubbing feature derived from SPARC-90 is enabled. The motivation of goal of this work is to investigate the modeling capability of GOTHIC with respect to pool scrubbing and develop and implement GOTHIC models to provide analytical support of experiments conducted in THAI facility.

Keywords

Pool scrubbing, aerosol deposition, THAI, GOTHIC,

MELCOR analyses of Severe Accident sequences in an integral PWR with passive systems

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Abstract

The SASPAM-SA Horizon Euratom project is investigating the maximization of the knowledge transferability and impacts of the project two generic design concepts, characterized by different evolutionary innovations in comparison with larger operating reactors, with a focus on the ability of Severe Accident (SA) programs to mimic integral Pressurized Water Reactors (iPWRs) during SA sequences. The effectiveness of the MELCOR system code in simulating severe accident scenarios that might occur in an iPWR is the main topic of this study.

The selected design, referred to as Design 2 in the SASPAM-SA project, is a 300 MWe iPWR characterized by passive safety systems and a dry spherical metallic containment. As a result, Design 2 serves as a general-purpose reactor design that may be representative of relevant reactor concepts like IRIS and SMART. In this framework, a MELCOR input deck was prepared by Sapienza University of Rome in collaboration with ENEA and CIEMAT using MELCOR 2.2 18019, developed by Sandia National Laboratories (SNL) for the USNRC, with the aim to assess DBA and BDBA code prediction capability.

The selected initiating event analyzed in this study is a small break Loss of Coolant Accident (SBLOCA) due to a double guillotine rupture of the Direct Vessel Injection line. This could be considered the only plausible LOCA in the primary circuit since integral reactor layouts intrinsically avoid Large break LOCAs because the reactor's main components, such as the Steam Generator and the pressurizer, are inside the Reactor Pressure Vessel (RPV). The developing SA scenario involves the failure of every passive safety system installed: the Emergency Heat Removal System, the Emergency Borated Tanks, the Long-term gravity makeup systems and the Automatic Depressurization Systems. This extremely unlikely scenario was selected as the worst case only to test the code prediction capability.

Transient analysis results highlighted that the MELCOR code captures the predominant phenomena driving the transient evolution of accidental scenarios, starting from a thermal-hydraulic behavior and including the core degradation, and material relocation in the RPV lower head. This preliminary study serves as a base for future developments in SASPAM-SA WP4, which will focus on more detailed simulations to predict the key phenomena, associated uncertainties, and mitigation strategies during In-Vessel-Melt-Retention in such PWR.

Keywords
SMR, Passive systems, iPWR, SASPAM-SA

Development of severe accident simulation code for sodium-cooled fast reactors: SIMMER-V (2) Development and verification of detailed fuel pin model

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ABSTRACT

The new detailed fuel pin model has been developed as a part of the SIMMER-V code to simulate thermal and mechanical behavior of the fuel pin from accident initiation to fuel pin failure. The SIMMER-III and SIMMER-IV codes have been developed mainly to simulate the event progression in transition phase (TP), and the initiating phase (IP) was simulated by the SAS4A code. Calculated results of IP are taken as the initial conditions of a TP analysis. In this data transfer from SAS4A to SIMMER, the differences in computational geometries and models can cause discontinuity in calculated event progression. In particular, the standard fuel-pin model of SIMMER-III and SIMMER-IV is too simple to represent the complex behavior necessary for simulating the IP. To solve these problems, an improved and detailed fuel-pin model has been developed at JAEA as a part of a new SIMMER-V code, a joint research program being conducted in collaboration with CEA. To develop the new model, the key physical phenomena relevant to fuel pin failure were extracted. They include: thermal behavior, fission gas release, fission gas swelling, mechanical deformation, and in-pin fuel motion. Each of these phenomena was modeled as a module, which was then incorporated to SIMMER-V. Considering the applicability of the model to wider range of core and fuel designs, it is necessary to simulate in detail the fuel motion in a central hole of the annular fuel pellet and the fuel ejection from the fuel pin cavity to the coolant channel. In order to simulate these phenomena and obtain numerical stability, the in-pin fuel motion was modeled based on the multi-phase fluid-dynamics algorithm of SIMMER-III. In addition, the model enables the multi-cavity formation for an axially heterogeneous core design. A verification matrix was constructed to check the performance of the new model, and test cases were prepared, focusing on verifying individual modules. Since the new detailed fuel-pin model has been developed to simulate the physical phenomena occurring in the IP, the calculated results are compared with SAS4A, to check the capability of the model. This study verified the performance of individual modules of the new detailed fuel-pin model of SIMMER-V.

KEYWORDS
SFR, SIMMER-V code, detailed fuel pin model, verification

Analysis of the combustion risk mitigation inside the containment during a postulated severe accident in a PWR using the code package AC²

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ABSTRACT

In the framework of the European research project "AMHYCO" the combustion risk inside the containment is investigated. Therefore, different severe accident scenarios are simulated and analyzed. A new nodalization for a generic containment of a pressurized water reactor (PWR) type KWU has been created within the project. Furthermore, different combustion mitigation measurements are postulated and the influence on the composition of the containment atmosphere is investigated.

A loss of coolant accident (LOCA) with a postulated failure of almost all active safety systems is conducted. During the in-vessel phase of the accident around 600 kg hydrogen are produced and released to the containment. During the ex-vessel phase of the accident additional 800 kg hydrogen and 7,000 kg carbon monoxide are released by molten corium concrete interaction. Both, the rising pressure inside the containment due to the steam release as well as the potential combustion risk due to the presence of combustible gases, may jeopardize the containment integrity.

The PWR KWU containment is divided into 12 zones, of which the dome zone with 42,654 m^3 is the largest zone. The break during the LOCA accident is postulated to occur on the cold leg at the pressurizer loop in the north-sided steam generator room. The compartments are connected with each other either via doors, flaps or open connections to enable both gas and liquid transportation. To reduce the hydrogen concentration inside the containment, 40 generic box-type passive autocatalytic recombiners (PARs) are installed all over the compartments. To investigate the PAR efficiency several simulations with different PAR performance rates (0 %, 50 %, 75 %, 100 %) are conducted. The varying PAR performance rate is intended to simulate the case that recombiners are not fully available, for example due to poisoning or maintenance. The PARs catalytically recombine H_2 and CO in presence of oxygen. In the framework of the AMHYCO project, a new generic correlation which coFconcentransiders low-oxygen and CO-rich atmospheres is developed and used for the simulations. The PARs work passively, which leads to an immediate recombination after the onset of H_2 and CO from the in-vessel circuit to the containment.

The results show, that even with only 50 % PAR performance the H_2 and CO volumetric concentration in the dome compartment decrease from 18.9 % respectively 6.7 % to 14 % respectively 6.1 % at the highest peak. With increasing PAR performance up to 75 % and 100 % the H_2 concentration decreases further to 12 % and 10 %. The CO concentration shows a similar decrease to 5.5 % (75 % PAR performance) and 5.2 % (100 %). (Figure 1 and 2)

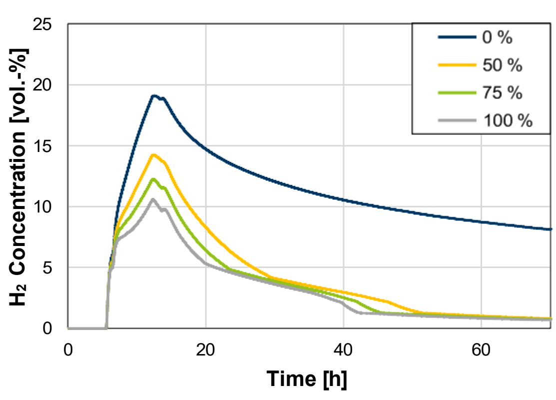

Figure 1: H_2 Concentration in Dome

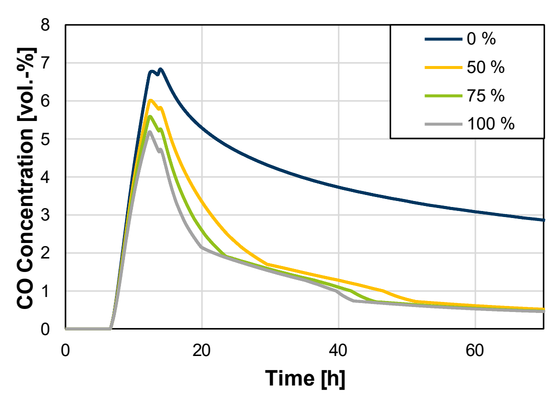

Figure 2: CO Concentration in Dome

As a result of the recombination, the total mass of the substances in the dome, due to its largeness used as the reference zone, decreases significantly. At the peak point, right after the melt in the cavity is flooded, the H_2 mass is halved from 1,000 kg without recombiners to 480-680 kg, depending on the performance. At the same time the present CO mass is lowered from 1,100 to 1,850 kg of CO. With ongoing accident progression, the burnable gas masses further decrease significantly to a minimum of 62 kg (H_2) and 550 kg (CO). (Figure 3 and 4)

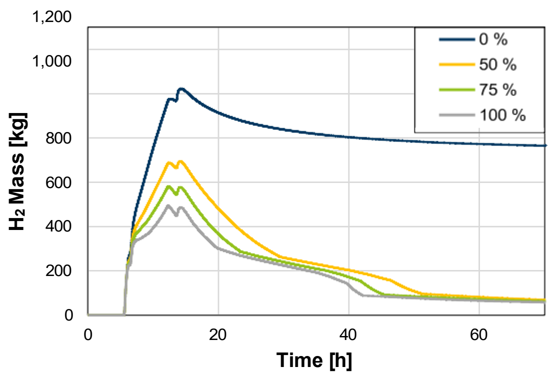

Figure 3: H_2 Mass in Dome

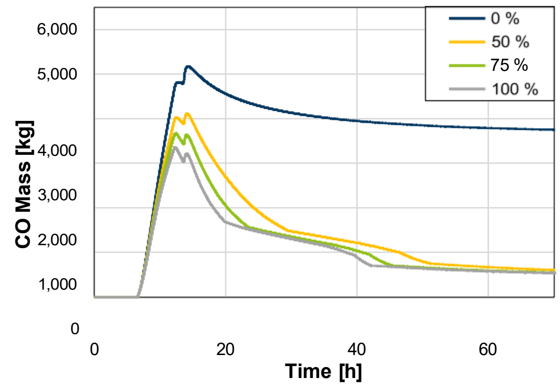

Figure 4: CO Mass in Dome

A comparison of the Shapiro diagrams (Figure 5 a-d) clearly shows that the flammability risk is strongly reduced. While the gas mixture in the dome during the unmitigated scenario remains in the detonation region as well as the burning region for a long period of time, the flammability in the mitigated scenarios is highly reduced. While the detonation region is still touched when only 50 % of the available PAR capacity is used, the 75 % and 100 % variants have a very similar progression. Here, the gas mixture is only in the comb latively short period of

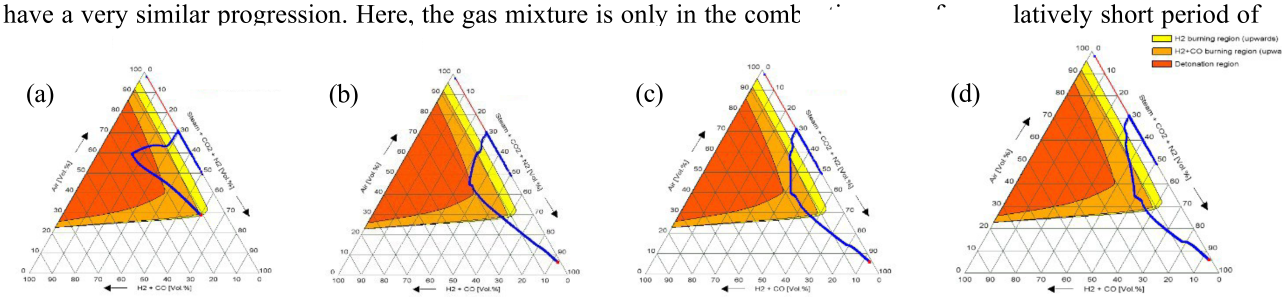

Figure 5 a-d: Shapiro Diagrams for 0 % (a), 50 % (b), 75 % (c) and 100 % (d) PAR Performance

The simulations are performed using the code package AC² 2021.1 containing the accident analysis codes COCOSYS developed by GRS.

KEYWORDS: *AC², Plant Simulation, Combustion Risk, Hydrogen, Passive Autocatalytic Recombiners*

Acknowledgement:

AMHYCO has received funding from the Euratom research and training programme 2019-2020 under Grant Agreement n°945057. The content of this paper reflects only the author's view. The European Commission is not responsible for any use that may be made of the information it contains.

Uncertainty quantification analysis with radiological consequences for a loss of cooling accident in a spent fuel pool

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Abstract

In the current field of analyzing severe accidents in nuclear reactors, incorporating a robust methodology for quantifying uncertainties has become essential. The intricacies of the physical phenomena occurring during a severe accident give importance to acknowledging and addressing the inherent unpredictability and variations that can influence outcomes.

The presented uncertainty quantification analysis has contributed to the Management and Uncertainties of Severe Accidents (MUSA) project, where a dedicated work package was devoted to quantify uncertainties affecting accident analyses in a Spent Fuel Pool. This effort also involved the evaluation of the potential advantages that can arise from quantifying these uncertainties in reducing radiological consequences.

A detailed modeling of a boiling water reactor spent fuel pool has been developed using the MELCOR code. Radionuclide classes have been duplicated to model decay heat distribution between hot and cold fuel assemblies. The effects on the accident progression of physical phenomena, such as core degradation, heat transfer, and radionuclide transport, are investigated by selecting 29 uncertain input parameters. Using the RAVEN tool, a random Montecarlo sampling strategy has been adopted to propagate uncertainties all over the input space. The source term and the selected uncertainty parameters are examined in a post-processing phase with a sensitivity study to evaluate and rank possible correlations.

The accident sequence designated for the analysis is a loss of cooling accident. All emergency and mitigation systems are unavailable, resulting in the degradation of fuel racks containing fresh, hot, and cold fuel and irradiated core internals. The MACCS code software, developed to assess the repercussions of severe accidents on neighboring environments, has been used for dispersion modeling and dose rate computations.

The study's primary findings on uncertainty quantification, combined with other MUSA project outcomes, can significantly enhance the assessment's reliability. This is achieved through a comprehensive grasp of potential risks and vulnerabilities, empowering decision-makers with informed insights for proactive risk mitigation and management strategies.

Keywords
SFP, Uncertainty Quantification, Sensitivity, Radiological Consequences, RAVEN

THS-15 Experimental facility: Effect of Surface roughness on CHF values

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ABSTRACT

The THS-15 (Thermal-Hydraulic Stand, est. design 2015) test facility was built in ÚJV Řež in 2018 to perform thermo-hydraulic experiments to verify the cooling capacity of the VVER-1000 reactor during a severe accident using the In Vessel Melt Retention (IVMR) strategy. As part of the international IVMR project under H2020 (IVMR - Grant Agreement 662157), a series of experiments have been performed on the THS-15 facility. In the first phase of the experiments, tests were performed to confirm that the reactor pressure vessel can be cooled from the outside under the conditions expected in the final phase of a severe accident (e.g., no boiling crisis, no pressure above the limit, etc.). These experimental conditions were initially determined thanks to analytical support and the results of severe accident calculations. The experiments of the first phase were successful in terms of the functionality of the IVMR strategy (for VVER-1000), and therefore the second phase of the tests was started, in which the critical heat flux (CHF) was studied according to specific conservative conditions. When comparing the CHF and the heat flux distribution around the RPV, it was possible to establish the so-called "safety margin to the CHF."

As the CHF values during the IVMR depend on many factors, possible improvements were considered for the third phase of the tests in order to increase the CHF values (through technical measures). The first measure assessed was to perform experiments with a deflector (a structure along the RPV that allows the intensification of the coolant flow). The results of the CHF values obtained during the experiments with the deflector were on average 20% higher than in the tests with the configuration without deflector.

As part of the following national project (named "Increasing the Safety Margin of IVMR Strategy Applications for VVER 1000", number TITSSUJB830) - the influence of another factor on the CHF values was investigated, namely the effect of different roughness of the RPV surface. The CHF values measured during the experiments in the original device configuration and then on the device where the RPV surface roughness was increased were compared.

KEYWORDS
Critical Heat Flux (CHF), pool boiling, In Vessel Melt Retention (IVMR), External Reactor Vessel Cooling (ERVC)

Cooperative Nuclear Safety Research Activities at the Nuclear Energy Agency in Response to the Fukushima-Daiichi Accident

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Since the 2011 Tōhoku earthquake and tsunami which hit the Fukushima Daiichi Nuclear Power Station (FDNPS), the Nuclear Energy Agency (NEA) is conducting nuclear safety research activities involving wide international collaborations to gain insights for a safe decommissioning of the FDNPS and for safety enhancements of NPPs in general. These activities are conducted under the auspices of the WGAMA (Working Group of Accident Management and Analysis) of the Committee of the Safety of Nuclear Installations (CSNI) and in joint nuclear safety research projects, with some dedicated directly to collecting insights from on-site investigations, to the analysis of the accident and to the preparation of the damaged plant decommissioning such as the BSAF (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant) Project, the PreADES (Preparatory Study on Analysis of Fuel Debris) project, the ARC-F (Analysis of Information from Reactor Building and Containment Vessel and Water Sampling in Fukushima Daiichi NPS) project and the TCOFF (Thermodynamic Characterisation of Fuel Debris and Fission Products Based on Scenario Analysis of Severe Accident Progression at Fukushima-Daiichi Nuclear Power Station) project. A number of other joint nuclear safety research projects are also investigating safety issues related to the FDNPs accident. Insights learnt from these research activities have been considered worldwide to improve nuclear safety.

As the most recent activities, the FACE (Fukushima Daiichi Nuclear Power Station Accident Information Collection and Evaluation) project has begun in July 2022, and aims to address key technical issues identified in recent investigations into the damaged plant, to improve understanding of the accident and to support the decommissioning activities.

There are still immense challenges ahead related to the FDNPs long term management. International collaboration should be continued, first through the on-going FACE project, to further inform decommissioning and to value more the insights from the accident analyses for severe accident management both for operating and future reactors. In order to provide directions for future research in the accident management area, NEA is organizing in September 2024 an international workshop on future research to enhance accident management in operating and future reactors, informed by Fukushima Daiichi insights.

FILTRATION EFFICIENCY OF ELECTROSTATIC PRECIPITATOR FOR IODINE PARTICLES IN DIFFERENT GAS ATMOSPHERES SIMULATING SEVERE ACCIDENT SCENARIOS

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ABSTRACT

In pursuit of carbon-neutral energy sources, the relevance of nuclear power plants (NPPs) has been increasing, requiring improved NPP safety. In the event of severe accidents (SA), the release of radioactive aerosols and the potential for hydrogen (H_2) explosions pose significant challenges. This study aims to enhance NPPs' safety by addressing the challenges of radioactive aerosol emissions and H_2 during SA.
The preliminary experiment investigating the secure removal of H_2 during the venting scenarios was conducted with the Electrostatic Precipitator (ESP) in the FunktioMat laboratory (https://sites.uef.fi/fine/front-page/funktiomat/). ESP was used with Formier (5%) and nitrogen gas at room temperature. The residual gas was collected from the ESP outlet and analysed with the microGC (Agilent 490) for traces of H_2. Though the final concentration of H_2 was reduced (~ 25%), in an inert atmosphere, the corona discharge in the ESP alone was unable to fully oxidise H_2. Extensive experiments with detailed atmospheric conditions and varying settings with different ESPs are needed to conclude our findings.
The study further aims to examine the effectiveness of ESP on the filtration of radioactive aerosols emitted during hypothetical SA. A particular focus will be applied to the chemistry and behaviour of iodine species. Gaseous iodine and iodine oxide particles are passed through an ozone feed, oxidising these particles, and filtered through the ESP. ESP-based filtration employs a humidified inlet for aerosol flow, enabling particle growth and enhanced particle charging for improved collection in the electric field.

KEYWORDS
Electrostatic precipitators, Nuclear power plants, severe accidents, hydrogen mitigation

Study on AP1000 accident diagnosis and treatment for loss of monitoring and control

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Abstract
The Fukushima nuclear accident showed that extreme external events may cause the loss of monitoting and control of control rooms (including main control room and remote shutdown station), because the previous emergency operating procedures (EOP) and severe accident management guidelines (SAMG) of nuclear power plants did not consider the loss of control room monitoring and control scenario, in order to improve the accident management system of nuclear power plants, it is necessary to study the accident diagnosis and treatment method for conditions where the control rooms lose monitoring and control. The SSG-54 (Accident Management Programs for Nuclear Power Plants) released by IAEA in 2019 clearly states that the scenario of instrument failure and loss of command and control should be considered in the management of severe accidents at nuclear power plants. NRC in 2019 issued the requirement to mitigate beyond-design-basis external events (MBDBE) in regulation 10CFR50.155, which proposes that nuclear power plants should set up mitigation strategies for such scenarios caused by extreme external events. In order to ensure the AP1000 nuclear power unit's ability to respond to loss of control room monitoring and control, this article carries out the following analysis: (1) reactor and spent fuel pool states monitoring method for loss of monitoring and control scenario by analyzing AP1000 instrumentation and control system from 0^{th} layer to the 2^{nd} layer; (2) prevention and mitigation strategies for AP1000 nuclear power units to deal with severe accidents based on accident mitigation objectives and AP1000 design features; (3) Study local operation methods for mitigation strategies when the control room is unavailable; (4) determine the diagnosis and treatment logic of loss of monitoring and control conditions. By studying the accident management methods of AP1000 NPPs for loss of monitoring and control condition, the ability of nuclear power plants to respond to extreme external events will be further improved, and the depth defense system and safe operation level of nuclear power plants will be enhanced.

Keywords
AP1000, nuclear power units, accident management, loss of monitoring and control

Failure modes of the reactor coolant pressure boundary in high-pressure core melt accident scenarios

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Abstract

In light water reactor core melt accident scenarios with an unsuccessful pressure relief, the persisting mechanical loads in the reactor coolant pressure boundary become superimposed by increasing thermal loads from hot gas flows generated in the core melt process. Location, time and size of the failure and subsequent depressurization can have substantial implications on the further course and consequences of the accident. The failure location determines not only the location of release of hydrogen and radionuclides, but also the risk of bypassing the containment as a last barrier for radioactivity. Recent investigations of the events at Fukushima Daiichi emphasize the importance of this issue and its cliff-edge nature.

The modeling of severe accident phenomena has been improved over the years by further development of lumped parameter system codes as well as by the use of mechanistic codes, such as computational fluid dynamics. The modeling of mechanical failure has been concentrated on specific components such as the reactor pressure vessel. Failure modes of other components, which may fail much earlier in the described scenario, are often not considered, or their assessment is based on expert judgment or questionable relations borrowed from other fields of application. This is not least because of the complexity of the reactor coolant pressure boundary, which includes a variety of components with several failure modes that in addition differ considerably from those at operational conditions. Furthermore, as failure times of different components are assumed very close, modeling effort must be equally distributed and failure modes should not be overlooked. For each failure mode, the knowledge of the exact phenomenology and connected uncertainties must be put in front of a proper quantitative modelling.

The work provides an overview of identified failure modes and their associated particularities. Where appropriate, deduced approaches for a practical failure assessment are briefly presented. A first group includes failure modes resulting from mechanical interactions between the system's components or with adjacent structures, e. g. by thermal expansion. A second group encompasses failure modes of single components, such as large piping, elbows, junctions, steam generator tubes, nozzles, penetrations, pre-damaged components, pumps, flanges, and valves. In a third group, failure modes at material level are described, which includes failure modes based on macroscopic deformation, such as buckling, kinking, plastic instability, or creep necking, as well as failure modes based on loss of internal cohesion and the associated characteristic types of microstructural damage.

The paper is based on results from the research projects RS1520, RS1555 and RS1602 of the reactor safety research program sponsored by the Federal Ministry for the Environment, Nature Conservation, Nuclear Safety and Consumer Protection.

Keywords
high-pressure core melt, HPME, C-SGTR, reactor coolant pressure boundary, failure modes

Parametric sensitivity studies for RELAP/SCDAPSIM model of QUECH-20 test

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ABSTRACT

The QUENCH-20 test was the first experiment equipped with a BWR fuel bundle conducted at the QUENCH facility in Karlsruhe Institute of Technology in 2019. The objective of the test was to understand the effect of B_4C and its stainless-steel cladding on BWR fuel assembly degradation during strong heat-up and subsequent flooding processes. One of the main aspects of this test is hydrogen generation and overall degradation of the BWR fuel bundle at the mentioned conditions.

For the numerical investigation of the QUENCH-20 test, the severe accident code RELAP/SCDAPSIM was used. The developed RELAP/SCDAPSIM model which uses the "PWR control rod" component to simulate absorber blades and the "Shroud" component for modeling the structural elements of the fuel bundle (fuel channel box, water channel box, and water wings) gives close calculation results to the experimental data. The calculated cladding and shroud temperatures showed ~5 % lower values (~50 degrees in the pre-oxidation phase and ~100 degrees in the quench phase) compared to experimental data.

Calculation results of hydrogen generation are less by ~10 % (~5 g) compared to measurements taken at the end of the QUENCH-20 test. Thus, it could be concluded that the results of the calculations show that the thermal behaviour is qualitatively and quantitatively predicted in good agreement with the experiment, but some discrepancies need to investigate in more detail. For this reason, it was decided to perform the parametric sensitivity studies. These studies investigate the influence of parameters related to initial, boundary conditions, and radiation heat transfer in the SCDAP modeling part. This poster presents the results of described parametric sensitivity studies. In addition, different values of the radiation view factors were varied. Evaluation of the radiation view factors showed that even small changes could cause the Cliff-edge effect on the calculation results.

KEYWORDS
QUENCH-20, BWR, RELAP/SCDAPSIM, parametric studies, hydrogen generation.

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IAEA
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ASSESSMENT OF HOW Zr-CLAD OXIDATION CAN AFFECT THE SPECIATION AND RELEASE OF FPs UNDER SEVERE ACCIDENT CONDITIONS

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ABSTRACT

Temperature and oxygen partial pressure (i.e. oxygen potential) are key parameters governing the speciation and release of fission products (FPs) from nuclear oxide fuels. Under severe accident conditions, large variations in oxygen potential are expected to occur within the fuel due to both temperature increase and oxidation of Zr-clad by water.

In this paper, a quantitative assessment of how Zr-clad oxidation can affect the speciation and release of fission products under accidental conditions is carried out using experimental feedback collected at CEA within the VERCORS and VERDON programs.

During the tests, total oxidation of the Zircaloy cladding occurs in the temperature range [600 °C - 1500 °C] and results in a strong release of di-Hydrogen, which causes a more or less significant reducing perturbation on the carrier gas ($Air/H_2O/O_2$) coming into contact with the fuel sample.

The magnitude of the reducing perturbation highly depends on the composition of the carrier gas. It is very large under un-buffered humid conditions (H_2O gas) whereas it can be much weaker, or even become non-existent, if the incoming carrier gas is initially buffered in O_2, using a H_2O/H_2 gas mixture. As soon as the Zr-oxidation stage is over, the oxygen potential suddenly increases to the level imposed by the incoming carrier gas.

The strong oxydo-reducing perturbation experienced by the gas interacting with the fuel is likely to significantly modify the speciation of volatile FPs (Mo, Cs, I) usually released in the temperature range where the cladding oxidation occurs.

KEYWORDS
Fission products, oxide fuel, zircaloy cladding, severe accident conditions, Hydrogen, source term

LPM VS. 3D ANALYSIS OF AN IN-VESSEL LBLOCA SEQUENCE USING THE ALMARAZ NPP GOTHIC CONTAINMENT MODEL

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ABSTRACT

Thermal-hydraulic codes are used to simulate possible accident scenarios in a nuclear power plant (NPP) containment building. However, its configuration is very large and complex and involves a wide range of different types of phenomena. Therefore, the thermal-hydraulics calculations in these types of simulations lead to a very demanding and challenging task, due to the difficulty of developing an optimal model that accurately represents the geometry while maintaining an affordable computational time, especially for 3D models.

Traditionally, Lumped Parameters (LP) models have been extensively used for licensing and severe accident analysis, as the models have proved to adequately represent average pressure and temperature trends with an affordable computational cost. However, they have limitations when it comes to simulating three-dimensional flow patterns, not being able to capture local effects caused by the containment geometrical constraints. Due to the increase of computational resources, these limitations can be overcome through the complementary use of 3D models, which provide a more detailed level of geometrical information, positioning them as a useful tool for the study of the 3D thermal-hydraulics inside the containment of a NPP.

GOTHIC is a thermal-hydraulic code with 3D capabilities, which uses a Cartesian mesh. It formulates the conservation equations using the porous media approach, which means that the geometry is constructed by defining simple solid shapes in the fluid domain. The solids, also known as blockages, can partially or completely occupy the cell volume and cell faces, meaning that the liquid and solid phases are allowed to coexist in the same cell unlike most of the CFD codes.

For this study, two GOTHIC models of the containment building of the Almaraz NPP have been compared. The first model is a single node LP type and the second one is a three-dimensional integral model, where all the rooms and compartments are represented using a single control volume.

In previous works carried out by the Universidad Politécnica de Madrid (UPM), the integral model of the Almaraz NPP was designed following a new methodology called "preventive methodology"(Vázquez-Rodríguez 2023). This methodology attempts to preventively avoid the appearance of problematic

configurations that lead to convergence errors and even the simulation interruption, such as the connection of two cells with significantly different free volumes. Therefore, this methodology consists of adapting the models geometrically to the Cartesian mesh, thereby decreasing the computational cost. The modelling process consists of developing a detailed CAD geometry from plant layouts, then simplifying the geometry in CAD in such a way that it complies with GOTHIC's modelling and domain discretization particularities.

The single node LP model has been developed from the 3D model, and particular attention has been paid to ensure the consistency of all parameters in both models. As a result, the free volume, material properties, thermal structures and boundary conditions are identical for both models.

The analyzed severe accident transient consists of a mass and energy release coming from a double-ended guillotine-break LOCA obtained from the UPM PWR-Westinghouse MELCOR model (Bocanegra et al. 2016) and the consequences of the transient evolution are unmitigated.

Subsequently, the results obtained from the LP and the 3D model of the Almaraz NPP were compared. Firstly, the averaged pressure and temperature values comparison shows a first pressure peak in both models almost identical (353 kPa), while the second pressure peak, caused by the zirconium oxidation, is slightly higher in the 3D model (36.861 kPa higher). This pressure difference can be explained by looking at the temperature evolution in each room of the 3D model, where, in certain rooms, the maximum temperature peak in that instant, exceeds the average temperature values.

This pressure and temperature difference might be caused by the nature of a single node LP model, in which, once the mass and energy are released, the vapor immediately interacts with all the thermal structures. As a result, the steam condensation occurs instantaneously, leading to an early cooling of the free volume and preventing the generation of sharp temperature peaks. In contrast, in the 3D model, there may be a time delay between the release of mass and energy and the interaction between the vapor and the containment structures. It is important to mention that it is possible to represent the temperature heterogeneity with LP models to a certain level by using a more complex nodalization.(Carlos Vazquez-Rodriguez et al. 2009)

Regarding the hydrogen concentrations, the LP model could only obtain an average value, not being able to reach the required local detail necessary to accurately predict the performance of hydrogen management systems. Meanwhile, the 3D distribution of the hydrogen demonstrates that locally, hydrogen concentration values are higher than the one obtained in the lumped model.

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KEYWORDS
GOTHIC, THERMAL HYDRAULICS, CONTAINMENT ANALYSIS.

SOME RESULTS OF THE AMICO PROJECT

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ABSTRACT

In the R&D project „Analyses for the modelling of the iodine behaviour in containment (AMICO)", co-funded by the German BMUV (project number 1501562), analytical work and COCOSYS calculations were carried out on different aspects of the fission product and especially the iodine behaviour in a severe accident. As part of AMICO Work Package (WP) 1.1 (Long-term development of the airborne iodine activity), the THAI test Iod-31 was calculated with COCOSYS V3 and the NewAFP module. In this experiment, after injecting iodine into the THAI test vessel, a water spray was activated for six defined time periods. This water spray transferred iodine from the atmosphere into the sump. Aim of the COCOSYS calculations was to reproduce the test results, especially the impact of the spray on the airborne iodine. It proved advantageous to define separate mass transfer coefficients (MTC's), which describe the exchange of iodine between aqueous phase and gas phase, for the spray system. MTC's which tend to be higher than the standard coefficients in COCOSYS and the simulation of the fixation of dissolved iodine in the sump proved to be suitable. Uncertainties result from experimental features that are difficult to model in the code, e.g. the short fall time of spray droplets, uncertainties about the surface of the vessel wetted by the spray cone and the iodine speciation in the sump. This affects the calculated reaction rates of iodine species in the gas phase, on dry and wet steel walls and in the sump. The COCOSYS calculations appear plausible overall, and yield indications for further experimental investigations to further validate the spray model in the code.

In a second step, the findings from the Iod-31 simulation on the effects of different mass transfer coefficients for the spray system were applied to a full-scale PWR containment simulation. As reference scenario, a large-break loss of coolant scenario was chosen, which, due to the assumed safety system failures, escalates into a nuclear accident with core damage. Three calculations were performed, each using a different set of pre-defined mass transfer coefficients for the spray system. The higher the resulting mass transfer coefficients for I_2 and for organic iodine (RI), the higher the predicted amount of airborne iodine. The increase of the RI in the variation calculations was higher than that of the I_2, making RI the dominant airborne iodine species. The higher amounts of RI and I_2 also lead to a stronger radiolytic formation of IO_x aerosols, which are washed out by the spray system and thus only accounted for about one percent of the airborne activity. This very clear result requires further analyses and evaluations.

In AMICO WP 1.2 and 1.3 (iodine modelling) various analytical and modelling work was carried out. For this purpose, a large number of new experimental data from national and international projects was evaluated and the existing modelling of the reaction with paint and steel, radiolytic iodine release from paint, radiolytic formation and decomposition of organic iodine in the gas phase and radiolytic nitric acid formation in the containment were checked and revised. Results of the AMICO WP 2 (CoPool modelling of a deep sump) and WP 3 (Iodine Uncertainty and Sensitivity Analysis) are not subject to this contribution.

KEYWORDS
iodine, organic iodine (RI), spray, mass transfer, modelling

Effect of the Stages of the Accumulators on THE Hydrogen Production During LOCA+SBO in BNPP VVER-1000

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Abstract
VVER reactors are Russian-designed PWRs and they are widespread over Asia, and Eastern Europe. Bushehr NPP, located in Iran includes one VVER-1000 V446 (Unit-1) in operation and two VVER-1000 V528 (Unit 2&3) under construction. BNPP unit-1 is somehow a unique VVER, as it is a combination of a German KWU design and a Russian VVER. One of the key features of the BNPP Unit-1 is related to its accumulator stages. The first stage of accumulators is VVER-dedicated and includes four tanks connected to the RPV. The second-stage accumulators are indeed a part of the KWU design and include eight tanks, connected to the loops. According to the plant configuration, second stage accumulators are activated only in Design Extension Conditions (DECs)BNPP-1 is the first VVER in the world, with two stages of accumulators, and some subsequent VVER models are equipped with two or even three stages of accumulators. Some research has been performed to assess the effectiveness of the second stage. These stages have different setpoints for the injection, the first stage injects at 5.88 Mpa, and the second stage at 2.5 Mpa. This paper aims to investigate the effect of the accumulators, especially the second stage on Hydrogen Production during various LOCA sizes, combined with SBO. These Scenarios are modeled by MELCOR 1.8.6. Prior to accident modeling, the steady-state is developed based on the engineering handbook of the plant. The Steady-state calculated parameters (e.g. cold/hot leg temperature, pressurizer pressure, primary pressure/level, bypass flow rate, primary flow rate, secondary pressure, secondary flow rate, and core pressure loss) have shown acceptable errors in comparison with the design values. The LOCA+SBO is modeled for some SBLOCA sizes (25mm, and 45mm), and the effect of the first/ second-stage accumulators on Hydrogen is being investigated in these break sizes. The next step is to model MBLOCA, and LBLOCA while considering different situations for the accumulators to achieve the amount of produced Hydrogen.

Keywords
VVER, Accumulators, Hydrogen, LOCA, SBO

COMPARATIVE STUDY OF TWO EXPERIMENTAL CONFIGURATIONS WITH AN INTERNAL COMPARTMENT IN THE PANDA FACILITY

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ABSTRACT

The containment building of a nuclear power plant is the last physical barrier preventing a radioactive leakage to the environment in the unlikely event of a severe accident, following the defense in depth principles [1]. Therefore, the evaluation of the different phenomena that could threaten the containment integrity, such as hydrogen accumulation, acquires crucial importance [2]. For this reason, several efforts have been made to experimentally recreate hydrogen risk scenarios in international collaborative frameworks such as the OECD/NEA HYMERES (Hydrogen Mitigation Experiments for Reactor Safety), HYMERES-2 or PANDA projects [3].

In these projects, experiments have been conducted at the large-scale PANDA facility at Paul Scherrer Institut (Switzerland) during the past years, allowing to extend the experimental database and to help validating the simulation codes used for safety analyses under severe accident conditions. These efforts are currently being extended within the framework of the OECD/NEA PANDA project, which includes two series (identified as P1A1 and P1A3) addressing the interaction of flow with complex structures, such as steam generator compartment models (representing the compartment where the steam generator is located), and activation of a spray phase during the experiment. The goal of the P1A1 and P1A3 experiments is to study how complex flow obstructions affect hydrogen distribution phenomena (e.g. mixing, stratification, natural circulation, etc.) and the modeling needs (physical models, mesh resolution, etc.) to analyze the experimental phenomena.

The steam generator compartment model for the P1A1 and P1A3 series was obtained through a scaling process based on a typical PWR containment building configuration. The test scenarios include different phases, namely: release of steam, injection of helium (used to simulate hydrogen) inside the steam generator compartment model and later activation of the spray system in the upper dome of PANDA Vessel 1. During the experiment, the containment is first pressurized (due to the steam and helium release) and later depressurized due to spray activation.

The difference between both configurations is purely geometrical: the P1A1 series includes one steam generator compartment model, as a first step to represent the interaction of the flow with typical PWR containment structures. Then, the P1A3 series adds a second compartment to its proposed configuration, with a connection between both structures. These modifications serve as a step forward to representing the containment building configuration with a higher level of detail, allowing the study of the interaction of the flow with additional structures, compared to P1A1, as well as its behavior through the interconnection. Both geometries have been modeled using GOTHIC, a thermal-hydraulic code for containment safety analysis. It solves the conservation equations for mass, momentum and energy for multicomponent, multiphase compressible flow, with the possibility of using both lumped parameters and/or multi-dimensional geometries, for three phases: vapor, continuous liquid and droplets [4]. Thereafter, both models have been compared to assess the impact of the addition of the second compartment on the model thermal hydraulics.

For both cases, pressure significantly increases in the PANDA vessel during the steam injection phase and remains nearly stagnant while helium is injected in the compartment, as the injected mass flow rate is not high enough to compensate the effect of the steam condensation. Lastly, the vessel is depressurized through the effect of the spray system activation. Although the values are very similar for both models, the pressure peak is slightly lower for the P1A3 series, due to the addition of the second compartment and the connecting tunnel. The introduction of additional metallic structures provides extra surface for steam condensation, consequently reducing the pressure peak.

In addition to the analysis of global magnitudes, the effect of the configuration modification was studied locally. The observation of the flow through the model suggests that the injected gases flow upwards, towards the upper compartment openings, and enter the vessel with enough momentum to induce mixing patterns, starting at the upper levels of the vessel dome and progressively descending to the lower parts of the vessel. To furtherly study this mixing effect, a criterion for defining the thermal stratification front was defined and calculated for every time step. As a result, it was observed that, this front is located at lower heights as the transient advances, confirming the presence of descending mixing patterns. The location of the stratification front is somewhat lower for the P1A3 series at every time step, implying that the presence of the second compartment promotes the mixing of the PANDA vessel atmosphere.

Subsequently, upon the close observation of the helium distribution during the helium injection phase, it was corroborated that the two compartments configuration presents rather lower local concentration values, in line with the theory of the enhanced mixing scenario. However, the observed differences are not significant and do not justify the need of performing the P1A3 experiments with the configuration proposed during the scoping calculations. Some modifications will be further studied, such as the increase of the injection flows, that would make the presence of the second compartment more relevant, as well as diverse geometrical modifications that allow the experiments to provide additional useful information for the experimental database, while still being representative of a generic containment configuration.

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KEYWORDS
PANDA, GOTHIC, HYDROGEN DISTRIBUTION, CONTAINMENT ANALYSIS, SPRAY SAFETY SYSTEM

Numerical Simulation of LIVE2D Two-Layer Melt Pool Experiment

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ABSTRACT

During a severe accident, the molten core migrates to the lower head of the reactor pressure vessel, eventually forming a melt pool. To ensure the in-vessel retention (IVR) of molten material within the reactor, it is necessary to maintain the heat flux of the melt pool sidewall below the critical heat flux at any angle. However, the RASPLAV experiment indicates that the stratification of the melt pool may pose a threat to the feasibility of IVR. Therefore, it is necessary to conduct experiments and simulations specifically for two-layer melt pools. This study simulates LIVE2D two-layer experiments under two different top boundary conditions. This study utilizes computational fluid dynamics (CFD) software Fluent for numerical simulations, incorporating models such as the WMLES turbulence model, the Solidification and Melting model, and the Volume of Fluid (VOF) multi-phase flow model. Due to the immiscibility, density stratification, and crusting formation requirements between the two materials, heat transfer oil and a eutectic salt mixture of 50% KNO3 and 50% NaNO3 are chosen as the materials for the LIVE2D two-layer experiments. In conclusion, the simulation results align well with the LIVE2D experimental data, validating the capability of CFD to simulate two-layer melt pools accurately. This approach establishes a solid foundation for simulating two-layer melt pools under corium materials in future studies.

KEYWORDS
In-vessel retention; Two-layer melt pool; Computational fluid dynamics (CFD); Interlayer crust; LIVE2D

SEVERE ACCIDENT R&D IN UJV GROUP

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ABSTRACT

The presented work summarizes the latest advancements in the ex-vessel corium interactions and cooling research and development (R&D) program carried out by the UJV Group – UJV Rez and its daughter company Research Centre Rez. The goal of this work is to provide the context of these activities, their synergies and interconnections, and to put them into the wider framework of development of nuclear energy in the Czech Republic. The high-level overview is provided in the first chapter. The R&D targeting GENII reactors focus mainly on severe accident research of VVER-1000 type of reactors and serves mainly to enhance the safety of the Czech VVER-1000 reactors in operation. An ex-vessel cooling strategy has been formulated and supported by a number of computational and experimental studies. The main outcomes are summarized in the second chapter. Part of the GENII program overlaps issues specific to new builds featuring GENIII and GENIII+ reactors. New materials, that can replace concrete in its applications inside the containment, resilient to high temperatures, radiation, and even chemical interactions with acids has been under development, based on geopolymer matrices. Results of all the test carried out so far show excellent behavior of these materials in prototypic conditions. An overview of these results is provided in the third chapter. In R&D focused on severe accidents in GENIV reactors, the gas-cooled fast reactor (GFR) concept is the main technology that has been developed. Ex-vessel strategy for this type of reactor was established, and a core catcher with innovative sacrificial materials has been under development, targeted to mitigate the consequences of a severe accident specifically in GFR with its unique corium compositions. Chapter four provides an overview of these activities.

KEYWORDS
Severe accidents, ex-vessel, core catcher, cold crucible

Investigation of IVMR Strategy for BNPP-1 VVER 1000

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Abstract

Bushehr-1 NPP, located in Iran is a VVER-1000 (type V-446) reactor that has some different features in comparison with other VVER-1000s, and even its reference plant i.e., VVER-1000 (Type V-320). These differences stem from the fact that BNPP-1 is a combination of VVER and KWU reactors. Some important features that may affect accident progression include different containment shape (spherical), different cavity, and more units of accumulators in BNPP-1. Bushehr-1 is operating since 2011, and currently, the most safety efforts for this plant is focused on SAMGs, and emergency plans development. According to the features mentioned above, SAM activities should be modeled dedicatedly for this plant, and other VVER- 1000s severe accident modeling may not be sufficiently reliable.

In Vessel Melt Retention (IVMR) is considered as a strategy to maintain the molten material within the vessel during severe accidents. During the IVMR, the reactor cavity is filled with water, and the heat from the molten material inside the vessel is transferred to this amount of water through the vessel wall. This strategy is incorporated mostly for small-size reactors (<500 Mw), and if it is aimed to perform this strategy for high-power reactors, it should be modeled more carefully. The important phenomena that should be analyzed in this strategy are related to heat fluxes from the molten material to the vessel, and the heat flux from the vessel wall to the outer water. The CHF limit is the determinant in the success of the IVMR. Currently, different strategies within SAMGs are being developed, and modeled for Bushehr NPP-1, and IVMR is a controversial one.

This paper aims to perform a preliminary assessment of the IVMR strategy for BNPP-1 VVER 1000, by considering it's dedicated features, especially the number of accumulators, and the cavity shape. BNPP-1 VVER 1000 incorporates two stages of accumulators. The first stage includes four borated water tanks with a setpoint of 5.88 Mpa, injecting the water directly into the vessel. The second stage includes eight tanks with setpoint of 2.5 Mpa, and injecting to the hot-legs, and cold-legs only during severe accidents. In comparison to BNPP-1 VVER 1000, the previous models have only the first stage, and most of the subsequent models have more than one stage of accumulators (two or even three stages). Most of the subsequent models also benefit from a core catcher which eliminates the need for IVMR strategy. In this research, the MELCOR code is used to analyze different LOCA sizes that are considered representative of SB/MB/LB LOCAs with SBO. The main goal is to calculate the heat fluxes, induced to the water, to conclude the effectiveness of IVMR, in different scenarios, and according to the second-stage accumulators. The additional water, provided by the accumulators can play an important role in increasing the inner vessel heat transfer and reducing the heat flux on the vessel wall.

Keywords

In-Vessel retention, VVER-1000, Severe Accidents

Assessment of RELAP5-3D condensation models for small modular reactor passive safety

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Abstract

This paper presents the RELAP5 condensation heat transfer models for small modular reactor (SMR) safety systems. The passive safety systems of advanced SMRs are designed such that steam released from the reactor vessel is condensed at the containment vessel by transferring heat to the outside cooling water pool. This heat removal system is referred to as a passive containment cooling system (PCCS). The steam condensation performance becomes greatly deteriorated due to the presence of non-condensable gases such as air. Previous experimental studies on PCCSs used small tube sizes for separate effects testing and large containment for integral effects testing. However, no study has considered scaling its evaluation with supportive data to validate the correlations and models for condensation that are suitable for SMRs. In this study, scaled geometry and representative test data were used to verify the scaling performance of RELAP5 condensation models in order to improve SMR thermal-hydraulics and safety analyses.

Keywords
Small modular reactor, passive safety, condensation heat transfer, RELAP5-3D, scaling

V&V of nuclear fuel oxidation behavior in sleeveless SiC-matrix during air ingress accident

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Abstract

As an advanced technology of high-temperature gas-cooled reactors (HTGRs), a direct cooling method has been proposed between a sleeveless fuel compact and helium coolant in the core. One crucial problem is air-ingress event following Depressurized-Loss-Of-Forced-Coolant (DLOFC) accidents associated with severe fuel oxidation at high temperatures.
Preventing unacceptable fuel degradation and minimizing risk of radioactive release to the environment, the fuel matrix needs resistance to oxidation. For reactor safety assessment, V&V study of the fuel oxidation behavior is needed; however, the experiments and modeling have not been fully developed. Thus, a long-term evaluation of fuel oxidation during the postulated accidental scenarios is poorly understood. To address this issue, this work focused on the investigation of oxidation behavior of silicon carbide, a promising candidate for fuel matrix, in air up to 1400 ℃. Based on the preliminary experiments, an empirical model for SiC oxidation kinetics was determined. To achieve the V&V of fuel safety assessment during the accidents in the HTGRs, validation tests were additionally conducted using real-sized High Temperature Test Facility (HTTF) at The University of Tokyo. The detailed experimental results, the proposed model, and V&V discussion will be presented at the conference.

Keywords
V&V, oxidation, high-temperature gas-cooled reactor, air ingress accident, sleeveless fuel compact

Numerical Analyses on Melt Water Interactions with Super Absorbing Polymers Added to the Cooling Water

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Abstract

The majority of the operating Nuclear Power Plants (NPPs) have not been designed to withstand a severe accident with core melting. Flooding of the cavity below the reactor pressure vessel (RPV) is one of the few options to mitigate an accident at a late stage. However, this has only been adopted by some of the operators of NPPs since success is not guaranteed and the melt water interaction bears the risk of an early containment failure due to strong steam explosions. Recently, an innovative potential solution to this problem was proposed, based on adding super absorbing polymers (SAP) to the coolant water in the cavity. The SAP material can take up water volumes orders of magnitude larger than their initial size and forms pools consisting of water and water-saturated particles. First scoping experiments have indicated the capability of the concept to enhance melt jet fragmentation and to suppress energetic interactions between melt and water. Additional experimental and analytical research is required to further develop and promote this promising concept.

However, current computational models for melt water interactions are presently not readily applicable for the simulation of coolant containing SAP. In a first step, the present paper discusses the properties of water/SAP coolant, which distinguishes it from pure water, and how these properties should affect the likelihood and potential strength of steam explosions. Then, the codes COCOMO-3D and IDEMO-3D, which are being developed at IKE, Stuttgart to model the premixing and explosion stages of steam explosions, have been partly enhanced and applied in parametric studies to investigate some of the effects under discussion, such as strongly increases apparent viscosity. The results are discussed considering their preliminary character due incomplete modelling. Further steps for developing FCI codes towards more realistic simulations of SAP/water pools are proposed.

Keywords

Severe Accidents, Fuel-Coolant-Interactions, Accident Mitigation, Multiphase Flow, Modelling

PLINIUS – experimental platform for nuclear excellence

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Abstract

As the world embraces nuclear energy as a viable solution for sustainable development and the fight against climate change, the understanding of severe accidents becomes vital to ensure the acceptance and continued growth of the nuclear industry in a safe and responsible manner. The importance of exploring and investigating severe accidents cannot be overstated, as this knowledge serves as the foundation for developing preventive measures and effective response strategies.
The recent achievements in successfully executed experiments on PLINIUS platform at CEA IRESNE, at Cadarache in France, underline significant progress in our collective efforts to enhance nuclear safety and sustainability.
Building upon past successes, this work emphasizes the significance of current experimental capacities to simulate and study severe accidents. The presented work highlights the ongoing R&D for GENII, III and GENIV as well as ongoing R&D for dismantling needs related to Fukushima Daiichi. In 2023, three big-scale experiments are successfully performed with dedicated and high performance instrumentation.. The experiments bring insight into the corium-concrete interaction under the top flooding in MERELAVA facility and the fuel-coolant interaction phenomena in KROTOS facility. Moreover, several blocks or corium samples are being fabricated in VITI and VULCANO facility for further studies related to leaching or debris cutting for post-accidental dismantling.
Recognizing the importance of international collaboration, our platform actively engages with partners across borders to facilitate knowledge exchange and leverage global expertise in the pursuit of comprehensive severe accident understanding. The current list of our engagements and future opportunities to collaborate is presented. Moreover, the presented approach in this work encompasses a modular framework that not only addresses the immediate R&D requirements. The CEA experimental team also offers adaptability of platform to address future needs in our ongoing endeavor to mitigate the risks associated with severe accidents.

Keywords: PLINIUS, experiments, severe accidents, R&D, collaboration

ASTEC VALIDATION OF SFP DEWATERING USING RESULTS FROM THE DENOPI PROJECT

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ABSTRACT

Loss of cooling in a Spent Fuel Pool (SFP) of a nuclear power plant can lead to the melting of fuel assemblies and to strong radiological consequences to the environment. In order to study the first phases of such accidents, up to the fuel assemblies uncovery, the DENOPI project was launched by the French Institute for Radiation Protection and Nuclear Safety (IRSN) supported and funded by the French Government and partners. Among the different facilities developed in the project, the MIDI facility aims at studying the complex thermal-hydraulics phenomena occurring in a large water pool heated from the bottom by electrical rods arranged in dedicated racks. MIDI is scaled by homothety to a typical French SFP. Different assembly arrangements (loading patterns) have been tested at different power levels, with either uniform power repartition, or hot and cold cells. In each test, the water level and temperatures at different elevations are followed, as well as mass flow rate entering each fuel rack. These experimental results also provide relevant data for the analysis and understanding of large natural convection loops that are expected in immersed passive heat removal systems of Small Modular Reactors. The forthcoming OECD/NEA POLCA project aims to extend such results database, in particular to assess the capability of thermal-hydraulics codes to reproduce the main tendencies of these experimental results.
The ASTEC code developed by IRSN is a system code dedicated to the simulation of major accidents in nuclear facilities that may lead to the release of radiological material. Recent works within the MUSA European project have shown the importance of reducing models uncertainties in the first phases of the accident, during the pool dewatering. In this paper, first simulations of MIDI tests are performed with ASTEC in order to assess and improve the capability of ASTEC to simulate the dewatering of a large water pool such as a SFP during a loss-of-cooling accident. Simulations are performed for a selection of MIDI tests with different heating patterns and power levels. Different models of subcooled boiling models from the literature are tested in ASTEC, stressing the key role of these models for an accurate prediction of the experimental flow.

KEYWORDS
Spent Fuel Pool, Loss-of-Cooling, ASTEC, subcooled boiling, natural convection.

FORMULATION OF MATERIAL PROPERTY FORMULA FOR CALCULATION OF DAMAGE IN REACTOR PRESSURE VESSEL DURING ACCIDENT EVALUATION

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ABSTRACT

From the results of the internal investigation of Fukushima Daiichi Nuclear Power Station Unit 2, it was confirmed that part of the fuel assembly (upper tie plate) had fallen to the bottom of the pedestal periphery. From this result, it could be presumed that RPV (Reactor Pressure Vessel) has a hole large enough for the upper tie plate to drop. One of failure mode of the RPV lower head would be assumed to be “mechanical failure”. In the mechanical failure, it is assumed that the RPV lower head will be damaged due to the accumulation of creep damage caused by core material above the creep temperature of the RPV substructure materials falling into the lower plenum. Such damage evaluation is performed by thermohydraulic-structure coupled analysis. Such analysis requires RPV material properties from the creep temperature range to near the melting point. In this study, we obtained the strength data of RPV material including welded joints form the creep temperature range to near the melting point and formulated the material property formulas (elastoplastic stress-strain formula, creep strain formula, creep rupture formula) necessary for mechanical failure evaluation.

KEYWORDS
Fukushima Daiichi Nuclear Power Station, Reactor Pressure Vessel, Material Property, Tensile, Creep,

The 11th European Review Meeting on Severe Accidents Research

May 13-16, 2024

KTH Royal Institute of Technology, Stockholm, Sweden

BOOK OF THE ABSTRACTS

11TH CONFERENCE ON
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FOREWORD

The 11th edition of the ERMSAR (European Review Meeting on Severe Accident Research) Conference, will be held from 13 to 16 May 2024 in Stockholm, hosted and locally organized by KTH, after 2 years for the previous 10th edition in Karlsruhe. For the first time it has been organized in the frame of the EURATOM SEAKNOT project and the Technical Area 2 of the NUclear GENeration II and III pillar (NUGENIA TA2) of SNETP Association, together with IAEA and OECD/NEA.

The Technical Programme Committee involved 19 researchers from diverse organizations (BT, CEA, CIEMAT, ENEA, Framatome GmbH, GRS, IAEA, INRN, IRSN, JSI, KIT, KTH, LGI, NEA and University of Pisa).

ERMSAR 2024 will gather about 146 participants from nearly 77 organizations settled in 26 countries worldwide (EU, USA, Canada, Republic of Korea, Brazil, Japan, India), which highlights ERMSAR as the reference international conference on Severe Accidents. 71 papers will be orally presented and 33 more will be exhibited in the posters stand. The conference is focused on the latest progress of international knowledge on severe accidents and is mainly an opportunity for researchers to discuss about future R&D priorities in this field. The Conference topics are highlighted in the following technical session list.

1. In-vessel corium and debris coolability
2. Ex-vessel corium interactions and coolability
3. Containment behavior incl. H2 explosion risk
4. Source term issues
5. Analysis, Management, and Consequences of Severe Accidents for Gen I/III reactors
6. Severe accidents in innovative reactor concepts such as Gen IV and Small Modular Reactors (SMRs)
7. Other applications (fusion reactors, interim SNF storage, Accident Tolerant Fuels (ATFs), etc.

In addition, the technical programme includes three plenary sessions, in addition to specific presentation on SEAKNOT and NUGENIA/TA2 latest activities and status:

- Singular careers in SA research
- Regulatory perspective and approaches for Severe Accidents in Small Modular Reactors
- Looking ahead in severe accident research

Next edition ERMSAR2026 will be again organized in the frame of the SEAKNOT project by CIEMAT Madrid in May 2026.

Editors: Fabrizio Gabrielli (KIT), Luis E. Herranz (CIEMAT), and Sandro Paci (University of Pisa)

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The Canadian Nuclear Regulatory Role, Approaches and Challenges for Severe Accidents in Small Modular Reactors

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ABSTRACT

The role of the Canadian Nuclear Safety Commission (CNSC) is to provide regulatory oversight of safe operation of nuclear power plants (NPPs) in Canada. One of the current regulatory challenges drawing attention is the impending deployment of novel reactor designs that are not water-cooled, including small modular reactors (SMRs). The challenges common to all international regulators and stakeholders include innovative nuclear technologies, such as accident-tolerant claddings and accident tolerant fuels (ATFs), Small Modular Reactors (SMRs) and first-of-a-kind novel reactor designs.

While CNSC has a robust and matured regulatory framework and requirements for oversight of severe accidents that includes an improvement since the Fukushima accident, most innovative SMR designs claim novel engineered passive safety features, innovative containment concepts and alternative approaches to demonstrating safety functions that need to be demonstrated in the event of a severe accident through adequate safety analysis including deterministic and probabilistic methods, especially with some designs crediting no operator actions for a prolonged period.

For SMRs proposed for deployment in remote locations, the demonstration of adequate provisions for severe accident management and emergency planning zones is a critical factor for their regulatory approval and licensing by the CNSC. This is particularly important as timely assistance during severe accidents becomes crucial, especially when emergency services are not readily available nearby.

The technology and design of SMRs are evolving rapidly, bringing about safety enhancements. While water-cooled SMRs may leverage previous large, water-cooled design provisions, passive components, no operator actions or interventions, and the capacity of SMRs to mitigate the consequences of severe accidents pose challenges. In response, proactive research development efforts are underway, including Phenomena Identification Ranking Tables (PIRT) for SMR designs, ATFs, in vessel retention (IVR) strategies, and strengthening of codes and regulations. These efforts, coupled with continuous dialogue between regulators, designers, operators, and standards organizations aim to position the CNSC to ensure nuclear safety, protect the public, and safeguard the environment in the event of a severe accident involving SMRs, advanced and conventional nuclear reactor systems in Canada.

In this presentation, CNSC regulatory challenges pertaining to expectations for severe accident and its mitigation for small modular reactors will be discussed, which confirms that CNSC is ready for future licensing of SMRs.

KEYWORDS
Severe Accident, Severe Accident Management, Small Modular Reactors