Twenty-Five Research Publications Involving RELAP5-3D from 2017-2018

George Mesina

April, 2018

Some of the many Research and Development publications involving RELAP5-3D from 2017 until April 1, 2018 are listed with bibliographical information and Abstract Summaries.

News about developments and applications of RELAP5-3D around the world is very helpful to researchers, developers, analysts, and the nuclear industry in general. It is hoped that this list of articles published between 1/1/2017 and 4/1/2018 will be informative and possibly lead to new collaborations and initiatives.

The article lists 25 scholarly publications on RELAP5-3D. In a similar article in the the 2017 Q3 newsletter, 22 more paper titles are presented. Abstracts for all the titles are available, indeed many of these articles can be downloaded from a simple web search; however for copyright considerations, either summaries of the abstracts or just the bibliographical information is recorded here.

Since the many search engines for the so-called "deep web" of science and engineering articles probe only a restricted number of journals and conferences each, there is no claim that the combined lists represent all research publications involving RELAP5-3D.

Some sites rate the relevance of research papers to the given search topic with a number of stars ranging from 0 to 5 with 5 being entirely about the topic and zero being irrelevant. These have been included for some of the publications.

The IAEA CRP on HTGR uncertainties: Sensitivity study of PHISICS/RELAP5-3D MHTGR-350 core calculations using various SCALE/NEWT cross-section sets for Ex. II-1a

Rouxelin, Pascal; Strydom, Gerhard; Alfonsi, Andrea; Ivanov, Kostadin
Nuclear Engineering and Design, Volume 329, Pages 156-166, 1 April 2018

Abstract Summary
The Coordinated Research Program (CRP) of the International Atomic Energy Agency (IAEA) applies Uncertainty and Sensitivity Analysis to High Temperature Gas-cooled Reactors. Phase II investigates propagated uncertainties from lattice to coupled neutronics/thermal hydraulics core

calculations. The analysis uses the two-dimensional lattice code NEWT, Serpent 2.1.27, the TRITON/NEWT-flux-weighted cross sections and the Idaho National Laboratory (INL) coupled code PHISICS/RELAP5-3D.

Verification of RELAP5-3D code in natural circulation loop as function of the initial water inventory

★★★★☆ *Bertani, C.; Falcone, N.; Bersano, A.; Caramello, M.; Matsushita, T.; De Salve, M.; Panella, B.*

Journal of Physics: Conference Series, Volume 923, Issue 1, article id. 012008, 2017-11-01.
Also
35th UIT Heat Transfer Conference, Ancona, 26-28/07/2017. pp. 1-10.

Abstract Summary

For high safety and reliability Generation IV and Small Modular Reactors (SMR) use passive systems that rely on natural circulation, which operate without external energy sources. The Decay Heat Removal system (DHR2) of ALFRED (Advanced Lead Fast Reactor European Demonstrator), the European Generation IV demonstrator of the fast lead cooled reactor uses the thermal-hydraulic system code RELAP5-3D, to study one- and two-phase natural circulation, particularly the effect of initial water inventory on natural circulation is analyzed. Simulations of the experimental tests using a 1D model at constant heat power and fixed liquid and air mass produce code predictions are comparable with experimental results. Numerical results show that low initial liquid mass inventory produces natural circulation that pulsated, not stable.

Potential Impact of Accident Tolerant Fuel Cladding Critical Heat Flux Characteristics on the High Temperature Phase of Reactivity Initiated Accidents

★★☆☆☆ *Maolong Liu; Nicholas R. Brown; Kurt A. Terrani; Amir F. Ali; Edward D. Blandford; Daniel M. Wachs*

Annals of Nuclear Energy 2017-12-01, Volume 110, December 2017, Pages 48-62

Abstract Summary

Scoping RELAP5-3D simulations are used to assess the potential impact of boiling heat transfer coefficients and critical heat flux (CHF) on peak cladding temperature (PCT) of key accident tolerant fuel (ATF) cladding types in a design basis accident (DBA), namely Hot Zero Power (HZP) Reactivity-Initiated Accident (RIA) in a Pressurized Water Reactor (PWR). The study focuses on bubble crowding CHF. Results include sensitivity of PCT, the duration of film boiling to key parameters in the boiling curve: nucleate, transition, and film boiling heat transfer coefficients, and CHF. Perturbations in boiling heat transfer characteristics for Zircaloy-4 cladding and key candidate ATF cladding types, such as FeCrAl and SiC/SiC claddings, are studied.

Pool temperature stratification analysis in CIRCE-ICE facility with RELAP5-3D© model and comparison with experimental tests

Narcisi, V.; Giannetti, F.; Tarantino, M.; Martelli, D.; Caruso, G.
Journal of Physics: Conference Series, Volume 923, Issue 1, article id. 012006, 2017-11-01.

Abstract Summary
The capability of the system code **RELAP5-3D**© to simulate mixed convection and thermal stratification phenomena in a HLM pool in steady state conditions is investigated by comparing code results with experimental data from the heavy liquid metal (HLM) CIRCE pool facility at ENEA/Brasimone Research Center, updated with ICE (Integral Circulation Experiments) test section which simulates the thermal behavior of a primary system. The pool was simulated by a 3D component with 1728 volumes, 119 of which were centered exactly on the thermocouples and a one-dimensional nodalization of the primary main flow path. Results of axial, radial and azimuthal temperature profile in the pool are in agreement with available experimental data and the code is able to well simulate operating conditions into the main flow path of the test section.

THALLIUM: An experimental facility for simulation of HCLL In-box LOCA and validation of RELAP5-3D system code

M. Utili; A. Venturini; M. Lanfranchi; P. Calderoni; A. Malavasi; M. Zucchetti
Fusion Engineering and Design 2017-11-01, Volume 123, November 2017, Pages 102-106

Abstract Summary
Experimental facility THALLIUM (Test HAmmer in Lead LithIUM) at the ENEA Brasimone Research Centre was designed to reproduce the hydraulic transient of the HCLL (Helium Cooled Lithium Lead) TBS (Test Blanket System) breeder loop following the rupture of a Cooling or a Stiffening Plate. It reproduces the geometry of the LLE (Lead Lithium Eutectic) loop of the HCLL TBS and it was installed. It can also be used to validate pressure wave propagation capability of the RELAP5-3D system code.

An analytical study of the pressure wave show that the fluid-structure interactions will decrease the speed of sound in the LLE. However, since RELAP5-3D does not consider fluid-structure interactions, the calculated wave speed differ from the experimental data. The RELAP5-3D nodalization and results from a pre-test simulation are presented and proved useful for the design of the facility.

Pre-test analysis of protected loss of primary pump transients in CIRCE-HERO facility

Narcisi, V.; Giannetti, F.; Del Nevo, A.; Tarantino, M.; Caruso, G.
Journal of Physics: Conference Series, Volume 923, Issue 1, article id. 012005, 2017-11-01.

Abstract Summary

The LEADER (Lead-cooled European Advanced Demonstration Reactor) project proposed a configuration for ALFRED (Advanced Lead Fast Reactor European Demonstrator), a super-heated steam generator, double wall bayonet tube type with leakage monitoring. To support the new steam generator concept, the ENEA CIRCE pool facility will refurbish the HERO (Heavy liquid mEtal pRessurized water cOoled tubes) test section to investigate a bundle of seven full scale bayonet tubes in ALFRED-like thermal hydraulics conditions.

The aim of this work is to verify thermo-fluid dynamic performance of HERO during the transition from nominal to natural circulation condition. **RELAP5-3D**© simulations using the validated geometrical model of the previous CIRCE-ICE test section show the HERO bayonet bundle offers excellent thermal hydraulic behavior and allows operation in natural circulation.

Experimental and RELAP5-3D results on IELLLO (Integrated European Lead Lithium LOop) operation

A. Venturini; M. Utili; A. Gabriele; I. Ricapito; A. Malavasi; N. Forgione
Fusion Engineering and Design 2017-11-01, Volume 123, November 2017, Pages 143-147.

Abstract Summary

The experimental facility IELLLO (Integrated European Lead Lithium LOop) was designed and installed at the ENEA Brasimone Research Centre to support the design of the HCLL TBM (Helium Cooled Lithium Lead Test Blanket Module),to collect experimental data, to evaluate performances of commercial instrumentation, and to validate the model developed with RELAP5-3D. The RELAP5-3D simulations fit the associated experimental results achieved very well.

Comparison between RELAP5 versions for a two-phase natural circulation analysis

Braz Filho, Francisco A.; Ribeiro, Guilherme B.; Sabundjian, Gaianê; Caldeira, Alexandre D., E-mail: fbraz@ieav.cta.br, E-mail: gbribeiro@ieav.cta.br, E-mail: alexdc@ieav.cta.br,
Conference: INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017

Abstract Summary

RELAP5 is one of the most used numerical tools to predict thermal-hydraulic and neutronic phenomena in nuclear reactors. RELAP5-3D is the latest version of this software family, but RELAP5-mod3 is used as benchmark for several nuclear applications and is still widely used in Brazilian research institutes. Comparison between RELAP5-3D and RELAP5-mod3 regarding use of passive heat transfer mechanisms, such as natural circulation, has drawn attention of several studies, especially after the Fukushima-Daiichi accident. When compared to the experimental data set, the RELAP5-3D version provided the best prediction results.

Benchmark Simulation of Natural Circulation Cooling System with Salt Working Fluid Using SAM

Ahmed, K. K.; Scarlat, R. O.; Hu, R.
Conference: 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-17), 09/03/17 - 09/08/17, Xi'an, Shaanxi, CN.

Abstract Summary
A RELAP5-3D model of a Fluoride Salt-Cooled High-Temperature Reactor (FHR), the Mark-1 Pebble-Bed FHR (Mk1 PB-FHR), and in particular its Direct Reactor Auxiliary Cooling System (DRACS) loop for emergency heat removal, provides steady state and transient results for flow rates and temperatures in the system that are used here for code-to-code comparison with System Analysis Module (SAM) from Argonne National Laboratory for a loss of forced circulation with SCRAM event.

Design and Testing for a New Thermosyphon Irradiation Vehicle

Felde, David K.; Carbajo, Juan J.; McDuffee, Joel Lee
DOI: 10.2172/1399956, ORNL/TM-2017/399, OAK RIDGE NATIONAL LABORATORY, Oak Ridge, TN 37831-6283, 2017-09-01

Abstract Summary
RELAP5-3D and TRACE, were used to simulate the tests he High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL).

Risk-Informed External Hazards Analysis for Seismic and Flooding Phenomena for a Generic PWR

Parisi, Carlo; Prescott, Steve; Ma, Zhegang; Spears, Bob; Szilard, Ronaldo; Coleman, Justin; Kosbab, Ben
INL/EXT-17-42666, Idaho National Laboratory, Idaho Falls, Idaho 83415, *SC Solutions, Marietta, Georgia, https://www.osti.gov/scitech/servlets/purl/1376899, **http://www.inl.gov/lwrs**, Jul 28, 2017.

Abstract Summary
This report describes the activities performed during the FY2017 for the US-DOE Light Water Reactor Sustainability Risk-Informed Safety Margin Characterization (LWRS-RISMC), Industry Application #2. The scope of Industry Application #2 is to deliver a risk-informed external hazards safety analysis for a representative nuclear power plant. Following the advancements

occurred during the previous FYs (toolkits identification, models development), FY2017 focused on: increasing the level of realism of the analysis; improving the tools and the coupling methodologies. In particular the following objectives were achieved: calculation of buildings pounding and their effects on components seismic fragility; development of a SAPHIRE code PRA models for 3-loops Westinghouse PWR; set-up of a methodology for performing static-dynamic PRA coupling between SAPHIRE and EMRALD codes; coupling RELAP5-3D/RAVEN for performing Best-Estimate Plus Uncertainty analysis and automatic limit surface search; and execute sample calculations for demonstrating the capabilities of the toolkit in performing a risk-informed external hazards safety analyses.

Transient Simulation of the Multi-SERTTA Experiment with MAMMOTH

Ortensi, Javier; Baker, Benjamin; Wang, Yaqi; Schunert, Sebastian
https://www.osti.gov/scitech/servlets/purl/1376883, Reactor Physics Design and Analysis
Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415-3840, 2017-07-11

Abstract Summary This work details the MAMMOTH reactor physics simulations of the Static Environment Rodlet Transient Test Apparatus (SERTTA) conducted at Idaho National Laboratory in FY-2017. TREAT static-environment experiment vehicles are being developed to enable transient testing of Pressurized Water Reactor (PWR) type fuel specimens, including fuel concepts with enhanced accident tolerance (Accident … The TREAT core results compare well with the safety case computed with point reactor kinetics in RELAP5-3D.

MODELING THE AMBIENT CONDITION EFFECTS OF AN AIR-COOLED NATURAL CIRCULATION SYSTEM

Hu, Rui; Lisowski, Darius D.; Bucknor, Matthew; Kraus, Adam R.; Lv, Qiuping
Proceedings of the 25th International Conference on Nuclear Engineering (ICONE-25), 07/02/17 - 07/06/17, Shanghai, CN, 2017-07-02.

Abstract Summary
The Reactor Cavity Cooling System (RCCS) is a passive safety concept under consideration for the overall safety strategy of advanced reactors such as the High Temperature Gas-Cooled Reactor (HTGR). One such variant, air-cooled RCCS, uses natural convection to drive the flow of air from outside the reactor building to remove decay heat during normal operation and accident scenarios. The Natural convection Shutdown heat removal Test Facility (NSTF) at Argonne National Laboratory ("Argonne") is a half-scale model of the primary features of one conceptual air-cooled RCCS design. An empirical model was implemented in the computational models of the NSTF using both RELAP5-3D and STARCCM+ codes. Accounting for the effects of ambient conditions, simulations from both codes predicted the natural circulation flow rates very well.

Steady state HTTR model in the RELAP5-3D

Scari, Maria E.; Reis, Patrícia A. L.; Costa, Antonella L.; Pereira, Claubia; Veloso, Maria A. F., E-mail: melizabethscari@yahoo.com, E-mail: patricialire@yahoo.com.br, E-mail: antonella@nuclear.ufmg.br, E-mail: claubia@nuclear.ufmg.br, E-mail: dora@nucle…
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

Spent fuel pool thermal-hydraulic analysis using RELAP5-3D

Ramos, M. C.; Fernandes, G.H.N.; Costa, A.L.; Pereira, F.; Pereira, C.
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

Brayton Cycle Numerical Modeling using the RELAP5-3D code, version 4.3.4

Longhini, Eduardo P.; Lobo, Paulo D.C.; Guimarães, Lamartine N.F.; Filho, Francisco A.B.; Ribeiro, Guilherme B.
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

Thermal hydraulic simulation of the CANDU nuclear reactor

Carvalho, Athos M.S.S. de; Ramos, Mario C.; Costa, Antonella L.; Fernandes, Gustavo H.N.
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

Thermal hydraulic core simulation of the MYRRHA Reactor in steady state operation

Ferandes, Gustavo H.N.; Ramos, Mário C.; Carvalho, Athos M.S.S.; Cabrera, Carlos E.V.; Costa, Antonella L.; Pereira, Claubia
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

TRISO fuel thermal simulations in the LS-VHTR

Ramos, Mario C.; Scari, Maria E.; Costa, Antonella L.; Pereira, Claubia; Veloso, Maria A.F.
Proceedings of INAC 2017: International Nuclear Atlantic Conference, Belo Horizonte, MG (Brazil), Oct 22-27, 2017.

Improvement of the RELAP5-3D Model of Condensation in the Presence of Noncondensables

Nolan A. Anderson; George L. Mesina

Proc. ASME 2017 Nuclear Forum collocated with the ASME 2017 Power Conference Joint With ICOPE-17, the ASME 2017 11th International Conference on Energy Sustainability, and the ASME 2017 15th International Conference on Fuel Cell Science, Engineering and Technology, ISBN: 978-0-7918-4059-7, ASME 2017 Nuclear Forum, Charlotte, North Carolina, USA, June 26–30, 2017.

Abstract Summary

Condensation of steam on the primary side of a steam generator in a pressurized water reactor (PWR) is one means of removing decay heat during some accident scenarios, including small break loss of coolant accident (SBLOCA). However, when noncondensable gasses mix with steam, it impairs condensation. To correctly predict plant behavior, nuclear power plant (NPP) safety analysis codes such as RELAP5-3D must model the effect of condensation in the presence of noncondensables properly.

Enhanced Verification for RELAP5-3D Parameter and Sensitivity Studies

Mesina, George L.; Anderson, Nolan A.

Proceedings of the 24th International Conference on Nuclear Engineering, Volume 3: Thermal-Hydraulics, ISBN: 978-0-7918-5003-9, doi:10.1115/ICONE24-61040, Paper No. ICONE24-61040, pp. V003T09A084; 7 pages, Charlotte, North Carolina, USA, June 26–30, 2016.

Abstract Summary

For SQA (software quality assurance), the code must be verified and validated (V&V) to ensure proper performance before release to users. Previous articles have covered the verification of the physical models, restart, and backup through extremely accurate and automated sequential verification applied on a comprehensive suite of test cases to ensure that code changes produced no unintended consequences. New developments have enabled the verification of multi-case and multi-deck processing. These features are frequently used in parameter and code sensitivity studies and therefore must be verified as working correctly. Both theory and application are presented.

RELAP5-3D Simulation of Natural Circulation in Fast Reactors with Lead-Bismuth Eutectic Alloy Coolant

Matev, Alex

ASME Journal of Nuclear Engineering and Radiation Science, doi:10.1115/1.4036986, Accepted manuscript posted June 9, 2017.

PHISICS/RELAP5-3D Adaptive Time-Step Method Demonstrated for the HTTR LOFC#1 Simulation

DOE OSTI.GOV
Baker, Robin Ivey; Balestra, Paolo; Strydom, Gerhard
2017-05-01
DOI: 10.2172/1374506

Preliminary numerical studies of an experimental facility for heat removal in natural circulation

Bertani, C.; De Salve, M.; Caramello, M.; Falcone, N.; Bersano, A.; Panella, B.
Journal of Physics: Conference Series
2017-01-01
DOI: 10.1088/1742-6596/796/1/012046 Volume: 796 Issue: 1 Pages: 012046

Verification of RELAP5-3D code in natural circulation loop as function of the initial water inventory

C Bertani[1], N Falcone[1], A Bersano[1], M Caramello[1], T Matsushita[2], M De Salve[1] and B Panella[1]
Published under license by IOP Publishing Ltd
Journal of Physics: Conference Series, Volume 923, conference 1
J. Phys.: Conf. Ser. 2017-01-01