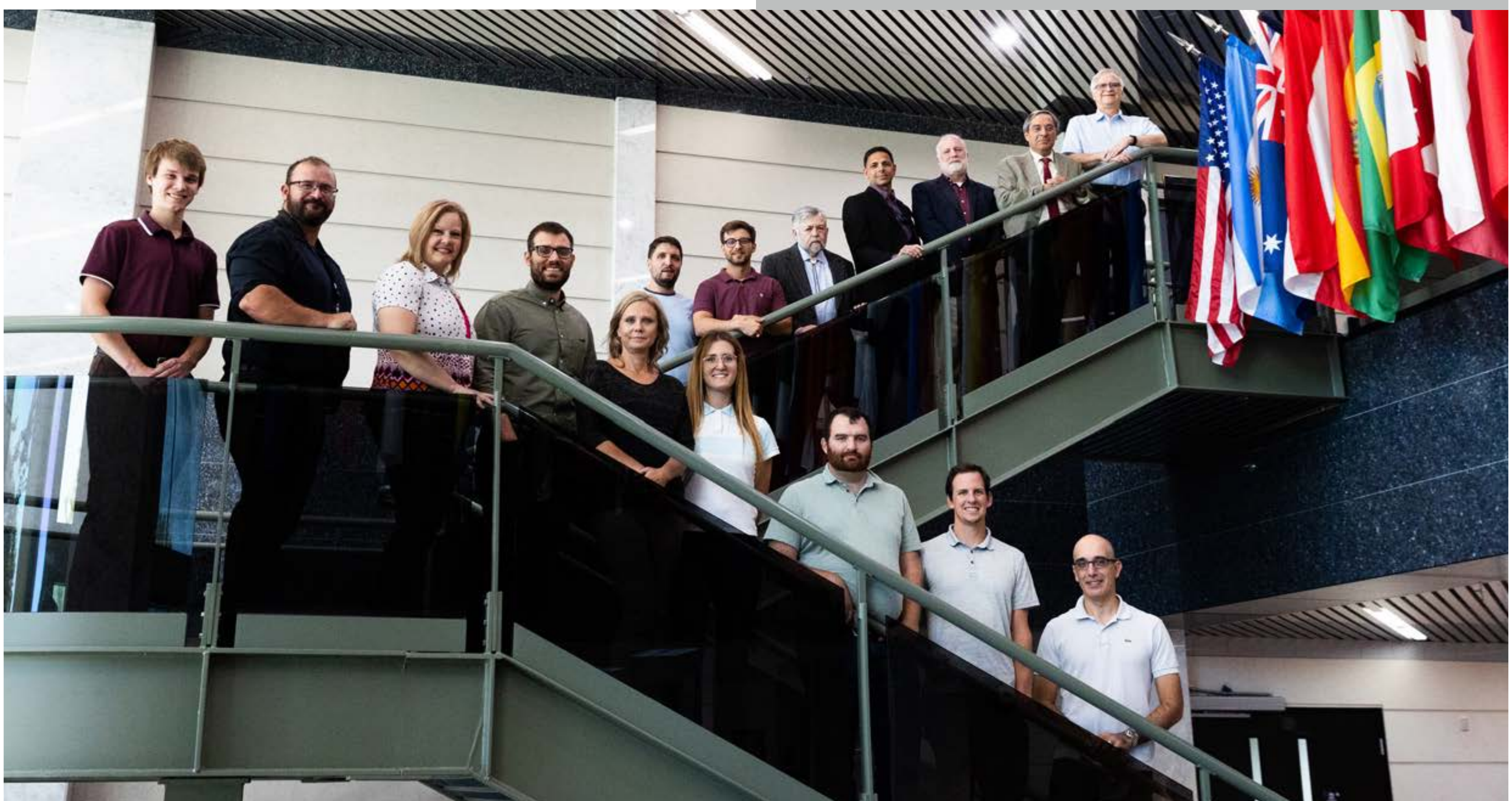
45th ANNIVERSARY

RELAP5-3D

Reactor Excursion & Leak Analysis Program



2024 International RELAP5 User Group

Idaho National Laboratory

45th Anniversary –
RELAP Throughout the Ages

Executive Summary

RELAP5-3D is the 45-years-young, internationally acclaimed computer program and one of the most successful Idaho National Laboratory (INL) programs of the last half-century. As part of the RELAP series that dates back to 1966, the code has a proud heritage and extensive usage in the burgeoning nuclear industry. This document presents the foundational principals, history, people, events, and actions that have produced a code that stands the test of time. RELAP5 is constantly under development and renewal, its team are dedicated to its success, INL management is committed to its continuance and advancement, a new consortium of laboratories has formed to marshal resources for its development, it is heavily used throughout the nuclear industry, and it has application and usage beyond nuclear. RELAP5-3D has a very bright future and is truly the premier systems code for modeling nuclear power plants today and will be so in the future.

CONTENTS

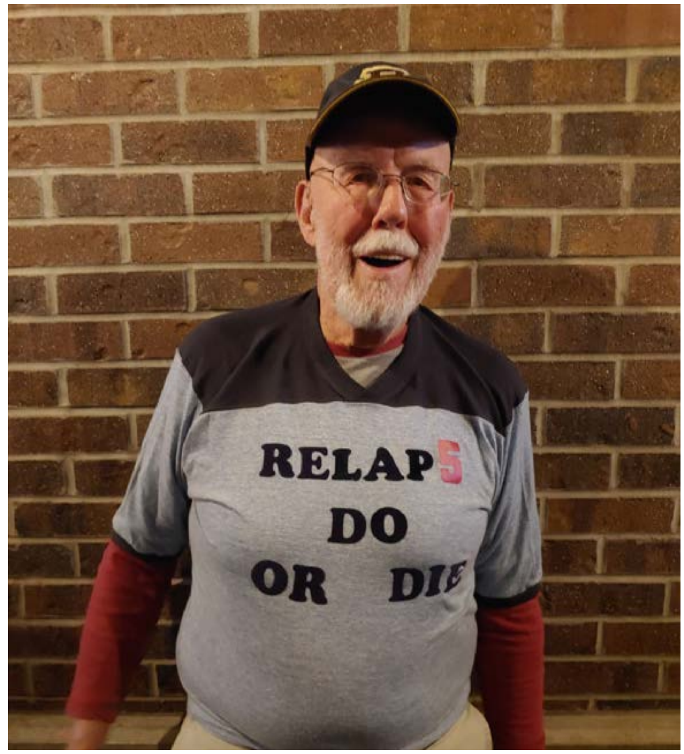
INTRODUCTION

Welcome to the 45th Anniversary of RELAP5!

Forty-five years. That is quite a milestone, especially for a computer program. How many other computer programs are still around 45 years later? Certainly no operating system, game software, or business program is. 45 years and it is still the premier program in its field, the most requested computer program at INL, and the largest license fee generator. How did INL and RELAP5-3D get here?

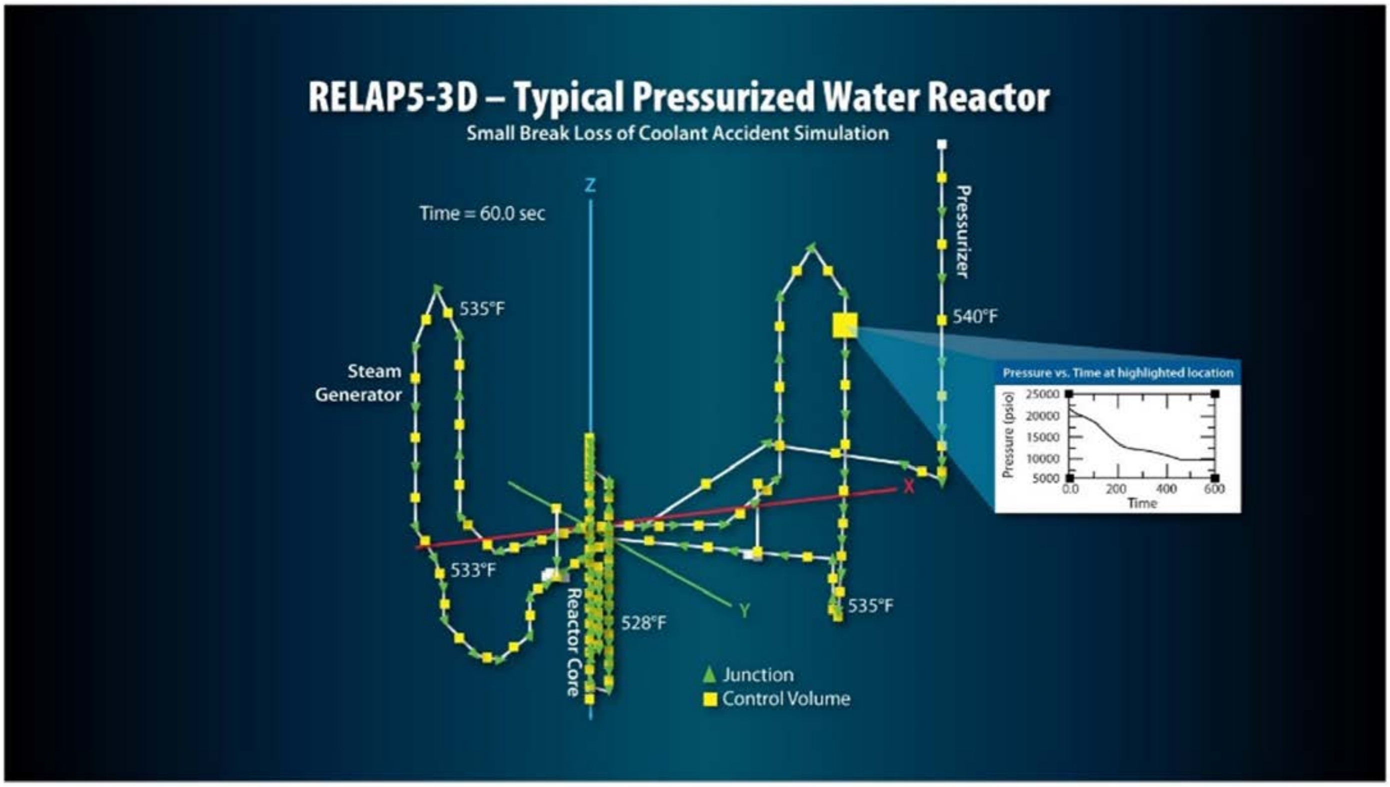
Our storied code got its start in 1979 after RELAP4, and the RELAP series, was terminated. Its new idea was to model the second fluid phase with equations, rather than using the standard Homogeneous Equilibrium Model (HEM) of the day. The small team had to prove the new concept could succeed and had little funding, but they put in lots of extra time, working late and on weekends. They wore T-shirts to work that said, "RELAP5 Do or Die!" That is the spirit that has spanned these 45 years. From its humble beginnings as a 4000 line "pilot code," the team has grown RELAP5 into the 350,000-line gold standard for system codes modeling nuclear power plants.

Any developer familiar with the code in its original form would not recognize it today. It has gone through extensive changes to keep up with the needs of the nuclear power industry and to stay abreast of the advances in the computer industry. The latter required adaptation to numerous vendor operating systems including CDC, Cray, IBM, HP, Sun, SGI, DEC, Windows and many different levels of those, not to mention all their native Fortran compilers. Such changes have always served to make the code more robust, for each caught different kinds of drawbacks and errors. Moreover, ANSI Standard Fortran also kept changing. Originally written in Fortran 66, RELAP5 has been converted to Fortran 77, Fortran 90, Fortran 2003, Fortran 2008 and finally GNU Fortran. For speed, the code has been vectorized, parallelized, and optimized with higher speed solvers and other improvements.



Vic Ransom wearing the RELAP5 "Do or Die" T-shirt

Over the years, RELAP5-3D has improved in accuracy and robustness from many factors. First, RELAP5-3D has adopted most every code development practice that computer science has produced. These have included code coverage analysis, memory analysis, initializations, better internal documentation, unit tests, validation, and verification. Second, billions of computer runs by our international user community have resulted in over two thousand User Trouble Reports (UTR), which computer science alone cannot find, that were reported and fixed. Third, as a result of the first two and the incorporation of tests for new features and capabilities, our test suites have continued to grow larger and test more. Each public code release is more thoroughly tested than the one before.



RGUI Displaying a Small Break Loss of Coolant Accident Conditions at 60 Seconds

Nowadays, RELAP5-3D is used to model much more than its original design specifications, which were Large Break Loss of Coolant Accidents (LBLOCA) for Pressurized Water Reactors (PWR). In the 1980s, the capability to model Boiling Water Reactors (BWR), Small Break LOCAs, and operational transients. The current code can model all commercial nuclear power plants of generations I, II, III, and IV. Beyond that, RELAP5-3D can also model Supercritical CO2 and Water Reactors (sCO2 and sH2O), High Temperature Gas-cooled Reactors (HTGR), Liquid Metal Reactors (LMR), and Molten Salt Reactors (MSR). And RELAP5-3D is always expanding to model the new reactor designs as they are being developed. In fact, RELAP5 can also be used to model other systems. We have 31 standard fluids to date and many more to add as needed. These include the fluids of the above-mentioned reactors, fluids used in experiments, refrigerants, even human blood. These are available because RELAP5-3D has many more applications than just modeling nuclear reactors.

The 45th Anniversary Booklet gives a short history of all the incredible work and long hours of development that produced the world's premier nuclear power plant systems code. Join us on the journey, meet the people who made it possible, learn about the things they did. And always remember:

RELAP5, Do or die!

Dr. George Mesina
RELAP5-3D Code Architect

Chapter 1. Value and Impact of RELAP5-3D

The impact and intrinsic value of RELAP5-3D can be demonstrated in many ways. Some of these include the speed and accuracy with which it performs its calculations, the help it provides to engineers analyzing existing power plants, the real feel it gives nuclear power plant operators in RELAP5-based simulators, its accurate evaluations of new reactor designs, and its support of nuclear power plant vendors, through Commercial Grade Dedication (CGD) for making license submittals to the Nuclear Regulatory Commission (NRC).

When RELAP5 was first conceived, several principals were established about its programming philosophy so that it would outclass its rivals and stand the test of time. These are speed; user-friendly; robustness; always up to date; and programming uniformity.

CODE RUNSPEED

Code runtime, the fastest among codes with comparable fidelity, has long been RELAP's largest selling point. RELAP5 used dozens of programming concepts to save time without compromising solution accuracy. For example, because reads and writes to disk are 100 times slower than memory operations, the code read all input from a disk file into a single array named fa, short for fast, disconnected the file, and worked entirely in memory. Because common blocks provided significantly faster memory than call arguments, most subprograms had no call arguments. Since older Fortran had no pointers, indexing of the FA array provided the capability, and this proved significantly faster than pointers once they became available. Equivalence statements to fa-array subsections gave mnemonic names for quantities like pressure and temperature. Programming was immensely complex and extremely difficult, but the code ran very fast. In modern Fortran, the code takes about twice as much time to run, but it is much easier to read and program.

Besides those tricks that have been replaced by conversion to Fortran 90, RELAP5 maintains two unusual speed tricks. First is multiplying by the reciprocal. Since division takes 3 times more clock cycles than multiplication, rather than divide every member of an array by the same value, that number's reciprocal is calculated and then multiplies each array element.

Something unique to RELAP5 is timestep backup. The 5 major multi-physics are calculated in sequence: trips, heat transfer, thermal fluids, neutronics, and controls. When the hydrodynamics calculation produces too large an error, normally the trips and heat transfer calculations would be discarded, the timestep halved, and two advancements at half the timestep would be taken to reach the targeted time. However, when flow reversal, noncondensable gas appearance, or water hammer in a single location causes the error, RELAP5 can maintain its timestep by backing up to redo only the hydro solution. The equations for the trouble location are written as if the condition already existed. This saves the trips and heat transfer solution and prevents the taking two full timesteps at half the size to get to the same time target.

Many other speed efficiencies are programmed into RELAP5.

USER FRIENDLY

RELAP5 was built with the user in mind. Until the mid-1990s, computer centers charged for computer time, computer memory, and paper. RELAP5 kept costs down for users in several ways. Arrays are declared as small as possible to reduce memory charges. It ran fast to reduce time charges. With small memory size, the code could even run entirely in cache memory and take about one tenth the time.

The most important user-friendly feature, which no other nuclear plant analysis code had, was that it always told the user every input error on a single run. This feature saved time, memory, and paper for the user. The other codes would tell the user the first error in input then quit. RELAP5 would substitute its internal water properties to overcome fluid property errors, then run to completion. This created some incorrect messages, much as a compiler does, but saved users a lot of money.

Error messages were explicit and helpful, or else became so after user trouble reports about the messages. Additionally, the RELAP5 team had a reputation for responding quickly to user trouble reports. The first graphical user interface for nuclear programs was developed for RELAP5, called the nuclear plant analyzer, helped users visualize the happenings in the transient. Plot programs and input aids were also developed.

ROBUSTNESS

One major principle for RELAP5 was asserted from the beginning by original code architect, Dick Wagner. "If it gets through input processing, it should run the entire transient."

To this end, testing always expands when new features are added to ensure that input processing and transient calculation failures produce clear and concise messages. When error conditions are found, they are catalogued, prioritized, and worked on in order. User trouble reports concerning any kind of code failure are given importance based on user needs.

STAYING CURRENT

The code also continues to adapt to changes in the computer architecture, operating systems, compilers, and other developments in the computer industry to ensure the code runs properly despite the changes. Also, as clients approach INL for upgrades to RELAP5-3D to meet their modeling needs, the team continues to add these features to the code. When doing so, it is important to avoid breaking other features and maintain overall code robustness.

UNIFORMITY and MAINTAINABILITY

Uniformity makes it easier for new team members to read and understand the code from one major section to another. A uniform style helps when finding errors, making improvements, and adding new code developments. Routine maintenance for changes to computer operating systems and computer architecture requirements are also aided by uniformity

From its creation, the code architect would rewrite any coding that was submitted and did not meet the code style guidelines. Uniformity was maintained in this way. When he retired, a code style guideline was written and enforced within coding produced by INL developers.

ANALYSIS and DESIGN

When putting a model together, the analyst has several resources. Volume 2 explains RELAP5 input modeling. The specifics of constructing lines of an input files are clearly and explicitly detailed in Volume 2 Appendix A. Volume 5 gives expert tips on improving an input model to handle many different kinds of issues and goes far beyond the instructions in Volumes 2 and 2A. Several graphical user interfaces (GUI) are available to help visualize the model during and after model construction. A recent development has been the use of spreadsheets to document and perform and the calculations that go into the model, and to simplify model modifications when needed. RELAP5 itself provides explicit feedback on input construction errors and other useful information about the input model that can help refine a model before running it.

The code's printed output presents easy-to-read snapshots of the transient calculation. The plot files, which are available in four formats from compact machine-dependent binary to large comma-separated-value files, can be displayed through several commercially available tools. Restart files have been perfected so that restarted transient calculations exactly match the original calculation unless the user modifies the model or output specifications. Graphical user interfaces for displaying the calculations are available to aid the engineer in analyzing reactor performance and aid in design of new plants.

Besides these features and capabilities, RELAP5-3D provides multi-dimensional physics in heat transfer, hydrodynamics, and neutron kinetics. It can couple with other computer programs on complex problems through domain decompositions so that another code, such as a computational fluid dynamics (CFD) code or containment code, provides more precise calculations. Five means or coupling RELAP5-3D with other codes exist, but most prominent of which is INL's R5EXEC program. It can model many kinds of existing plants and is easily adapted to modeling more. It is used on nuclear power plant training simulators and, with a commercial grade dedication (CGD), used by companies for licensing submittals to the NRC.

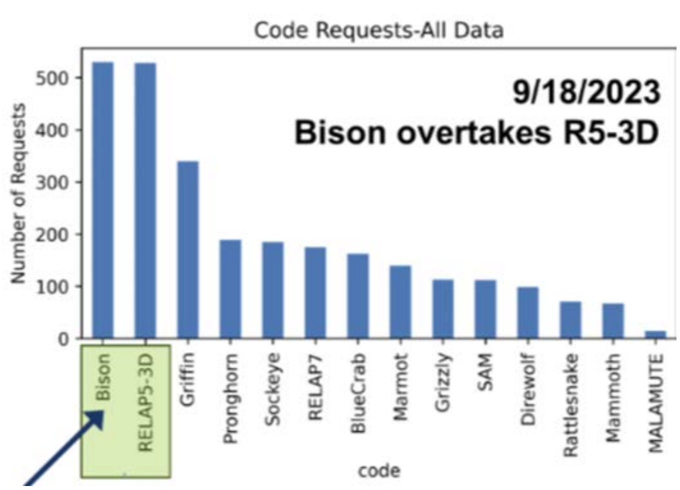
WHY IS RELAP5-3D THE BEST?

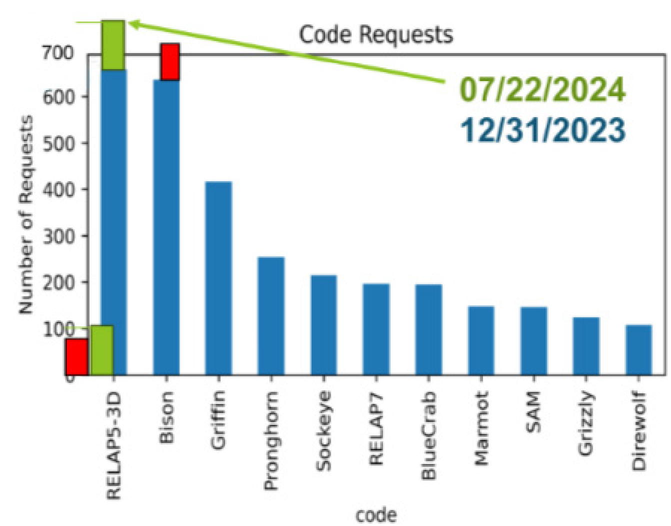
"RELAP5-3D leverages the collective knowledge of hundreds of nuclear reactor experts and benefits from the experience of literally thousands of users. As such RELAP5-3D has not only succeeded in its long-held nuclear safety mission it serves as the standard-of-excellence that emerging nuclear systems analysis computer codes strive to obtain." Robert P. Martin Ph.D. P.E. BWXT

Apart from nuclear power, RELAP5-3D is used for numerous other applications involving piping flow: from commercial steam flow to cryogenic storage and delivery systems, to flow of coolant through rocket engines, and many more. RELAP5-3D is a truly versatile program.

MOST REQUESTED

With its international RELAP User Group (IRUG), myriad applications, and the team's do or die attitude, RELAP5 has been the most requested code at INL for decades, surpassed for one month in 2023 by BISON, and has taken the lead again.





Except for 1 month, RELAP5-3D has been INL's most requested code for 30+ years.

Chapter 2. Historical Perspective

Soon after the birth of commercial nuclear energy, the Nuclear Regulatory Commission identified a need for reactor safety analysis software. In 1966, Idaho scientists began developing the Reactor Excursion and Leak Analysis Program (RELAP) to model reactor coolant and core behavior in a pressurized water reactor. Much has happened in the 58 years since then.

Section 2.1. The RELAP Series

NOTE: This is excerpted from the Foreword of the January 2016 edition of Nuclear Science and Engineering by Dr. George Mesina.

The earliest forerunner of RELAP5-3D, FLASH-1, used HEM (Homogenous Equilibrium Model) to model the fluid flow with explicit numerics for time advancement. Computer power was severely limited in the 1960s, so FLASH-1 used three control volumes to model the system: the pressurizer volume, which connected to the hot volume, which exchanged fluid with the cold volume (Figure 1).

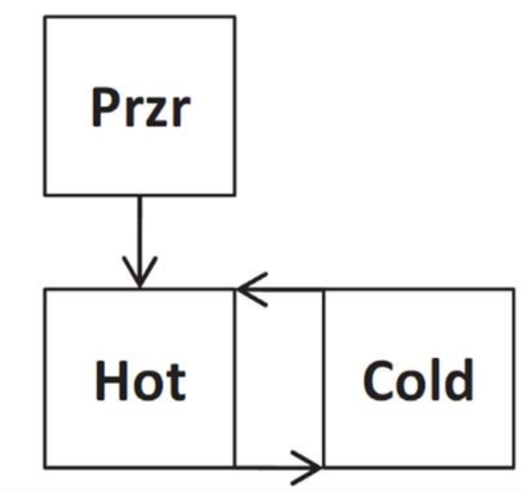


Fig. 1. FLASH-1, 1966.

Doug Hall, Analyst/Project Manager Using RELAP5 in 1980's

The secondary side was modeled as a constant heat transfer coefficient. FLASH-1 had a choked flow model for LOCA analysis. In 1966, INL, known then as the National Reactor Testing Station operated by Phillips Petroleum Company, put the Advanced Test Reactor into operation for testing materials and generating isotopes, and RELAPSE-1 (or RELAP1) was created. A revised version of FLASH-1 for the IBM-7040 and CDC-660 computers, RELAP1 was written in FORTRAN IV to calculate pressures, temperatures, flow, mass inventories, reactivities, and power for PWRs during a reactivity event or LOCA.

RELAP1 represented the primary system as three lumped volumes with pressure-dependent coolant pumps and a flow-dependent heat exchanger (Fig. 2). It also used point kinetics with several reactivity functions, a two-point heat transfer model with three modes, and interpolated steam tables from 1 to 3200 psia. RELAP2 was released in 1968. It used the same three-volume system, same leak/fill capability,

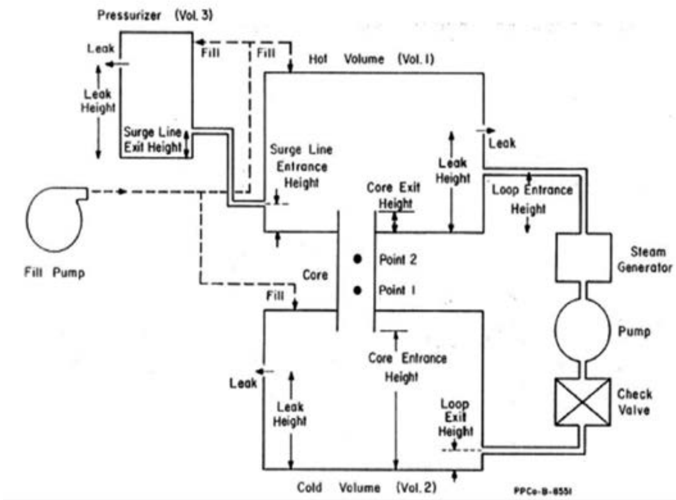


Fig 2. RELAPSE-1 (RELAP1), 1966.

and same heat transfer as RELAP1. It incorporated bubble separation and other models for boiling water reactors (BWRs). Moreover, the program had improved stability, ran twice as fast as RELAP1 on the IBM and CDC computers, and was ported to other machines as well.

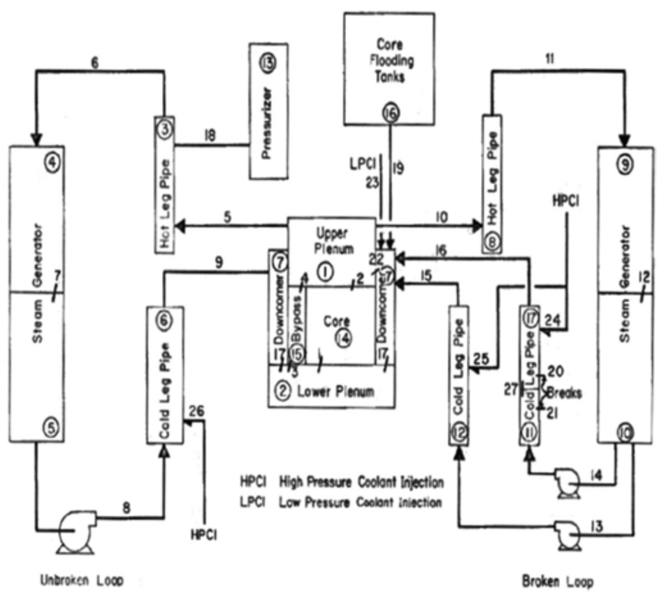


Fig. 3. RELAP3, 1970.

In 1970, RELAP3 evolved from RELAP2. It featured 20 volumes, trip logic, valves, pressure-dependent fill and leak, fuel pins/plates, a heat conduction model, and expanded heat transfer models (Fig. 3).

In 1973, RELAP4/MOD1 was released. This code featured up to 100 control volumes, true one-dimensional (1-D) flow, a momentum flux term dP/dA and form losses, a two-fluid slip model, molecular nitrogen for the accumulator, representation of the secondary system as a flow network, and reflood heat transfer (Fig. 4).

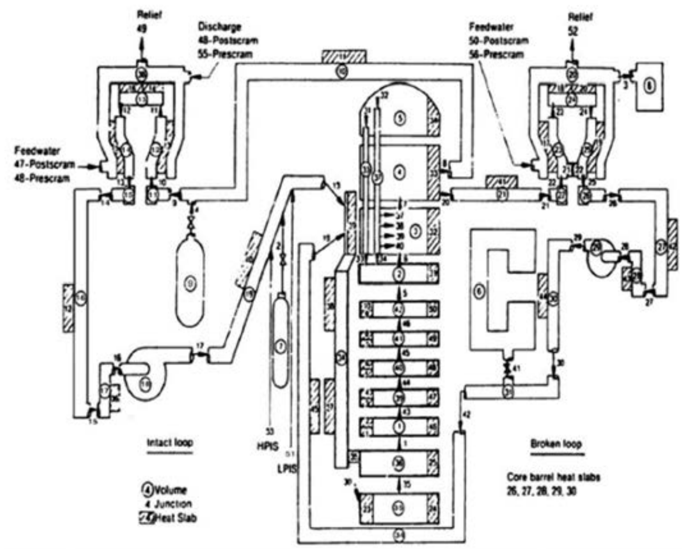
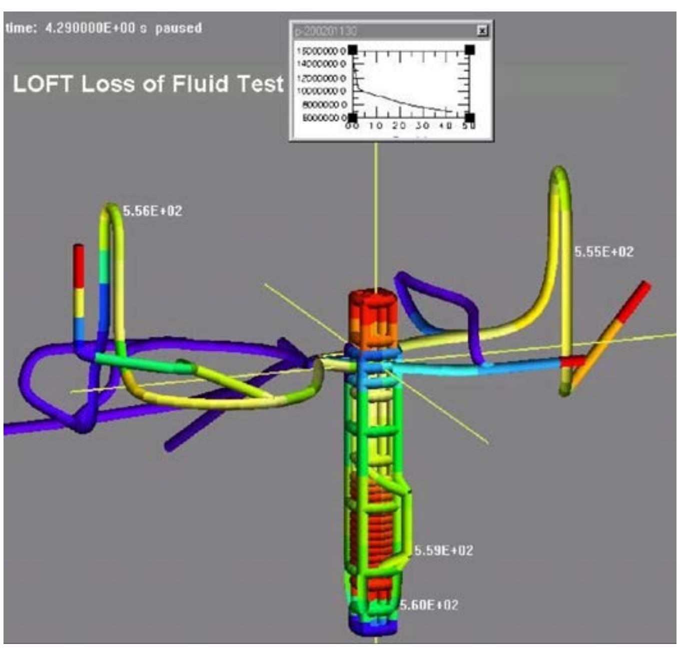


Fig. 4. RELAP4, 1973.

The heat transfer capability was expanded to include modeling of reflood and a fuel gap. The zirconium-steam metal-water reaction was incorporated. Also, implicit numerics were introduced for time advancement. Besides the ability to model SB-LOCAs for PWRs and BWRs, RELAP4/MOD1 could also model large-break LOCA (LB-LOCA) scenarios. Eventually, there were 13 "mods" of RELAP4 written in the ANSI FORTRAN 1966 standard. In 1979, the first line of RELAP5 was written, and a pilot code, with no input processing, was used to demonstrate the new methodology.

The fundamental improvement difference between RELAP5 and all previous versions of RELAP was going from one-fluid HEM to a two-fluid model with a different set of governing equations for the liquid and gas phases. Other improvements were made as well. Interfacial momentum terms and mass and energy exchange terms were added. An important improvement to 1-D fluid flow was the addition of modeling cross-flow. A multichannel fuel rod model was incorporated. Many new models, correlations, trips, and controls were added. A new timestepping scheme was put into place, using semi-implicit numerics for time advancement.

The ability to simulate SB-LOCAs and many operational transients was introduced in RELAP5 at the start with mods 0, 1, and 1.5. These had two mass equations, two momentum equations, but only one energy equation. With mod 2, the second energy equation was introduced. Other capabilities added since mod 2 include nearly implicit time advancement, a counter-current flow limiting (CCFL) model, mixture level, and thermal front tracking. Table 1 lists the various mods (versions) of RELAP5 and the dates they were released. RELAP5 included many programming improvements over RELAP4, including conversion from FORTRAN 66 to FORTRAN 77, improvements to the input processing and error messages, expanded output, memory deallocation through a C-language memory removal procedure, many useful debugging features, vectorization, parallelization, and adaptation to a host of new computer platforms.



RGUI Rendering: LOFT model with 3D Pipe color map of void fraction, user annotation, point and click plot, and display node selection

Section 2.2. Major RELAP Milestones

RELAP5-3D is a best-estimate computer simulation software dedicated to the nuclear power plant operational transient and accident thermal-hydraulics analysis. It is the premier software for reactor safety analysis, reactor design, operator training, and university education.

Early 1960's: The homogeneous equilibrium model (HEM) based fluid flow analysis code FLASH-1 was developed to model loss-of-coolant accident (LOCA) analysis with three control volumes.

1966: RELAP1 (or RELAPSE-1) was a version of FLASH-1 revised by INEL. The code was written in FORTRAN IV to calculate pressures, temperatures, flow, mass inventories, reactivities, and power for PWRs (Pressurized Water Reactors) PWRs during a reactivity event or LOCA.

1968: RELAP2 released. It could resolve BWR system behavior and ran two times faster than RELAP1.

1970: RELAP3 released. It featured a maximum 20 control volumes; components and expanded heat transfer model could be simulated. Built by ANSI FORTRAN 1966.

1973: RELAP4/MOD1 released. It could handle up to 100 control volumes and true one-dimensional (1-D) flow for large-break LOCA (LB-LOCA) scenarios.

1979: RELAP5/MOD0 developed. The main improvement was moving from one-fluid HEM to a two-fluid model with a different set of governing equations for the liquid and gas phases.

1980's: The code was updated to FORTRAN 77. Significant improvement on capability included modeling small-break LOCA and operational transient analysis. This RELAP5 was updated until RELAP5/MOD3.3.

1990-95: RELAP5-3D was developed by INL with support from NRC and the U.S. Department of Energy (DOE). Non-NRC developments are made removable for NRC releases.

1999 Copyright of RELAP5

1996: CRADA to produce a nuclear power plant operator training simulator version of RELAP

1997: INL copyrighted the non-NRC-funded parts of the RELAP code.

1998: The International RELAP5 Users Group (IRUG) was organized to support international RELAP5-3D users and to disseminate the code worldwide.

1999: The copyrighted IRUG code was named RELAP5-3D. Notable features differing from the NRC code version were full three-dimensional hydrodynamics with rectangular, cylindrical, and spherical geometries.

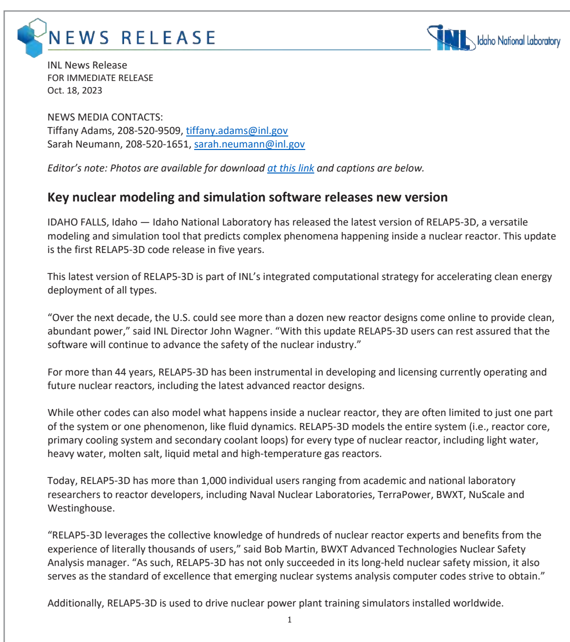
2002: First Windows operating system compatible RELAP5, namely Ver.2.0.3 was released.

2006: RELAP5-3D/Ver. 2.4 was the final FORTRAN 77 code version.

2010: RELAP5-3D/Ver. 3.0 the Beta-test release of the fast-running FORTRAN90 version.

2012: RELAP5-3D/Ver. 4.0.3, the first release of the FORTRAN90 version with comparable performance to version 2.4.

2023: RELAP5-3D/Ver. 4.5.2 is the most recent release and the most robust verified and validated product of the RELAP5 series.



NEWS RELEASE

Idaho National Laboratory

INL News Release
FOR IMMEDIATE RELEASE
Oct. 18, 2023

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Editor's note: Photos are available for download at this link and captions are below.

Key nuclear modeling and simulation software releases new version

IDAHO FALLS, Idaho — Idaho National Laboratory has released the latest version of RELAP5-3D, a versatile modeling and simulation tool that predicts complex phenomena happening inside a nuclear reactor. This update is the first RELAP5-3D code release in five years.

This latest version of RELAP5-3D is part of INL's integrated computational strategy for accelerating clean energy deployment of all types.

"Over the next decade, the U.S. could see more than a dozen new reactor designs come online to provide clean, abundant power," said INL Director John Wagner. "With this update RELAP5-3D users can rest assured that the software will continue to advance the safety of the nuclear industry."

For more than 44 years, RELAP5-3D has been instrumental in developing and licensing currently operating and future nuclear reactors, including the latest advanced reactor designs.

While other codes can also model what happens inside a nuclear reactor, they are often limited to just one part of the system or one phenomenon, like fluid dynamics. RELAP5-3D models the entire system (i.e., reactor core, primary cooling system and secondary coolant loops) for every type of nuclear reactor, including light water, heavy water, molten salt, liquid metal and high-temperature gas reactors.

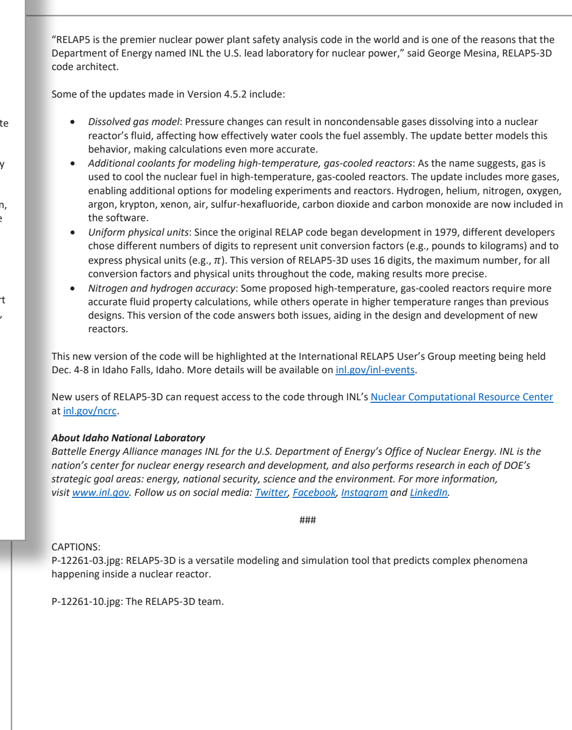
Today, RELAP5-3D has more than 1,000 individual users ranging from academic and national laboratory researchers to reactor developers, including Naval Nuclear Laboratories, TerraPower, BWXT, NuScale and Westinghouse.

"RELAP5-3D leverages the collective knowledge of hundreds of nuclear reactor experts and benefits from the experience of literally thousands of users," said Bob Martin, BWXT Advanced Technologies Nuclear Safety Analysis manager. "As such, RELAP5-3D has not only succeeded in its long-held nuclear safety mission, it also serves as the standard of excellence that emerging nuclear systems analysis computer codes strive to obtain."

Additionally, RELAP5-3D is used to drive nuclear power plant training simulators installed worldwide.

1

INL News Release for RELAP5-3D Version 4.5.2



"RELAP5 is the premier nuclear power plant safety analysis code in the world and is one of the reasons that the Department of Energy named INL the U.S. lead laboratory for nuclear power," said George Mesina, RELAP5-3D code architect.

Some of the updates made in Version 4.5.2 include:

- *Dissolved gas model*: Pressure changes can result in noncondensable gases dissolving into a nuclear reactor's fluid, affecting how effectively water cools the fuel assembly. The update better models this behavior, making calculations even more accurate.
- *Additional coolants for modeling high-temperature, gas-cooled reactors*: As the name suggests, gas is used to cool the nuclear fuel in high-temperature, gas-cooled reactors. The update includes more gases, enabling additional options for modeling experiments and reactors. Hydrogen, helium, nitrogen, oxygen, argon, krypton, xenon, air, sulfur-hexafluoride, carbon dioxide and carbon monoxide are now included in the software.
- *Uniform physical units*: Since the original RELAP code began development in 1979, different developers chose different numbers of digits to represent unit conversion factors (e.g., pounds to kilograms) and to express physical units (e.g., π). This version of RELAP5-3D uses 16 digits, the maximum number, for all conversion factors and physical units throughout the code, making results more precise.
- *Nitrogen and hydrogen accuracy*: Some proposed high-temperature, gas-cooled reactors require more accurate fluid property calculations, while others operate in higher temperature ranges than previous designs. This version of the code answers both issues, aiding in the design and development of new reactors.

This new version of the code will be highlighted at the International RELAP5 User's Group meeting being held Dec. 4-8 in Idaho Falls, Idaho. More details will be available on inl.gov/inl-events.

New users of RELAP5-3D can request access to the code through INL's Nuclear Computational Resource Center at inl.gov/ncrc.

About Idaho National Laboratory
Battelle Energy Alliance manages INL for the U.S. Department of Energy's Office of Nuclear Energy. INL is the nation's center for nuclear energy research and development, and also performs research in each of DOE's strategic goal areas: energy, national security, science and the environment. For more information, visit www.inl.gov. Follow us on social media: Twitter, Facebook, Instagram and LinkedIn.

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CAPTIONS:
P-12261-03.jpg: RELAP5-3D is a versatile modeling and simulation tool that predicts complex phenomena happening inside a nuclear reactor.

P-12261-10.jpg: The RELAP5-3D team.

Chapter 2.3. Detailed History of RELAP5

2.3.1. Subcodes

The base code was named RELAP5 (Reactor Excursion and Leak Analysis Program) and was number 5 in the series of NRC (Nuclear Regulatory Commission) funded RELAP codes that were originally intended to model only LBLOCAs (Large Break Loss of Coolant Accidents) in PWRs (Pressurized Water Reactors). A great deal of coding was funded by other governmental agencies to develop capabilities needed for their programs. These coding additions were protected by pre-compiler directives and could be removed at any time by a pre-compiler, leaving a freestanding and fully operational RELAP5 code; or some of them could be left in to create a RELAP5 based code with the desired extra features. These variations of RELAP5 had names.

- SCDAP (Severe Core Damage Analysis Program) models the progression of light water reactor core damage including core heatup, core disruption, debris formation, debris heatup, and debris melting.
- ATHENA (A Thermal Hydraulic Energy Network Analyzer) incorporated coding that was in part developed for the DOE fusion program. It included the third fluid field, extra fluids, and magnetohydrodynamics.
- BETTIS-designated coding was funded by the DOE Office of Naval Reactors and included the 3D component, variable gravity, level tracking, and much more.
- SIMULATOR coding was funded by a DOE CRADA between SAIC and INEL to build a real-time version of RELAP5 for use in operator training simulators.

2.3.2. Code Designations and Releases

At various times the code was released to clients after a number of code developments were added. Some of these were code corrections in answer to UTRs (user trouble reports). Between public releases, periodically (often monthly, sometimes weekly) the code was released internally and tested by the analysts and developers. There were two designations for RELAP5 code versions.

- The older method, called "alphabet soup" by the team, used a base name appended with a number or letter designation and each of those was the next letter in the alphabet sequentially. When z was reached, double letters were used, so the sequence went 0, 1, ..., 9, a, b, ... y, z, aa, ab, ac, and so forth. The external releases used the same name as the internal release.
- The number-based method used a base name appended with a 3- to 4-digit number with each digit separated by a period. The first number indicates a major change, such as conversion to Fortan 77 or Fortran 90. The second number indicates the external release to the public. The third number indicates the internal release. For code patches for user bug fixes, an external release would get a fourth digit for the patch number.
- Frequently with the MOD numbers, only the major change and external release numbers are given with the number method. Also, sometimes the periods were suppressed.
- Base names changed several times
  - RELAP5 for the base code alone without SCDAP coupled
  - SCDAP/RELAP5 for the base code coupled with SCDAP
  - ATHENA (A Thermal Hydraulic Energy Network Analyzer)
  - "selp" for Scdap/RELaP
  - "rbc" for Relap5-Based-Code (which included the NPA)
    - NPA is the Nuclear Plant Analyzer, the first RELAP5 graphical user interface
  - "rlpsim" ReLaP SIMulator
  - "rlpdoe" ReLaP Department Of Energy

Number Designation	Release Date	Alphabet Soup
RELAP5/MOD3.0	March 1990	5m5
selp70	February 1991	selp70
selp80	March 1992	selp80
RELAP5/MOD3.1	March 1993	selp8ko
selp8r-BETTIS	May 1993	selp8r
RELAP5/MOD3.1.1	November 1993	rbc3.1.1
rbc3.1.8ba-BETTIS	November 1993	rbc3.1.8ba
SCDAP/RELAP5/MOD3.1A	March 1994	rbc3.1.8ba
SCDAP/RELAP5/MOD3.1B	April 1995	rbc3.1.8bb
SCDAP/RELAP5/MOD3.1C	May 1995	rbc3.1.8bc
SCDAP/RELAP5/MOD3.1D	June 1995	rbc3.1.8bd
RELAP5/MOD3.1.1.1	April 1994	rbc3.1.1.1
rbc3.1.8bm-BETTIS	May 1994	rbc3.1.8bm
RELAP5/MOD3.1.1.2	June 1994	rbc3.1.8br
RELAP5/MOD3.1.2	August 1994	rbc3.1.2
RELAP5/MOD3.1.2.1	November 1994	rbc3.1.2.1
RELAP5/MOD3.1.2.2	December 1994	rbc3.1.2.2
RELAP5/MOD3.1.3	March 1995	rbc3.1.3
ATHENA/MOD3.1	May 1995	rbc3.1.3
RELAP5/MOD3.1.4	May 1995	rbc3.1.4
rbc3.dk-BETTIS	August 1995	rbc3.2dk
RELAP5/MOD3.2.1	August 1995	rbc3.2dk
RELAP5/MOD3.2	October 1995	rbc3.dj2
RELAP5/MOD3.2	October 1995	rbc3.2ea

Number Designation	Release Date	Alphabet Soup
RELAP5/MOD3.2.1	April 1996	rbc3.2en
RELAP5/MOD3.2.1.1	April 1996	rbc3.2eo
RELAP5/MOD3.2.1.2	May 1996	rbc3.2ep
rlpsimaw1	February 1997	rlpsimaw1-Bettis
rlpdoebf-BETTIS	July 1997	rlpdoebf
RELAP5-3D 1.0.0	July 1997	rlpdoebf
RELAP5-3D 1.0.05	September 19, 1997	rlpdoebf05
RELAP5-3D 1.0.08	September 24, 1998	rlpdoebf08
RELAP5-3D 1.1.0	November 23, 1998	rlpdoebg
RELAP5-3D 1.1.7	August 4, 1999	rlpdoebn
RELAP5-3D 1.1.72	October 28, 1999	rlpdoebn2
RELAP5-3D 1.2.0	May 5, 2000	Alphabet discontinued
RELAP5-3D 1.2.2	June 26, 2000	Alphabet discontinued
RELAP5-3D 1.3.5	March 14, 2001	Alphabet discontinued
RELAP5-3D 2.0.3	August 21, 2002	Alphabet discontinued
RELAP5-3D 2.2	October 30, 2003	Alphabet discontinued
RELAP5-3D 2.4	October 5, 2006	Alphabet discontinued
RELAP5-3D 3.0.0	November 29, 2010	Alphabet discontinued
RELAP5-3D 4.0.3	July 12, 2012	Alphabet discontinued
RELAP5-3D 4.1.3	October 8, 2013	Alphabet discontinued
RELAP5-3D 4.2.1	June 30, 2014	Alphabet discontinued
RELAP5-3D 4.3.4	October 9, 2015	Alphabet discontinued
RELAP5-3D 4.4.2	June 25, 2018	Alphabet discontinued
RELAP5-3D 4.5.2	October 18, 2023	Alphabet discontinued

Chapter 3. User Group Meetings

3.1. History of International RELAP Meetings

International RELAP meetings and International RELAP5 User Group (IRUG) meetings have been held annually since 1990, with the exception of only eight years: 1992, 1994, 1995, 1997, 2001, 2017, 2020, and 2022. Technical topics discussed at each meeting are listed.

1990 Joint RELAP5-TRAC-BWR International User Seminar

September 17 – 21, Chicago IL

- Grid Heat Transfer Models for RELAP5/Mod2
- Modelling the Onset of Flow Instability for Subcooled Boiling in Downflow
- Stratified Flow Benchmark Calculations Using RELAP5/Mod2
- Small Break LOCA Analysis for the High Flux Isotope Reactor
- Assessment of RELAP5/Mod 2 Against Integral Start-up Tests in the Vandellos 2 Power Plant
- Benchmarks of 10 cm2 Scaled Cold Leg Breaks on the MIST Facility
- Assessment of RELAP5/MOD2 Using Semiscale Intermediate Break Loss-of-Coolant Experiment S-IB-3
- RELAP5/MOD2 and Loop Seal Clearing During a SBLOCA
- RELAP5/MOD2 Studies of a Steam Line Break Fault in the Sizewell B PWR Including the Modeling of Mixing within the Reactor Pressure Vessel
- Major Rupture of a Main Feedwater Pipe
- Results of Benchmark Experiments for RELAP Calculations of Boiling in Narrow Channels with No Circulation Loop
- Best-Estimate Analysis of Station-Blackout for a PWR Plant
- Steady-State Simulations of 30-Tube Once-Through Steam Generator on RELAP5/MOD3 and RELAP5/MOD2 Computer Codes
- RELAP5/MOD3 New Models
- RELAP5/MOD3 Developmental Assessment
- Simulating Stead-State for a Once-Through Steam Generator with RELAP5/MOD3
- Simulation of UPTF Test 11 with RELAP5/MOD2 and RELAP5/MOD3
- An Analysis of RELAP5/MOD3 Predictions of a Small Break LOCA Conducted at the ROSA-IV Large Scale Test Facility
- RELAP5/MOD3 Simulation of IAEA Standard Exercise 3
- Impact of Computer Operating System on RELAP5 Calculational Results
- RELAP5 Modeling of a Savannah River Reactor
- Air-Water Hydraulics Modeling for a Mark-22 Fuel Assembly with RELAP5
- Air Ingestion into the External Loops of a Savannah River Site Reactor during a Postulated LOCA
- Benchmarking the RELAP5 r-0 Model of an SRS Reactor to the 1989 L-Reactor Tests
- Modeling Advanced Neutron Source Reactor Station Blackout Accident Using RELAP5
- RELAP5/MOD3 Assessment Using BETHSY Natural Circulation Data
- Applicability of RELAP5 for AP600 and PIUS Safety Analysis
- Savannah River Site Reactor Hardware Design Modification Study

- Post Test Calculation of PKL-II-B4 with RELAP5/MOD3
- Prioritization and funding for model improvements
- Improving User Support Services
- User guidelines
- Two-phase capabilities
- LOCA applications
- Use of RELAP5 in the PC environment
- Difference between RELAP5/TRAC results
- RELAP5 use for BWR modeling application and analysis
- Fluid temperature spikes (interphase heat transfer)
- Nuclear plant analyzer
- Code portability
- "An interactive Preprocessor for RELAP5"
- "Desk-top RELAP5 Simulator for Digital Feedwater Control System Testing"
- Running RELAP5/MOD3 on a Desktop Workstation
- The Use of the IsoVu Graphics Display Systems for Post-Processing RELAP5 Calculations
- Demonstration of the Nuclear Plant Analyzer using RELAP5
- SIMONE – An Interactive Environment for Running RELAP5

1991 RELAP5/TRAC-B International Users Seminar

November 4 – 8, Baton Route LA

- Addition of Three-Dimensional Modelling
- Assessment of RELAP5/MOD3 CHF Model
- Investigation in Interphase Heat and Mass Transfer
- An Assessment of Heat Transfer Packages Used in TRAC-PF1 (Version 3.9 and 5.3) and TRAC-BF1
- Development and Testing of a 3D Nodal Kinetics Algorithm for TRAC/BF1
- Modifications to the TRAC=BF1 Containment Calculation for Small Break Blowdowns
- Modeling Condensation Steam Quenching Effects in a MSIV Closure Transient: A Comparison of Predictions by RETRAN-02 With and Without Nonequilibrium Pressurizer Model Option to Predictions by TRAC-BR1
- BWR Core Thermal-=Hydraulic Analysis with TRAC
- Nuclenor's Experience using TRAC=BR1 in Operational Transients Analysis: An Assessment of a MSV's Closure at Full Power
- A MSIV Closure Transient Best Estimate Analysis Using TRAC-BF1
- BWR6 Hot Channel Analysis Using RELAP5
- Assessment of RELAP5 Critical Flow Models Against Separate Effects Data
- RELAP5/MOD3 Assessments of FLECHT SEASET Unblocked Forced Reflood Tests
- Modelling of Counter-Current Flow Limit for ATR Side Plate Using RELAP5/MOD3
- Assessment of RELAP5/MOD3/V5M5 Against the UPTF Test No. 11 (Countercurrent Flow in PWR Hot Leg)
- Prediction of Flow Rates Under Counter-Current Flow Limitation (CCFL) and Void Fractions Using RELAP5/MOD3
- RELAP5 Multi-Dimensional Constitutive Models
- Feedwater Line Thermal Hydraulic Behavior During LOCA Conditions for Quad-Cities Nuclear Power Station
- Timing Analysis of PWR Fuel Pin Failure under LOCA Conditions
- Post Test Calculation of PKL II B5 with RELAP5/MOD3
- RELAP5/MOD3 Simulation of Natural Circulation Inside the MSVV During the Event of MSLB
- Analysis of a Stuck Open PORV in the Yankee Plant using RELAP5/MOD3 Computer Code
- Modeling Operator Actions During a Small Break Loss-of-Coolant Accident in a Typical Babcock and Wilcox Nuclear Power Plant

- RELAP5/MOD2 Analysis of a Steam Generator Tube Rupture Event with Maximum HHSI and Loss-of-Offsite Power
- Post-Test Calculation of BETHSY Test 6.2 TC Using RELAP5/MOD3
- RELAP5 Modeling RBMK Reactor Applications
- RELAP5/MOD3 Assessment Using Semiscale 50% Feedline Break Test S-SB-11
- River Bend Station RELAP5
- Reactor Water Clean-up System Transient Using RELAP5/MOD2
- RELAP5/MOD2 Post-Test Calculation
- RELAP5/MOD3 PWR NPP Startup Tests Modeling
- Air-Water Hydraulics Modeling for a Mark-22 Fuel Assembly with RELAP5 Part II (U)
- Inadvertent Rotovalve Closure Accident Analysis of the Savannah River K-Reactor
- An Analysis of SBLOCA in the Septifoil System of a Savannah River Reactor
- RELAP5/MOD3 Benchmarks of Low Pressure Subcooled Vapor Voiding in an Annular Flow Channel
- RELAP5 Development and Applications at Savannah River Site (U)
- SCDAP/RELAP5 Assessment
- Experience in the Assessment of SCDAP/RELAP5 Computer Code
- Application of RELAP5/MOD3 to Analyze External Cooling of Reactor Pressure During a Core Melt Accident
- PaRaELAP: Realtime Multiprocessing RELAP5
- Application of RELAP5 Methodology in Real Time
- Outage Assessment Using RELAP5
- RELAP5/MOD3 Simulation of the Water Cannon Phenomenon
- Installation
- Tornado Depressurization Analysis Using RELAP5/MOD3
- A Mechanistic Evaluation of the LOCA Success Criteria in PRA

1993 RELAP5 International Users Seminar

July 6 – 9, Boston MA

- Posttest Analysis of the UPTF Test #26 with RELAP5/MOD2/V251
- Experiences and Results of Pre-Test and Post-Test Calculations Based on PKL III Experiments using RELAP5/MOD2
- Simulation of OECD LOFT LP-01-6 with RELAP5/MOD3
- Mod 3 Assessment: Approach Strategy and Execution
- Development of Parallel Computational Technique Based on Multiple Processes and its Application of Large Thermal Hydraulic Computer Programs
- Installing and Benchmarking RELAP5/MOD3 Version 8k on Various Computers and Workstations
- An Integrated Spreadsheet-Based Input Model Tool for RELAP5
- Pressure Waves in a Feedwater Line of a PWR Induced by Condensing of Steam in a Feedwater Preheater
- RELAP5/MOD2 Analysis of BETHSY Test 7.2C: Reflexer Condenser Mode Steam Generator Operation with Non-Condensible Gas in the Primary Side
- RELAP5/MOD3 Modelling of Water Slug Propelled by Non-Condensible Gas
- Simulation of Stuck Fuel Bundles in the Inclined Fuel Transfer System Tube Using RELAP5/MOD3
- RELAP5 Analysis of the Reactor Fessel Head Vent System
- Application of RELAP5 for BWR/4 Transients Analysis
- RELAP5/MOD3 Simulation of the Loss of Residual Heat Removal During Midloop Operation Experiment Conducted at the ROSA-IV Large Scale Test Facility

- Large Primary-to-Secondary Leakage Accident in LOVIISA VVER-440 Reactor
- Assessment of RELAP5 and Verification of Modelling Methods of VVER-Type Reactor Analysis
- Reassessment of the Safety of VVER-440/213-Type NPPs: Application of RELAP5/MOD2 for DEB Analyses
- Application of RELAP5/MOD2 and MOD3 Codes to Isolation Condense Related Experiments
- RELAP5 Code Assessment for PWR NPP Complete Loss of AC Power at Low Temperature
- RELAP5/MOD3 Accident Analysis for River Bend Station
- Primary Feed & Bleed Thermal Hydraulic Analysis for ASCO Nuclear Power Plan
- RELAP5/MOD3 AP600 Problems
- RELAP5/MOD3 Code Coupling Feature
- RELAP5 Modifications to Model Sudden Area Changes with Improved Total Energy Conservation
- Incorporation of Droplet Interfacial Area in RELAP5/MOD3
- SCDAP/RELAP5 Severe Accident Capabilities: A UK View
- The Implications of Design and Modeling Differences Between MAAP MELCOR and SCDAP/RELAP5 for Severe Accident Analysis of Nuclear Power Plants
- Extensions to SCDAP/RELAP5 Code for the Modeling of Thermal-Hydraulic Behavior in Porous Media
- Benchmarking Simulation System Codes with Best Estimate System Codes
- RELAP5/MOD3 Analysis of SRS Supplemental Safety System (Gas Pressurizer) Tests
- Analysis of FLECHT SEASET Reflood Tests with RELAP5/MOD3
- A Numerical Study of the Three-Dimensional Hydrodynamic Component in RELAP5/MOD3
- Application of Godunov Method for Boron Tracking in RELAP5/MOD3
- Stability and Convergence of Numerical Methods in RELAP5
- Smoothing Transitions Between Constitutive Relations in RELAP5/MOD3
- RELAP5/MOD3 Analysis of a Heated Channel in Downflow
- CCFL Model Verification on OECD ISP-33 Pactel
- Assessment of RELAP5 Via the Analysis of Selected BETHSY Tests
- A Comparison of Boiling Correlations in ATHENA to Nitrogen Data
- The Analysis of Thrust Loads Acting on a RPV After Lower Head Failure at High System Pressure in a Siemens PWR
- Modelling of a Horizontal Steam Generator for the Submerged Nuclear Power Station Concept
- On RELAP5 Simulated High Flux Isotope Reactor Reactivity Transients: Code Change and Application
- Pressurization Simulation of LCC Ductwork During the Event of MSLB
- Development and Developmental Assessment of RELAP5/MOD3 (Ver80 and 8k)
- RELAP5/MOD2.5 Simulation Results for the Separate Effects Test Experiments: Phase 1
- Simulation of Marviken Critical Flow Test 21 and Jet Impingement Test with RELAP5/MOD3

1996 OECD/CSNI Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements

November 5 - 8, Annapolis, MD

- Capabilities of Current Generation of Thermal-Hydraulic Codes and Future Plans
- Methodology, Status and Plans or Development and Assessment of RELAP5 Code
- Methodology, Status and Plans or Development and Assessment of TRAC Code

- Methodology, Status and Plans or Development and Assessment of CATHARE Code
- Methodology, Status and Plans or Development and Assessment of TU and CATHENA Codes
- Methodology, Status and Plans or Development and Assessment of ATHLET Code
- Methodology, Status and Plans or Development and Assessment of HEXTRAN, TRAB and APROS
- Thermal Hydraulic Modeling Needs or Passive Reactor
- The Role of Uncertainty in Code Development
- The Role of the PIRT Process in Identifying Code Improvements and Executing Code Development
- Current and Anticipated Uses of the Thermal-Hydraulic Codes at the NRC
- Current and Anticipated Uses of the CATHARE Code at EDF and FRAMATOME
- Current and Anticipated Uses of the Thermal-Hydraulic Codes in Germany
- The Italian Experience on T/H Best-Estimate Codes: Achievements and Perspectives
- Current and Anticipated Uses of Thermal-Hydraulic Codes at the Japan Atomic Energy Research Institute (JAERI)
- Current and Anticipated Uses of Thermal-Hydraulic Codes in NFI
- Current and Anticipated Uses of Thermal-Hydraulic Codes for BWR Transient and Accident Analyses in Japan
- Current and Anticipated Uses of Thermal-Hydraulic Codes in Korea
- Current and Anticipated Uses of Thermal-Hydraulic Codes in Spain
- Current and Anticipated Uses of Thermal-Hydraulic and Neutronic Codes at PSI
- Capabilities Needed for the Next Generation of Thermo-Hydraulic Codes for Use in Real-Time Applications
- Thermal Hydraulic-Severe Accident Code Interfaces for SCDAP/RELAP5/MOD3.2
- Interface Requirements to Couple Thermal-Hydraulics Codes to Severe Accident Codes: ICARE/CATHARE
- Interface Requirements to Couple Thermal-Hydraulics Codes to Severe Accident Codes: ATHLET-CD
- Interface Requirements to Couple Thermal-Hydraulics Codes to 3D Neutronics Codes
- Development o an Integrated Thermal-Hydraulic Capability Incorporating the PANTHER Neutronic Codes
- An Analytical Study on Excitation of Nuclear-Coupled Thermal-Hydraulic Instability Due to Seismically Induced Reasonance in BWR
- Interface Requirements for Coupling a Containment Code to a Reactor System Thermal-Hydraulic Code
- Thermal-Hydraulic Interfacing Code Modules for CANDU Reactors
- Status of Thermalhydraulic Modelling and Assessment: Open Issues
- Dividing Phases in 2-Phase Flows and Modeling of Interfacial Drag
- Advances in Modelling of Condensation Phenomena
- Multi-Dimensional Reactor Kinetics Modeling
- 3D Neutronics Codes Coupled with Thermal-Hydraulic System Codes for PWR, BWR AND VVER Reactors
- Perspectives on Multifield Models
- Problems with Numerical Techniques: Application to Mid-Loop operations Transient
- Elimination of Numerical Diffusion in 1-Phase and 2-Phase Flows
- Advanced Numerical Methods for Three-Dimensional Two-Phase Flow Calculations

- Recent Advances in Two-Phase Flow Numerics
- Current and Planned Numerical Development for Improving Computing Performance or Long Duration and/or Low Pressure Transients
- Current Implementation and Future Plans on New Code Architecture, Programming Language and User Interfaces
- Parallelization and Automatic Data Distribution for Nuclear Reactor Simulations
- TOOKUIL: A Case Study in User Interfaced Development or Safety Code Application
- Requirements or a Multifunctional Code Architecture
- Development of the Simulation System "IMPACT" or Analysis of Nuclear Power Plant Severe Accidents
- Thermal-Hydraulic Codes for LWR Safety Analysis – Present Status and Future Perspective
- Views of the Future of Thermal-Hydraulic Modeling
- Advances and Needs on Thermal-Hydraulic Modeling
- Numerical Techniques and Coupling Interface Requirements
- User Needs and User Interfaces

1996 RELAP5 International Users Seminar

March 17 – 21, Dallas, TX

- RELAP5/MOD3.1.1 Dynamic Stability Simulations and Code Improvements
- A Simplified Overall Heat Transfer Coefficient Method for Verifying Reactor Power Estimates in a Pressurized Water Reactor
- RELAP5 Dynamic Valve Model and Validation
- A Direct Steady-State Initialization Method for RELAP5
- Overview on the Results of the RCA Project on Core Degradation
- SCDAP/RELAP5 Assessment of Reflood Oxidation Models
- Small Break LOCA Analyses as Basis of Severe Accident Cases for a 3-Loop PWR Using RELAP5/MOD3.1
- An Assessment of SCDAP/RELAP5/MOD3.1 Using Representative Plant Calculations
- Application of SCDAP/RELAP5 for Bundle Experiments
- RELAP5/MOD3 Horizontal Interphase Drag Discontinuity
- A BWR Upper Plenum Model for Core Spray and CCFL Breakdown
- Analysis of Numerical Flow Anomalies in RELAP5
- Application of the CSAU Methodology to BETHSY SBLOCA Test 9.1B Using RELAP5/MOD3.1
- RELAP5 Nodalization Study for BETHSY Experiment
- Simulation of Waterhammer Type Hydraulic Transients with RELAP/MOD3.1
- Calculation of the Hydrodynamic Forces in the Pressurizer Safety Valve Pipeline of LOVIISA NPP Using RELAP5/MOD3.1 Code
- Validation of RELAP5/MOD3 Hydrodynamic Pipe Loading Post-Processor R5Force
- Developing the Code-Systems in the Virtual Space of RELAP5 Part 1: Need, Purpose and Philosophy of General Coupling the Codes
- Developing the Code-Systems in the Virtual Space of RELAP5 Part 2: Interaction with Kinetics Core Model in a BWR ATWS with Boron Injection
- Using RELAP5/MOD3.2 on a Personal Computer Using OS/ Warp
- Experiences Gained in Use of RELAP5 Input Model of VVER-440/213 Reactor
- RELAP5 Applications to RBMK-1500 Safety Evaluation
- RELAP5 Beyond the Year 2000
- The RELAP5 Thermal-Hydraulics Model Upgrade at the Comanche Peak Training Simulator
- R5/V3.2 R/T 3-D Real Time RELAP Simulator Application

- Kinetics Benchmark Problem
- Improved Level Tracking in RELAP5
- RELAP5 Graphical User Interface
- Transient Simulation of Feedwater Vaporization During a DBA LOP/LOCA Using RELAP5/MOD3.1
- Main Steam Line Break Simulation of SBWR with RELAP5/MOD3
- The Realistic Analysis of the LBLOCA for Maanshan Nuclear Power Plant by Using RELAP5AY
- Licensing FSAR Transients with a Best Estimate Code: The Belgatom Approach
- Evaluation of BWR Water Level Backfill Modification using RELAP5
- Simulation of Berkeley Noncondensible Experiment with RELAP5
- Simulation of Condensation in Presence of Noncondensible with CATHARE

1998 RELAP5 International Users Seminar

May 17 – 21, College Station, TX

- RELAP5-3D
- RELAP5-3D Kinetics
- RELAP5-3D Graphical User Interface
- Simulation of the Stop Flow Rate at the RBMK Fuel Channel Inlet When Removing the Residual Heat
- Calculation to Support Increase in Operating Power for Natural Circulation TRIGA Reactor
- Impact of the Initiation of Secondary System Cooling Actions in a SBLOCA Scenario for the Asco NPP
- Annular Critical Heat Flux Determination for Russian ADE Reactors
- Advancements at DCMN of University of Pisa Finalized to the Use of RELAP5/MOD3.2 Code
- Development of Computer Code Models for the Penn State Thermal-Hydraulic Test Loop Facility
- RELAP5 Training Activities in Support of the International Nuclear Safety Program
- RELAP5 User Problems
- Analysis of Experimental Data on DNB and Post-DNB Heat Transfer by RELAP5/MOD3.2 Code
- Code Validation/Assessment RELAP5/MOD3.2 Simulation of the Experiment Conducted at the IIST Facility: Loss of RHR During Mid-Loop Operation with Pressure Vessel Venting
- Assessment of RELAP5/MOD3.2 on Experimental Data of Water Level Behavior in VVER-440 Steam Generator by Transient Regimes with Six MCP Tripping and Stop Feedwater Supply on KOLA NPP
- Post-Test Calculations of the ISB-VVER Small Break LOCA Experiment Using RELAP5/MOD3.2
- ATHENA Heat Structure Generalization
- Simulation of ITER Plasma Facing Components with ATHENA
- Coupling MATLAB and RELAP5 for Prototyping Novel Applications with RELAP5
- Coupling of the Neutronics Code PANTHER with RELAP5 The BELGATOM Approach to Simulate the core Response of Non-Symmetric Reactivity Transient
- RELAP5/MOD3.2 – NPA for Personal Computer
- Penn States Experience with Coupling RELAP5 and CONTAIN Using PVM Technology
- Bernoulli Corrected Upwind Differencing Scheme
- New RELAP5-3D Linear Equation Solver
- Assessment of RELAP5/MOD3.2 Heat and Mass Transfer Models for Large Volume with Horizontal Tube Bundles Against Russian Experimental Data
- Assessment of RELAP5-3D Using Data from Two-Dimensional RPI Flow Tests
- Assessment of RELAP5/MOD3 Against FLECHT-SEASET Steam Generator Separate Effects Test
- Code Validation for RELAP5/MOD3.2 for the RBMK Reactor Design Using SEL and HTL Experimental Data
- Assessment of the RELAP5 Multi-Dimensional Component Model Using Data from LOFT Test L2-5

- The Post-Test Calculation of 2.4% Break LOCA Test at the Integral Test Facility ISB-WWER Using the Thermal Hydraulic Code RELAP5
- Flow Stagnation in the Upper Portion of "BETHSY" Pressure Vessel
- Fluid Transient Analysis of Torus Vent System Piping Due to Rupture Disk Rupture
- RELAP5/MOD3 Capability to Simulate the Performance of Isolation Condenser Systems
- Simulation of HIFR Research Reactor with RELAP5/MOD3.2
- Study of Operational Transient in WWER-1000 Plant Using RELAP5/MOD3.2
- International RELAP5 Users Group Discussion
- Graphical User Interface

1999 International RELAP5 User Group Meeting

July 28 – 30, Park City, UT

- RELAP5-3D Development Status
- IRUG Program
- INEEL Activities in Support of INSP
- An Assessment of the RELAP-3D Using the Edwards-O'Brien Blowdown Problem
- Assessment for Russian-Designated Reactors
- RELAP5/MOD3.2 Validation Studies Based on Data of "Kurchatov Institue" Experiments on Coolant Flow Stop Under Decay Heat Conditions in RBMK Fuel Channel Model
- RELAP5 Code Validation for RBMK Reactors – KS Facility
- Annular Geometry Critical Heat Flux Correlation Generation
- Utilization of RELAP5 Code for Safety Analyses of VVER Reactors
- Comparative Assessment of VVER Standard Problem Number 1: 2.4% Coolant Leak from Downcomer with ECCS Water Supply to the Hot Leg of the Intact Loop in the ISB-VVER Test Facility
- Assessment of the RELAP5-3D Code Using UPTF Test 6, Run 131
- Development of a Code with Capability of Internal Assessment of Uncertainty
- A New Code Uncertainty Methodology
- Assessment of the RELAP5-3D Code Using UPTF Test 6, Run 131
- Development of a Code with Capability of Internal Assessment of Uncertainty
- A New Code Uncertainty Methodology
- Interfacing an External Subroutine with RELAP5-3D Kinetics
- Consideration of Non-Linearities in the Momentum Equations
- Development o LOCA Licensing Calculation Capability with RELAP5-3D in Accordance with Appendix K of 10CFR50.46
- A Perspective on the Future of Nuclear Power
- Modifications to the Critical Heat lux Correlations of RELAP5/MOD3.2 for Annular Fuel Elements
- RELAP5 User Problems
- RGUI-RELAP5-3D Graphical User Interface
- RGUI-Hands On

2000 International RELAP5 User Group Meeting

September 12 – 14, Jackson Hole WY

- Effect of Nodalization on the Accuracy of the Finite Difference Solution of the Transient Conduction Equation
- A Coupled RELAP5-3D/CFD Methodology with a Proof Principal Calculation
- Verification of Calculation Technique Critical Heat Flux in RELAP5/Mod 3.2 Code Applied to the MIR Research Reactor on Experimental Data Obtained for the Annulus at Average Water Pressures

- RELAP5/Mod3.2 Post Test Simulation and Accuracy Quantification of LOBI Test A1-93
- Simulation of a Control Rod Ejection in a VVER-1000 Reactor with the Program RELAP5-3D
- Assessment of the ATHENA Code for Calculation the Void Fraction of a Lead-Bismuth/Steam Mixture in Vertical Upflow
- RELAP5 Analyses of Heated Vacuum Drying System Pressurization
- oRELAP5-3D Model for the Kursk1 NPP
- Assessment of the RELAP5-3D Subcooled Boiling Models for Low Pressure Conditions
- RELAP5-3D Development Status
- Development of LOCA Licensing Calculation Capability with RELAP5-3D in Accordance with Appendix K of 10CFR50
- Application of ATHENA/RELAP to Fusion Loss of Cooling Accidents
- New Developments and Value of the RELAP5-3D Graphical User Interface (RGUI 1.2)
- RELAP5-3D User Problems
- Using RELAP5-3D to Design a Small Factory-Built Generation 3 Reactor
- Modeling the Seattle Steam Co. Distribution System with RELAP5-3D
- Improved Solution of Field Equations
- Flux Limited Upwind Differenced Scheme in RELAP5-3D
- New Assessment of RELAP5-3D Using a General Electric Level Swell Problem
- A Generic Semi-Implicit Coupling Methodology for Use in RELAP5-3D
- INEEL Activities in Support of INSP (International Nuclear Safety Program)

2000 International RELAP5 User Group Meeting

September 5 – 7, Sun Valley ID

- An Integrated RELAP5-3D and Multiphase CFD Code System Utilizing a Semi-Implicit Coupling Technique
- INSP IRUG-related Activities in FY-2001
- The Development and Application of SCDAP-3D
- Pygmalion Users Manual
- Performance and Safety Studies for Multi-Application Small Light Water Reactor (MASLWR)
- TMI-1 MSLB Coupled 3-D Neutronics/Thermal Hydraulics Analysis: Application of RELAP5-3D and Comparison with Different Codes
- Utilization of a RELAP5 RCS and Secondary Plant Model in a Nuclear Power Plant Training Simulator
- RELAP5-3D Development & Application Status
- SCDAP-3DAnalyses
- Modification of RELAP5-3D in Accordance with Appendix K of 10 CFR 46
- TRAC-B/PC and RELP5/MOD2.5-PC Comparisons to the Oh Critical Heat Flux Experiments
- Input Data Preparation for the 3D Neutronic Model VVER-440/213 of System Code REAP5-3D and Its Testing
- Quantifying Code Variability for LBLOCA with RELA5-3D (Work in progress)
- Developments and New Directions for the RELAP5-3D Graphical User Interface
- RELAP5-3D User Problems
- Validation & Verification: Fluent/RELAP5-3D Coupled Code
- Transient Analysis Needs for Generation IV Reactor Concepts
- An Executive Program for Use with RELAP5-3D
- Full-Scope Simulators Running Real-time RELAP5-R/T

2002 International RELAP5 User Group Meeting, September 4 – 6, Park City UT

2002 Sixth International Information Exchange Forum

April 8 – 12, Kyiv Ukraine

2002 International RELAP5 User Group Meeting

September 4 – 6, Park City UT

- Status of RELAP5-3D Development and Application Status
- A Mass and Energy Conserving Form of Explicit Coupling for Use with RELAP-3D
- Incorporation of COBRA-TF in an Integrated Code System with RELAP5-3D Using Semi-Implicit Coupling
- Extensions of SCDAP/RELAP5-3D for Analyses of Advanced LWRs and HTGRs
- Adaptive Modeling of Thermal Non-Equilibrium in Two-Phase Flow Simulation Systems
- The Generation IV R&D Roadmap Overview
- Coupling of MELCOR to Other Codes under an Executive Program Using PVM Message Exchange
- Using the Coupled MELCOR/RELAP5 Codes for Simulation of the Edwards Pipe
- Breaking the Barrier Between Systems and Component Modeling; Coupling RELAP5-3D and FLUENT
- FLUENT/RELAP5-3D Coupled Code
- A New Assessment of the LOFT-Wyle Blowdown Test WSB03R Using RELAP5-3D
- A New Assessment of the Large-Tank General Electric Swell Problem Using RELAP5-3D
- RELAP5/MOD3.2 Analysis of INSC Standard Problem INSCPO-PR7
- Thermal Hydraulic Code Assessment Activities of the Technical University of Catalonia
- BRMK SP-2 Validation Results (KS PH Rupture Simulation)
- Development and Assessment of Appendix K Version of RELAP5-3D for LOCA Licensing Analysis
- RELAP5-3D Conversion to FORTRAN 90
- Analysis of Transients in an Actinide Burner Reactor Cooled by Forced Convection of Lead Bismuth
- Performance and Safety Studies for Multi-Application Small Light Water Reactor
- RELAP5 Applications in MGR Waste Package Analysis Methods
- SCAP/RELAP5-3D: A State-of-the-Art Tool for Severe Accident Analyses
- SCAP/RELAP5-3D-CONTAIN Linkage
- RELAP5-3D User Problems
- RELAP5 Input Builder GIUs THUMB and Pygmalion

2002 International RELAP5 User Group Meeting, September 4 – 6, Park City UT

2003 International RELAP5 User Group Meeting

August 27 – 29, West Yellowstone MT

- Status of RELAP5-3D Development and Application
- Improvements to the RELAP5-3D Turbine and Radiation Enclosure Models
- Addition of Noncondensable Gases into RELAP5-3D for Analysis of High Temperature Gas-Cooled Reactors
- RELAP5-3D Compressor Model
- Algorithm to Correct for Control Rod Cusping in the NESTLE Multidimensional Neutronics Module of RELAP5-3D
- Programming Improvements in RELAP5-3D
- RELAP5-3D User Reported Problems & Requested Improvements
- A Comparison of the Momentum Equations in RELAP5-3D and Fluent
- Coupling RELAP5-3D and Fluent for VHTR Analysis
- Modeling of Two-Phase Flow and Boiling with Fluent
- Prismatic Core VHTR Analysis Using RELAP5-3D/ATHENA
- A Parametric Study of the Thermal-Hydraulic Response of Supercritical Light Water Reactors During Loss-of-Feedwater and Turbine-Trip Events
- RELAP5-3D Integrated Primary System Reactor (IPSR) Simulations
- Peach Bottom Turbine Trip Benchmark Analysis with RELAP5-3D Coupled Code
- Simulation of Reactor Coolant Pump Asymmetric Transient Occurred in Vandellos-II NPP
- RELAP5-3D Model for the PANDA Facility. Application to the Simulation of an ISP-42 Experiment
- RELAP5-3D Validation Study Using MB-2 Prototypical Steam Generator Stead State Data
- SCDAP/RELAP5-3D Analysis Supporting Improved In-Vessel Retention Margins for High Power Reactors
- An Unmet Challenge: Application of SCDAP/RELAP5-3D to Analysis of Severe Accidents for Light Water Nuclear Reactors with Heavily Fouled Cores
- Modeling of Supercritical Pressurized Water Reactor with SCDAP/RELAP5-3D
- Framatome ANP's Realistic Large Break LOCA Methodology
- On a numerical Horizontal Flow Anomaly and the Second Law of Thermodynamics
- Extension of SNAP Model Editor to Support the RELAP5-3D Code
- RGUI

2003 International RELAP5 User Group Meeting

2004 International RELAP5 User Group Meeting

August 25 – 27, Sun Valley ID

- Fusion R&D Analysis Methods
- Status of RELAP ATHENA Development and Applications
- Improvements to the Pressurizer Component Model
- Improvements to the Steady-State Mode
- RELAP-3D Architectural Developments in 2004
- RELAP5-3D Reported User Problems and Requested Improvements
- A Startup Study of Power Conversion Cycle for SCWRS
- Assessment of Margin for In-Vessel Retention in Higher Power Reactors
- Discussion on the Calculational Envelope of the Fluent Computational Fluid Dynamics Code and the RELAP5 Systems Analysis Code When Using Segregated Solvers
- General IV Reactor R&D Plan
- 11% Upper Plenum Break: Application of RELAP-3D and Comparison with Other Codes
- Developmental Status of the Multi-Dimensional RELAP5 in Korea
- AREVA's Activities Related to VHTR Thermal-Hydraulic Analysis Using RELAP5-3D
- Modeling of a Power Conversion System with RELAP5-3D and Associated Application to Feedwater Blowdown Licensing Analysis for the ABWR Containment Design
- Contribution to Large Break LOCA Analysis for Commercial NPP Using RELAP5-3D
- Water Bulk Acceleration by Rapid Air Injection
- An Evaluation of a Novel Safety Concept for the SCWR
- Assessment of a Molecular Diffusion Model in RELAP-3D

2004 International RELAP5 User Group Meeting

- GFR-He/CO2 Analysis Using RELAP5-3D/ATHENA
- Java Conversion of XMGR5
- Parallel Programming Tutorial

2005 International RELAP5 User Group Meeting

September 7 – 9, Jackson Hole WY

- RELAP5-3D/MOD3.3 Merger
- RELAP5-3D/MOD3.3 Features
- Status of RELAP-3D Development and Applications
- Compressor Component Model
- Implementation of Molten Salt Properties into RELAP-3D
- RELAP5-3D Architectural Developments in 2005
- Coupling of RELAP5-3D and SIMULATE-3K for Transient Analysis
- RELAP-3D Reported User Problems and Requested Improvements
- A Study of Power Conversion Cycle for the Small-Modular Sodium Cooled Fast Reactor
- Summary of VHTR Methods R&D
- Modeling the GFR with RELAP5-3D
- Assessment of RELAP5-3D Using NACOK Natural Circulation Data
- Integrated Training and Accident Analysis (ITAAS) Simulation of the RBK-1000 Kursk 1 NPP Using RELAP5-3D Thermohydraulic
- Analysis of a Proposed Gas Test Loop Concept
- A RELAP5/MOD3 Model for the Estimation of Air Pull-through for a Draining Tank
- HTTR RCCS Mockup Model Using RELAP5-3D
- Evolving the ATHENA Model of a LOCA Analysis for the Generation IV Gas-Cooled Fast Reactor
- Downcomer Modeling Operations for LOFT Experiment L2-5: Sensitivity Analyses
- Java Conversion of RGUI
- Open Discussion on RELAP5-3D/MOD3.3 Merger

2006 International RELAP5 User Group Meeting

August 16 – 18, West Yellowstone MT

- Status of RELAP5-3D Development and Applications
- Modeling Compact Counter-Flow Heat Exchangers with RELAP-3D
- Enhancements to the RELAP-3D Heat Conduction Model
- FORTRAN 90 Conversion of RELAP-3D
- Restructuring RELAP-3D
- RELAP5 Analysis of Operator's Response to Nuclear Power Station Blackout
- Status of NGNP Methods R&D
- Status of the Activities Related to the use of RELAP5-3D at the Technical University of Catalonia

- SNAP Graphical User Interface
- Development of a Gas Cooled Fast Reactor RELAP5-3D Thermal-Hydraulic Model
- Modeling EBR-II LOFWS and LOHSWS using RELAP5-3D
- Applicability of RELAP5-3D for Thermal-Hydraulic Analyses of a Sodium-Cooled Actinide Burner Test Reactor
- RELAP-3D Coolability Calculation for Irradiated, Dried NpPu Targets in Proposed Storage Rack Configuration
- Licensing Analysis of RPV and BOP Blowdown with RELAP-3D/K for Lungmen ABWR Containment Design
- Use of RELAP-3D for Dynamic Analysis of Closed-Loop Brayton Cycle Coupled to A Nuclear Reactor
- The Use of RELAP-3D Code in the OECDNEA VVER-10 CT-1 and CT-2 Benchmark
- Performance Assessment of the Two-Phase Pump Degradation Model in RELAP5-3D
- RELAP5 Modeling of the INL/OSU GRTS Facility
- AREVA's Activities Related to VHTR Thermal-Hydraulic Analysis Using RELAP-3D
- The use of RELAP5-3D Code in the OECDNEA BFBT Benchmark
- Development of a 3D Neutron Kinetic-Thermal Hydraulic Model for an RBMK Reactor by RELAP5-3D Code

2007 International RELAP5 User Group Meeting

November 7 – 8, Idaho Falls ID

- Evaluation of Fluid Conduction and Mixing Within a Subassembly of the Actinide Burner Test Reactor
- Non-Condensable Gas Solubility Modelling
- Experience Using RELAP5-3D for Reduced Enrichment Research and Test Reactor Analysis the Good and the Not So Good
- Progress Report: Fortran 90 Conversion
- RELAP5-3D Versions and Accounting
- GAMMA Code for NGNP Air Ingress Analysis
- RELAP5-3D Developmental Assessment Plan
- SCDAP/RELAP5-3D – A Mechanistic Tool for Severe Accident Analysis
- Symbolic Nuclear Analysis Package (SNAP)
- STREAMLINING OF RELAP5-3D
- NGNP Methods: Summary of Approach and Plans

2008 International RELAP5 User Group Meeting

November 18 – 20, Idaho Falls ID

- Coupled Thermal-hydraulic and Neutronic Model for the Ascó NPP using RELAP5-3D/NESTLE
- Towards the Development of Thermal-Hydraulic Models aimed at the Auxiliary Systems Design in Nuclear Fusion Technology
- Three-Dimensional Neutron Kinetics/Thermal-Hydraulics VVER1000 Main Steam Line Break Analysis by RELAP5-3D© Code
- Outline of the Assessment Activities by RELAP5-3D© performed at University of Pisa: Post-Test Analyses of PKL-III PSB-VVER AND ROCOM Facilities Experiments
- Atucha-2 PHWR Three Dimensional Neutron Kinetics Coupled Thermal-Hydraulics Modelling and Analyses by the RELAP5-3D©
- RELAP5 Analysis of Two-Phase Decompression and Pressure Wave Propagation
- A System Analysis Code to Support Risk-Informed Safety Margin Characterization
- Verification and Validation of Corrected Versions of RELAP5 for ATR Reactivity Analyses
- Stratified Flow Phenomena Graphite Oxidation and Mitigation Strategies of Air Ingress Accident
- Assessing the RELAP5-3D Conduction Enclosure Model
- INL Experimental Program to Support Validation of CFD Codes

- Real Time Models in 3KEYMASTER Simulation Environment
- U.S. LWR Sustainability Program
- RELAP5 Analysis of Two-Phase Decompression and Rarefaction Wave Propagation under a Temperature Gradient
- The Use of RELAP5-3D for Subchannel Analysis of SFR Fuel Assemblies
- Experimental RELAP5-3D Time Step Improvements
- RELAP5-3D in Fortran 90
- Lunar Fission Surface Power Design – RELAP5 Point Kinetics
- Status of Attila for the Advanced Test Reactor (ATR)
- NGNP Methods: Summary of Approach and Plans

2009 International RELAP5 User Group Meeting

August 10 – 12, Park City UT

- Summary of Features Contained in the New Fortran 90 Code Version
- RELAP5-3D in Fortran 90
- 3D Reactor Kinetics Upgrades to RELAP5-3D
- 3KEYRELAP5 – Improvements and Applications, Abstract
- Use and Applications of RELAP5-3D© at the San Piero a Grado Nuclear Research Group of the University of Pisa
- NGNP Applications of RELAP5-3D
- Viscous Effects in RELAP5-3D
- Natural Convection cooling of the GTL-1 Experiment in the Advanced Test Reactor
- ATR Experience with Attila™ – Update
- Space Reactors
- Institutionalize Card 1 Option 3
- Application of RELAP5-3D to an Innovative Sodium Cooled Reactor
- RELAP5 Comparisons With COMSOL Outcomes for a Simple Hot Spot Evaluation in HFIR, Abstract
- RELAP5-3D Developmental Assessment Results
- Snap Automated Testing Framework for the RELAP5-3D Developmental Assessment
- Conservation of Fluid Mass and Energy by RELAP5-3D During a SBLOCA Abstract
- Conservation of Fluid Mass and Energy by RELAP5-3D During a SBLOCA
- Streamlined Electronic Process for Generating RELAP5-3D Input Models Abstract
- Streamlined Electronic Process for Generating RELAP5-3D Input Models
- Enlargement of the Assessment Database for Advanced Computer Codes inRelation to the WWER Technology: Benchmark on LB-LOCA Transient in PSB-VVER Facility
- Recent RELAP7 Deveopments
- RELAP5-3D Licensing and Export Control

2010 International RELAP5 User Group Meeting

September 20 – 23, West Yellowstone MT

- 3D nodalization performances of MASLWR ITF by RELA5-3D v2.4
- Accuracy Based Generation of Thermodynamic Properties for Light Water in RELAP5-3D
- Air Cooling Analysis of an ATR Fuel Element Using RELAP5 and ABAQUS
- CASL – The Consortium for Advanced Simulation of Light Water Reactors
- Circulating Flow Inside the Bundle of a Vertical Condenser Under Low Flow Rate
- Design and Analysis of the PB-AHTR using RELAP5
- Development of a Standard for V and V of Software Used to Calculate Nuclear System Thermal Fluids Behavior

- Development of RELAP5-3D Model or VVER-440 Reactor
- DTSTEP – Development of an Integer Time Step Algorithm
- HTTF Analyses Using RELAP5-3D
- Industry Use of Thermal Hydraulic Codes
- Measurements of Version 3
- OECD-NRC Benchmark Based on NUPEC PWR Subchannel and Bundle Tests (PSBT) – Comparisons among RELAP5-3D v2.4.2, RELAP5-Mod3
- PVM Coupling Junction Component
- RELAP5 Analyses of a Deep Burn High Temperature Reactor Core
- RELAP5-3D Flexible Wall Component Fluid Property Improvements
- RELAP5-3D Software Licensing Updates
- The use of RELAP5 in the Structural Verification of a Swedish NPP
- Use of RELAP5-3D at GRNSPG0-UNIPI
- 2D_POW_LDLOCA
- Hot Channel 1D
- Test Tresca and Deformed
- Viscous Stress Terms for the RELAP5-3D Momentum Equations
- Water Packer Nearly-Implicit Scheme

2011 International RELAP5 User Group Meeting

July 26 - 28, Salt Lake City UT

- NOCR FB1 Conv Movie
- VHTR CONF Movie
- Zweibaum Movie
- INSTANT/PHISICS — RELAP5 coupling
- Beta F90 Version Status
- RELAP5-3D Fluid Properties Modifications
- RELAP5-3D Developmental Assessment
- RELAP5-3D Analyses Supporting HTTF Design
- Summer 2011 INL internship project Pygmalion conversion 90/95
- RELAP5 and CASL
- Multipliers for Single-Phase Heat Transfer Coefficients in RELAP5-3D
- RELAP5-3D Software Licensing Updates
- Hybrid Capsule Assembly for Irradiation Testing
- RELAP5-3D Architecture and Style
- RELAP5-3D 2011 Solver Improvements
- Recent Applications of RELAP5-3D at GRNSPG
- Using RELAP5 to Analyze Pressure Relief Systems for Noncondensable Gases
- RELAP5-3D Activities at the ENEA – SIMING Lab
- RELAP5-3D Uncertainty Analysis
- Advanced Neutronics: PHISICS project
- VHTR Modeling and Experimental Validation Studies
- Mars Hopper Project
- Texas A&M RCCS Experimental Facility RELAP5-3D Simulations
- Introduction to mPower
- RELAP5-3D AT THE INL
- LWR Sustainability (LWRS) Program Status Update
- A Pilot Framework for Modern Reactor System Analysis Codes
- Salt Cooled High Temperature Reactors and Porous Media Flow Modeling in RELAP5-3D

2012 International RELAP5 User Group Meeting

2012 International RELAP5 User Group Meeting

October 23 – 24, Sun Valley ID

- Advanced Testing
- AER Dynamic Benchmark Results
- Improved Accuracy for Two-Phase Downflow Scenarios
- International Applications of Nuclear Safety Codes
- Metal Water Reaction Improvements
- Performance Measures for 4.0.3
- Recent Applictions of RELAP5-3D at GRNSPG
- RELAP5-3D Calculations Supporting HTTF Operation
- RELAP5-3D Code Activities in ENEA
- RELAP5-3D Developmental Assessment Update
- RELAP5-3D Model Improvment
- RELAP5-3D Participation in CASL
- RELAP5-3D Software Licensing
- RELAP5-3D Statistics Based Uncertainty Anaylsis
- RELAP5-3D-Phisics Coupling Advancements
- Resolution of Flow Direction Depenedence of Critical Flow Models in RELAP5-3D
- Selected Improvements of Opt Level 1 & 2
- Selected User Problems
- Verification Testing
- Version Status
- Window 7 and Pygi Development for Version 4.0.3

2013 International RELAP5 User Group Meeting

September 12 – 13, Idaho Falls ID

- RELAP Training
- General Approach to Integration of RELAP5 with Other Codes and Remote Web Based Execution
- Selected RELAP5-3D User Problems
- RELAP5 Version 4.1.3
- RELAP5-3D Kinetics Upgrades
- HTTF Scoping Calculations
- Comparison of MYRRHA RELAP5 Mod 3.3 and RELAP5-3D Models on Steady State and PLOF Transient
- Evaluation of Variations in the ATR Axial Power Distribution on Core Safety Margins
- Thermal-Hydraulic Analysis Resultsof a Seismically-Induced Loss of Coolant Accident Involving Experiment Out-of-Pile Loop Piping at the Idaho National Laboratory Advanced Test Reactor
- User Tool Upgrades — Pygmalion Status and RGUI Station
- Comparison of NRELAP5 to an ORNL THTF Test and the NuScale Critical Heat Flux Tests
- NuScale CHF & RELAP5-3D
- Architectural Issues and Developments in RELAP5-3D
- Developer Guidelines for RELAP5-3D Programming, 2013

- Improvements in RELAP5-3D Plot and Strip Files
- Verification/Restart/Backup Testing For RELAP5-3D
- RelapManager-A B&W mPower Safety Analysis Tool
- Overview of the RELAP5-3D Code Activities in ENEA
- Raven as Probabilistic Risk Assessment Tool for RELAP5-3D
- Assessment of RELAP5-3D for Future Reactor Designs
- RELAP5-3D Validation Using HTTF Data
- RELAP5-3D Software Licensing Updates
- Status of the PHISICS/RELAP5-3D Coupling and Application to Phase I of the OECD/NEA MHTGR-350 MW Benchmark
- Thermal-Hydraulic Research Activity
- RELAP-7 Code Development Status Update and Future Plan
- Verification and Validation of a Single-Phase Natural Circulation Loop Model in RELAP5-3D

2014 International RELAP5 User Group Meeting

September 11 – 12, Idaho Falls, ID

- RELAP5-3D Advanced Training
- A Quantitative Approach for Making Qualitative Assessment Judgments
- Application of RELAP5 to the BR2 and RHF Research Reactors for the GTRI Fuel Conversion Project
- BWR Probabilistic Risk Analysis using RELAP5-3D and RAVEN
- Capabilities of PHISICS/RELAP5-3D
- HTTF RELAP5-3D Applications and Assessment
- Impact of Pressure Relief Holes on Core Coolability for a PWR During a Large-Break Loss of Coolant Accident With Core Blockage Using RELAP5-3D
- MYRRHA Primary Heat Exchanger stability analysis
- New Governing Equations for the Realistic Representation of 2 Phase Flow
- R&D And Experimental Activities at ENEA Connected with the Use Of RELAP5-3D Code
- RELAP5 Applications & Improvements At NuScale
- RELAP5-3D Auxiliary Tools
- RELAP5-3D Improvements 2014
- RELAP5-3D Verification 2014
- RELAP5-3D Version 4.2.1 Developmental Assessment
- RELAP-7 Code Development Status Update
- Selected User Problems and Version 4.2.1 Features
- Simulation of EBR–II SHRT–17 Test by RELAP5-3D© Code
- Status of Recent Nodal Kinetics Upgrades in RELAP5-3D
- User–Reported Problems September 2013 – August 2014
- Variable Gravity

2015 International RELAP5 User Group Meeting

August 13 – 14, Idaho Falls ID

- RELAP5-3D Advanced Training
- Demonstration of BEUP Analysis of LB-LOCA with RELAP5-3D for High Burnup Fuel
- Development of a Quality Assured HTTF RELAP5-3D Input Model
- Development of a RELAP5-3D Property Library for Use by Other Computer Codes
- Dual Number Differentiation in RELAP5-3D
- H2ON Property Table Errors UP-15020
- Impact of Pressure Relief Holes on Core Coolabilityfor a PWR During a Large-Break LOCA With Core Blockage Using RELAP5-3D
- Improvements – Multi Deck and Strip Features
- Improvements in Sequential Verification
- Jacobian Evaluation Project
- Modeling Moving Systems With RELAP5-3D (Watch)
- Modeling the SCO2 Power Cycle of a generic dual Coolant Fusion Reactor With RELAP5-3D

- moving-motion-video
- Phisics-RELAP5-3D Coupled Suite for Industry Applications
- Qualitative and Quantitative Evaluation of Coupling Approaches for Coupling RELAP5 and LabView
- RELAP5 Applications at GRNSPG – NINE_ Thirty Years of Activities
- RELAP5-3D Auxiliary Tools – 2015 Update
- RELAP5-3D Gas-Cooled Reactor Activities
- RELAP5-3D Software Licensing Updates
- RELAP-7 Code Development Status Update for 2015
- SCREED – A Supporting Tool for V-V and Uncertainty Evaluations of Best Estimate System Codes for Licensing Applications
- Six Field Governing Equations for RELAP5
- The BEPU Evaluation Model with RELAP5-3D for Licensing nthe Atucha II NPP
- Thermal-Hydraulic Analysis of an AHTL In-Pile Tube in the Southwest Loop of the ATR
- User Problems-Version Features
- Validation of a RELAP5-3D Point Kinetics Model of Treat

2016 International RELAP5 User Group Meeting

October 3 – 7, Idaho Falls ID

- RELAP5-3D Advanced Training and RAVEN Training
- 2016 User Problems
- A Statistical Method for Benchmarking Nuclear Plant Models
- Equations of State for PbLi
- FLASHback – RELAP at Fifty
- Improvements to PHISICS_RELAP5-3D Capabilities for Simulating HTGRs
- Jacobian Matrix Analysis Tool for RELAP5-3D
- Modification to the Condensation Model in the Presence of Noncondensables
- RELAP5 3D Simulations of Hot Leg Break LOCA Scenarios
- RELAP5-3D Analyses for the US-DOE LWRS RISMC Program, Industry
- RELAP5-3D Licensing Updates
- RELAP5-3D Two-Phase Behavior and Predictions at Low Pressures
- RELAP-7 Current Status and Future Development
- Risk-Informed Safety Margin Characterization
- Seismic Loop LOCA with Double-ended Guillotine Break at IPT Inlet and Outlet
- Thermal Hydraulic Analysis of a Gas-Cooled Reactor
- Thermal-Hydraulic Analysis of a Versatile Coupled Test Reactor
- Update on RELAP5-3D CHF Geometry Correction Factors
- Use of Systems Analysis Codes in the Nuclear Industry

2018 International RELAP5 User Group Meeting

April 30 – May 4, Idaho Falls ID

- RELAP5-3D Introductory Training and RAVEN Training
- Acceleration of UQ and PRA with RAVEN Hybrid Modeling of a Blend of Surrogate and RELAP5-3D Models
- Advancements of PHISICS – RELAP5-3D Package for Time-Dependent Transient Calculations
- Application of RELAP5-3D for the Risk Informed External Event Analysis
- Consistent Fluid Property Evaluation for RELAP5-3D
- Development of New H2O95N Fluid Properties
- Integrating Classical PRA Models into RELAP5-3D
- Kinetics Advances in RELAP5-3D
- Measuring Risk Importance in a Dynamic PRA

Framework Using RAVEN – RELAP5

- Multi-Physics Best Estimate Plus Uncertainty (MP-BEPU) Analysis Framework LOTUS and RELAP5-3D
- Ongoing RELAP5-3D Related Activities at SCK-CEN
- PbLi Equations of State
- RELAP5-3D Backup Improvements
- RELAP5-3D Licensing Updates
- RELAP5M3_3 – RELAP5-3D – CFD Calculation of Reactor Downcomer Velocity and Temperature Fields in PTS Analyses
- RELAP7 Progress and Future
- Selected User Problems
- System Modeling of the HTTR with RELAP5-3D
- Timing Studies of Recent RELAP5-3D Code
- Validation of a Detailed RELAP5-3D Model of TREAT

2019 International RELAP5 User Group Meeting

April 15 – 19, Idaho Falls ID

- RELAP5-3D Advanced Training
- Accident Tolerant Fuels – A PRA Comparison
- Analysis of Accident Tolerant Fuel Using RELAP5-3D
- Analysis of the Versatile Test Reactor Using RELAP5-3D
- Analysis Using RELAP5-3D as Compared with a Test Mockup
- Bison-TRACE Coupling
- Development of Quantitative Verification Capabilities for use with RELAP5-3D and R5EXEC
- Direct Sparse Matrix Linear Solver for the Nodal Kinetics in RELAP5-3D
- Dynamic PRA of a Multi-unit Plant
- FPoliSolutions RISA Technology Status and Path Forward
- Jacobian Consistency Tool for RELAP5-3D
- RELAP5-3D – the Path Forward
- RELAP5-3D Application to Risk-Informed Systems Analysis
- RELAP5-3D Software Licensing Updates
- RELAP-7 Development Status
- Selected User Problems
- Six-Field Model and RELAP5-3D
- TWERL for TREAT Pre-Conceptual Design
- Upgrade to GFORTRAN and FORTRAN 2003
- Validation of a Simple RELAP5-3D Point Kinetics Model of the Full-Slotted MARCH Core in TREAT
- Validation of RELAP5-3D Using HE-FUS3 Data

2021 International RELAP5 User Group Meeting

September 13 – 17, Virtual Meeting

- RELAP5-3D Basic and Advanced Training
- Application of RELAP5-3D for Liquid Metal Reactor Safety
- Application of RELAP5-3D to High Energy Deposition Transients within TREAT
- Application of Surrogate Models for Best Estimate Plus Uncertainty Analysis by RELAP5 Code
- Development of Best Estimate Plus Uncertainty (BEPU) Application for RELAP5-3D
- Human Reliability Analysis
- JAEA HTTR Secondary System Modeling
- Modern Error Scaling
- Plutonium Fuel Services Irradiation Experiment in the Advanced Test Reactor
- Regression Testing for BR2 RELAP5 Models
- RELAP5-3D Architectural Upgrades through Gnu Fortran Adaptation
- Remote RELAP5-3D Information
- Risk-Informed ATF Analysis for Generic PWR & BWR
- Thermal-Hydraulic Modeling of MARVEL Microreactor

2023 International RELAP5 User Group Meeting

December 4 – 8, Idaho Falls ID

- RELAP5-3D Beginning Training
- RELAP5-3D 4.5.2 and Beyond
- Uniform Physical Units
- Dissolved Gas Model
- Molten Salt Point Kinetics
- Molecular Diffusion and Fluid Updates
- Remote RELAP5-3D Container
- Application of RELAP5-3D to Subchannel Analysis by Using the Experimental Data of NACIE-UP Facility
- Sapienza Contribution to the IAEA CRP on the Benchmark Analysis of FFTF Loss Of Flow Without Scram Test
- Assessment and Application of RELAP5-3D to Gen-II, -III, and -IV Reactor Designs
- RELAP5-3D Simulations for HTGR at the Canadian Nuclear Laboratories
- Reverse Natural Circulation Virtual Benchmark Activity
- HTTF Benchmark
- Domain Overlap R5: STAR and Open FOAM
- Coupling with MOOSE
- MARVEL
- Consortium Merge of RELAP5 Versions
- HENRI System in the TREAT
- Analysis of SiC ATF Reliability in Accident Conditions
- Lesson Learned from the Assessment of the MULTID Component
- OGMA-Sim: A Generic API for Enterprise Data Management and Ontology of Large-Scale Simulation Sets using RELAP5-3D
- Pathfinder: Underwater Nuclear Power Plant
- RELAP5-3D Application LWRS
- PCAT

2023 International RELAP5 User Group Meeting

- A Roadmap Toward Improved Transient CHF Prediction Within RELAP5-3D
- Flow Boiling Transient Critical Heat Flux Prediction in RELAP5

2024 International RELAP5 User Group Meeting

September 9 – 13, Park City UT

- RELAP5-3D Intermediate Training
- Ongoing RELAP5-3D Validation Activities for HTGR Analysis
- RELAP5-3D Modeling for Supporting Thermal-Hydraulics Analyses of Research Reactors
- Nuclear Thermal Propulsion using RELAP5-3D Coupled to MOOSE
- PCAT Modeling and Initial Testing Results
- RELAP5-3D Modeling of the Lotus Molten Salt Reactor
- MARVEL Design and Safety Analysis
- TREAT Sodium Loop Modeling and Simulation
- RELAP5-3D Consortium
- Version Maker
- Precompiler Directive Removal
- Four Pressure Model
- Development of a MULTID Model to Perform PIRT-Relevant Transient Analysis Related to ALFRED Reactor

Chapter 4. International RELAP5 User Group

The International RELAP5 User Group (IRUG) comprises organizations licensed to use the RELAP5 nuclear power plant safety analysis code, developed at Idaho National Laboratory (INL) since 1979. The objective of IRUG is to serve industry, government, and academia by providing and maintaining a versatile RELAP5 package for safety and engineering analyses of nuclear power reactors. IRUG oversees RELAP5–3D improvements, ensuring it remains a leading reactor safety systems code.

Membership is open to individuals and organizations, including governmental entities, commercial companies, and academic institutions. Requests for access must be vetted by the Nuclear Computational Resource Center (NCRC). Membership fees, based on usage level (from basic to superuser), are charged annually and adjusted as needed. Basic Membership allows access to RELAP5-3D on the INL supercomputer platforms, while higher access levels allow the code to be ported to the member's own platform. The highest levels allow access to the source code. All access levels are subject to US Government and INL scrutiny.

IRUG leadership is comprised of a chairman, vice-chairman, and secretary. Voting rights are based on membership level, however, all attendees may bring topics to the floor at the annual business meeting. Afterwards, meeting minutes are distributed prior to each meeting.

As of 2024, there are 87 IRUG organizations around the World

- Canada
- Brazil
- United Kingdom
- Spain
- Czech Republic
- Belgium
- Greece
- Italy
- South Africa
- Japan
- Indonesia
- Croatia
- Belgium
- France
- Taiwan Republic of China
- Thailand
- Jordan
- Slovakia
- California
- Connecticut
- Florida
- Georgia
- Hawaii
- Idaho
- Illinois
- Indiana
- Maryland
- Massachusetts
- Michigan
- Missouri
- New Jersey
- New Mexico
- New York
- North Carolina
- Ohio
- Oregon
- Pennsylvania
- Tennessee
- Texas
- Utah
- Virginia
- Washington
- Washington DC
- Wisconsin

4.1. List of RELAP5-3D and R5EXEC Manuals

Membership in the user group also grants access to certain of the RELAP5-3D manuals. The manuals that are accessible are based on membership level.

4.1.1. RELAP5-3D

1. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume I: Code Structure, System Models and Solution Methods, INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
2. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 2: User's Guide and Input Requirements, INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
3. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 2A: User's Guide and Input Requirements, INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
4. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 3: Developmental Assessment INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
5. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 4: Models and Correlations, INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
6. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 5: User Guidelines, INL/MIS-15-36723, Rev. 4.5, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2023).
7. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 6: Independent assessment and Applications, INL/MIS-15-36723, Rev. 4.4, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2021).
8. RELAP5-3D CODE DEVELOPMENT TEAM, RELAP5-3D Code Manual Volume 7: Programmers' Manual, INL/MIS-15-36723, Rev. 4.4, Idaho National Laboratory, Idaho Falls, ID, USA, https://relap53d.inl.gov/SitePages/Home.aspx/ (2018).

4.1.2. R5EXEC

These manuals and the associated R5EXEC Program are available only to members of the laboratories that comprise the RELAP5-3D Consortium. R5EXEC drives calculations teaming RELAP5 with CFD, Containment, or other programs that are properly equipped with R5EXEC access code, which uses Parallel Virtual Machine (PVM) calls. The executive starts each coupled code and then tells them what to exchange, when to do it, and what time to stop and wait for further directives.

1. J. Hope Forsmann and Walter L. Weaver, III, Programmer's Manual for R5EXEC Coupling Interface in RELAP5-3D Code, INL/EXT-05-00203, Rev. 2, Idaho National Laboratory, Idaho Falls, ID, USA (2015).
2. J. Hope Forsmann and Walter L. Weaver, III, Programmer's Manual for R5EXEC, INL/EXT-05-00159, Rev. 2, Idaho National Laboratory, Idaho Falls, ID, USA (2015).
3. J. Hope Forsmann and Walter L. Weaver, III, Application Programming Interface for R5EXEC Program and Associated Code Coupling System R5EXEC, INL/EXT-05-00107, Rev. 2, Idaho National Laboratory, Idaho Falls, ID, USA (2015).
4. J. Hope Forsmann and Walter L. Weaver, III, Application User Guide for the R5EXEC Coupling Interface in the RELAP5-3D Code, INL/EXT-15-35005, Rev. 2, Idaho National Laboratory, Idaho Falls, ID, USA (2015).

Chapter 5. Licensing, Membership and Access Levels

RELAP5-3D is a commercial code with licensing fees. The license fees are based upon the six RELAP5-3D membership levels. The membership level with the most features is Diamond while Academic is the entry level membership. The membership level defines the benefits provided to the license holder. The RELAP5-3D membership levels and their associated benefits are shown in the tables below.

Licensing agreement discussions with the INL Technology Deployment office are facilitated by the Agreements Administrator. Requests are made through the INL Nuclear Computational Resource Center (NCRC) website. Organizations requesting a license will identify a technical point of contact (TPOC) for their license who will register on NCRC for a High-Performance Computing (HPC) account, with the RELAP5-3D Program Manager as their HPC account sponsor.

The TPOC requests the RELAP5-3D software, which initiates contact with the Agreements Administrator. A RELAP5-3D license is based on a standard license agreement, but the requesting organization can opt for a customized one, which involves additional iterations with the INL Technology Deployment office.

Once payment is processed, the organization receives their RELAP5-3D license. Adding users to the license is flexible and can be done at any time using the NCRC. Each user applies for an HPC account and with provided credentials, requests RELAP5-3D access based on the organization's license. Since the U.S. Department of Energy (DOE) has designated RELAP5-3D as a 10 CFR 810 code, an export control review is conducted based on Personal Identifiable Information. The INL Code Oversight Group ensures compliance with DOE requirements. Both reviews are independent, but the export control review takes precedence. Incomplete or vague information provided by the requesting organization will delay this process.

License fees and associated member benefits are reviewed regularly. Increases in fees are provided to the users during the annual IRUG meeting.

Chapter 6. Looking Forward

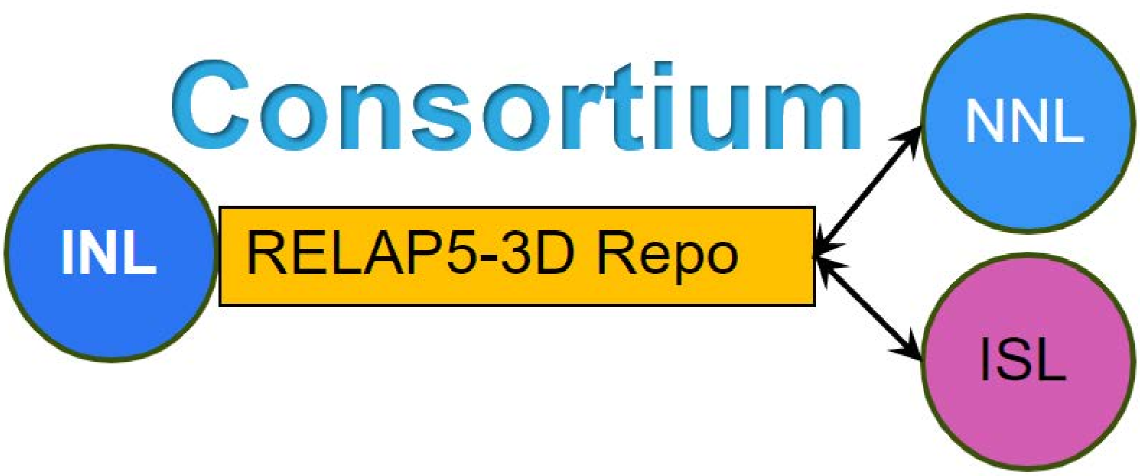
RELAP5-3D has a very bright future. The new team has been assembled that will take it into the future for next couple decades, management is committed to it, the nuclear industry is rapidly growing in many new directions and has needs that the code can easily adapt to and supply. And yet, many ideas for growing the code for future needs and usage are not driven entirely by the nuclear industry. Some are driven by advances in the computer industry and others are foreseen by the code's leaders, developers, and users. This chapter presents some of the myriad forward-looking capabilities that make sense for the advancement of RELAP5-3D.

6.1. RELAP5-3D Consortium

A consortium of laboratories was organized in November 2023 to advance RELAP5-3D development and analysis capabilities by sharing ideas, funding, and resources among the entities. The formation of its inner workings and operations are still underway, but it is comprised of three organizations:

- Idaho National Laboratory
- Naval Nuclear Laboratories (NNL) and
- Information Systems Laboratories (ISL)

Since it is difficult to find trained Fortran programmers, combining resources ensures an increased supply to all members for multiple purposes. Because there are many ways to program



Fortran code, even within the guidelines of style and programming manual constraints, our extended RELAP5-3D Teammates will consult, plan programming, debug, and devise the best ways to maintain the code.

The primary value of the consortium is to save time and funding. Before the consortium, besides funds used to develop coding at one of the three labs, more funds had to be expended to get the code into either the INL or NNL version. The consortium prevents this waste. Moreover, the usual time of 1.5 years to merge code versions, and fund it, will also be saved. Training in Fortran and RELAP5-3D usage and programming can be merged also. All of these valuable savings are contingent on devising means to implement the exchange flawlessly.

To begin, a GIT repository has been created for holding the common version of the code. This common code, also called the base code and the Consortium code, was agreed upon after months of careful consideration and further months of work to create and test it. A complication exists because both NNL and INL have "higher" code versions with proprietary or other restrictions, which are kept in a separate GIT repository.

For various reasons, a means to separately develop the consortium code and the higher codes, then move allowable coding from one repository to the other within the lab doing the work was needed and has been devised. INL developed and maintains a version maker program that configures code versions for a particular client class by remove coding from INL's higher version, called the TOTAL version. The version make was upgraded to place special markers in the consortium code wherever coding was removed for the total version. When code is added or modified in the total version, it is moved to the consortium version as is, or does not move if it is restricted from access by the consortium. On the other hand, coding moves from the consortium version to the total version without restriction, but it must be tested whenever the coding appears immediately above or below one of the special marks to be certain it is properly placed.

INL created a continuous integration scheme called CIVET. The Consortium version now uses CIVET to test any code update before allowing the new coding or modification of existing coding into the GIT repository. This test system now incorporates all our test cases, but runs only a standard set daily and, because it takes a long time, the extensive set on weekends. This will ensure even greater robustness in RELAP5-3D going forward.

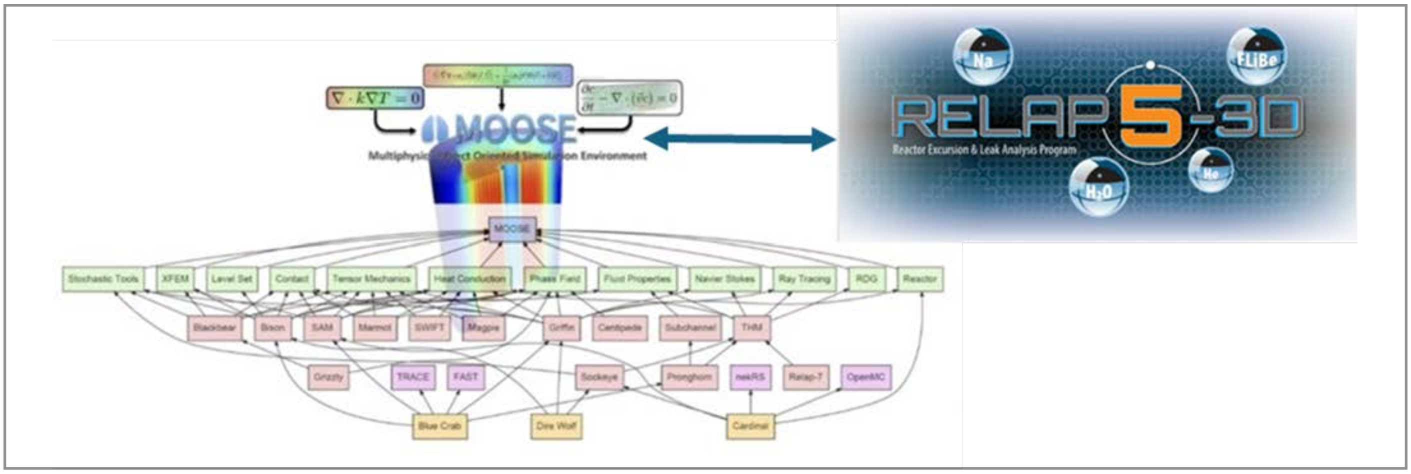
6.2. Manuals

All RELAP5 manuals are currently in Framemaker format. This format is outdated and cumbersome to work with compared with other editors. A consortium decision was made to rewrite the manuals in LaTex.

Besides reformatting, the manuals will be reworked contextually. Volume 2A is revamped to make things clearer and more uniform of exposition. Some sections, especially pumps, have been expanded and further clarified.

Volume 1 has been reviewed and edited. Many errors were located and flagged for correction. Material such as the Dissolved Gas Model (DGM) have also been added. Cross references with Volumes 2A and 4 have been improved. Units are given where they were previously missing. Volume 4 has also been reviewed, errors and improvements marked, and will be converted. Volumes 5 and 6 will simply be converted, though the latter needs updates.

Developmental Assessment (DA) Volume 3 is being reworked in multiple ways. First, the verification cases are being separated from the validation cases. A table of the test cases will be created up front to show how the code performed on each and will include a numerical measure of agreement. The SNAP script used to run the DA will be replaced by a Python script and need not use SNAP anymore. The plots may or may not be imported directly into Volume 3.



MOOSE-RELAP5-3D Coupling

In addition, Volume 3 will be expanded into fluid sections. Currently only aqueous test cases are included. Expansion to include liquid metals, gases, and certain molten salts will be included. This will greatly help modelers, designers, and companies making licensing submittals.

6.3. COUPLING RELAP5-3D

6.3.1. MOOSE

Coupling is the solution by several programs of complex problems by domain decomposition and applying each code to the portion of the domain where it excels. Most such methods have each code treat the other as supplying boundary conditions.

RELAP5-3D has been coupled with other codes in 5 difference ways. Through RELAP5 Interactive variables (which reads typing on the keyboard and other such external input); transfer via disk files (effective with slower codes such as CFD); with computer sockets (such as with PVM); via STAR-CCM+ user defined functions; and memory-to-memory.

A new Memory-to-Memory coupling of RELAP5-3D with MOOSE is nearly complete and being tested and extended. Coupling with the MOOSE herd will combine the system modeling capabilities of RELAP5-3D and RELAP-7 with the MOOSE visualization capabilities. The first combination will be MOOSE/BISON and RELAP5-3D; it will provide much improved modeling of fuel rods to RELAP5-3D and access to whole plant hydro to BISON. Other combinations will reap great rewards soon as well. This is an excellent teaming of top INL programs to model nuclear plants at higher fidelity than previously possible.

6.3.2. R5EXEC Upgrade

The R5EXEC program couples RELAP5-3D with other codes to better model complex systems. The original R5EXEC, initially called the PVMexec program initially, acts like the control program of a power plant simulator. It uses domain decomposition and applies different programs to the domain to which they are best suited. The executive reads its own input, spawns processes that run the programs, tells each program when to perform input, exchange data, start its calculations, and at what simulation time to pause and wait for further instructions. It also controls error conditions and shutting down. All the communication is through computer sockets using the Parallel Virtual Machine (PVM) protocol.

The PVM software has had reliability issues, and no new development has taken place in two decades. Alternatives will be considered. The Message Passing Interface (MPI) is more reliable and still receiving updates within the computer industry. It has not offered all the features that R5EXEC uses from PVM, but it still being developed and receives updates. Other replacements will also be considered. If a suitable alternative communication protocol can be found, the coding will be converted to use that protocol.

6.4. Best Estimate Plus Uncertainty (BEPU)

Best-Estimate Plus Uncertainty analysis allows the user to perturb selected closure laws and some of their most relevant parameters through input. There is a great deal of interest in this in the industry. It is a fundamental feature to enable the development of advanced analysis techniques to support the Light Water Reactor (LWR) fleet.

In fact, this capability has already been developed for that purpose in an orphaned version of RELAP5-3D and proved very valuable; however, it was not incorporated into the mainline for lack of funding. The capability to perform a full BEPU analysis would enhance the code in several more ways including parameter studies, verification, validation, and debugging.

6.5. Liquid Metals Advanced Simulation Capabilities

Many new kinds of reactor designs call for the use of liquid metals. RELAP5-3D needs better modeling capabilities to provide the anticipated accuracy level needed for analyzing these designs. Some anticipated improvements include:

- Implementing liquid metal thermal conductivity simulation for improving RELAP5-3D prediction for flow regimes at low Peclet numbers.
- Introducing a user capability to modify laminar and turbulent Reynolds numbers for improved prediction of pressure drops in wire-wrapped bundles.
- Implementing an Electro-Magnetic Pump as hydrodynamic component.
- Introducing Cover Gas-expansion volume as new hydro component.
- Adding test cases to the standard, verification, and validation test suites for these.

6.6. Third Primary Fluid Field

Introduction of a third field (e.g. droplets) of the primary coolant will improve simulation of LOCA scenarios. The third primary fluid field would increase the accuracy of peak clad temperature prediction and mass and energy transport in components simulating, for example, in the downcomer, hot legs, etc.

A third field was programmed into RELAP5-3D decades ago and the coding still exists in older code versions as an example of how it may be implemented. A doctoral dissertation by Glenn Roth provides recent theoretical development for 6 primary fluid fields and could be used as a starting place for development of a third field.

6.7. Python Usage

Given the growing popularity of Python in science and engineering fields it is important to harness this powerful trend in the day-to-day use of RELAP5-3D.

6.7.1. RELAP5-3D Service Scripts

Recent graduates and younger team members and computer staff have been trained in Python and not shell scripting, and therefore can develop new scripts with greater ease, speed, correctness, and performance when writing in Python. In fact, some of the installation and service shell scripts, used for decades for RELAP5-3D operations, are being converted to Python. More will be converted.

6.7.2. Python-wrapped RELAP5-3D

There is plenty of utility that can be added to RELAP itself by using Python. RELAP5-3D would still build and run independently of Python but would be able to take advantage of the powerful capabilities it offers. For example,

- A RELAP5-3D pre-processor that allows users to embed python substitutions, or even Python scripts, within their input decks could add tremendous capability and flexibility to input.
- A Python-based post-processor could allow users to harness powerful Python libraries like numpy, scipy and the famous matplotlib. Such a post-processor capability could provide an inexpensive alternative to some commercial postprocessor options.

6.8. Simulator Building Starter Packs

Every new nuclear power plant design needs a training simulator. A plethora of reactor simulators have been built off RELAP5-3D and is a fertile area for growth of RELAP5-3D. However, other application areas for RELAP5-3D also should have simulators, and demonstrating the value quickly and easily is a huge selling point for generating new clients and penetrating new markets.

Currently there are very inexpensive electronics like Raspberry-Pi and Arduinos that negates the need for expensive industrial equipment, and they can run RELAP5-3D! In the past, connecting with RELAP involved some modifications to the source, however, with recent advances in coupling capabilities users no longer need source access. We can make binaries available, with associated API documentation, to allow customized programming without complex licensing agreements. In fact, we can compile complete starter packs for universities and industry that can offer turn-key solutions to their simulator needs.

6.9. Other Code Improvement Ideas

Improve CHF-post CHF predictions for new Accident Tolerant Fuel technologies. This will help the current fleet of LWR to stay competitive by improving the plant performances through power uprates.

Extend NESTLE capabilities to N-neutron group, to allow simulation of fast-reactor systems.

Improve RELAP5-3D team capabilities by participating to IAEA and OECD/NEA benchmarking actives.

6.10. Educational Outreach

When students learn to use RELAP5-3D in school, they come out of school, take jobs, and typically request access to the code from their employers. This grows the user base. Therefore, RELAP5-3D Team has collaborated with many professors for teaching both graduate and undergraduate courses on RELAp5-3D. We have also served on master and doctoral committees of students of IRUG-affiliated universities. INL does not expect the professors or universities to pay a fee, normally, but for a research project that used the code, INL does require the professor to publish a paper in a conference or journal.

The RELAP5-3D group is working hard to allow more flexibility with our training materials. Once this is achieved, we want to reinvigorate the use of RELAP5-3D in classrooms. We will do this by engaging with universities to provide workshops as well as establishing a public-facing user community on GitHub. We have many enthusiastic plans on other media platforms as well, such as YouTube videos.

Chapter 7. RELAP5 Team

7.1. Volume 1 Authors

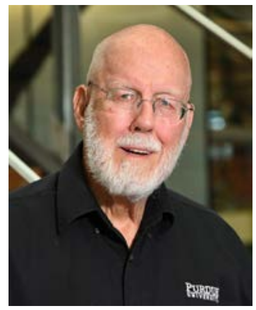
Development of a complex computer code such as RELAP5 is the result of team effort and requires the diverse talents of a large number of people. Special acknowledgment is given to those who pioneered and continue to contribute to the RELAP5 code, in particular, V. H. Ransom, J. A. Trapp, and R. J. Wagner. A number of other people have made and continue to make significant contributions to the continuing development of the RELAP5 code.

- V. T. Berta
- K. E. Carlson
- C. D. Fletcher
- E. E. Jenkins
- E. C. Johnsen
- G. W. Johnsen
- J. M. Kelly
- H-H. Kuo
- N. S. Larson
- C. E. Lenglade
- M. A. Lintner
- C. C. McKenzie
- G. L. Mesina
- C. S. Miller
- G. A. Mortensen
- P. E. Murray
- R. B. Nielson
- S. Paik
- R. A. Riemke
- R. R. Schultz
- A. S-L. Shieh
- R. W. Shumway
- C. E. Slater
- S. M. Sloan
- M. Wamick
- W. L. Weaver
- G. E. Wilson

Tri-Fathers/25th anniversary of RELAP 2003

7.2. RELAP Biographical Sketches

7.2.1. The RELAP Tri-Fathers



Dr. Victor Ransom
Project Leader and Researcher

Dr. Ransom's engineering career began following his graduation with a B.S. in Chemical Engineering from the University of Idaho in 1955. America's fascination with flight and space travel made the Aerospace Industry a rapidly growing industry and Vic was recruited by North American Aviation in Canoga Park California. There he got his first taste of developing transient simulation methods and computer programs for the analysis of supersonic flow.

In 1959 an opportunity for advancement opened at the Aerojet General Corporation in Sacramento California. This new position gave him supervisory responsibility over Aerojet's fluid mechanics and heat transfer groups. Here he developed and directed a large-scaled effort in simulation methods for liquid rocket engine systems. To advance his understanding of applied mathematics he began evening courses at Sacramento State; however, by the end of 1965 his interest in applied math and transient simulation led him to Purdue University.

Under the supervision of Dr. J. D. Hoffman, Vic pursued an investigation into numerical solutions of three-dimensional supersonic flow fields applying the Method of Characteristics. The Method of Characteristics is related to the application and interpretation of eigenfunctions of the mathematical model of interest. His final project was one of the first computer codes for the numerical solution of three-dimensional supersonic flow.

After graduation Dr. Ransom returned to Aerojet to continue his contribution to supersonic flow. In 1973 after Aerojet Nuclear was awarded the managing contract of the Idaho National Engineering Laboratory Dr. Ransom jumped at the opportunity to return to his Idaho home and try his hand at the nuclear power plant simulation. Shortly after arriving in Idaho, he along with Dr. John Trapp and Dick Wagner began building the foundation of what would become the RELAP5 best-estimate thermal-hydraulic computer code for light water reactor safety analysis.

In 1979 they received formal NRC funding to pursue the development of RELAP5 from its origin as the PILOT code. Over the next six years RELAP5 would develop into a world class systems code. Along the way Dr. Ransom and Dr. Trapp would contribute to many technical questions surrounding the modeling. of two-phase flow including: the role of numerical damping a two-pressure model of two-phase flow a choked flow model and the use of Characteristics to develop numerical boundary conditions. After the completion of RELAP5/MOD2 in 1985 Dr. Ransom continued his activity in RELAP5 development as a consultant. At that time, he was recognized by the INEL and promoted to Scientific and Engineering Fellow - the highest technical level for an employee at the INEL. In 1991 Dr. Ransom left the INEL to serve as professor and head of the Department of Nuclear Engineering at Purdue University.

His technical contributions are well documented in the peer reviewed literature and have garnered awards from the NRC (citation for RELAP5 development) the Idaho ANS Section (Fellow) the Society for Computer Simulation (citation) the University of Idaho (elected to the Alumni Hall of Fame) and most recently elected to fellow grade American Nuclear Society. He currently serves on the NRC Thermal Hydraulic Experts Committee and regularly consults on two-phase flow fundamentals.

Dr. John Trapp
Theoretician and Researcher

Mr. John Trapp developed the first draft of RELAP5 equations and numerical scheme; two mass conservation equations and two linear momentum equations. It was a constant temperature simulation with no energy equations. He simulated several constant temperature test problems. Additionally, he worked as a consultant on RELAP5 to develop a complete thermal hydraulic analysis using independent energy equations one for vapor phase and one for liquid phase. At each stage of the development several test problems were simulated to verify the code predictions. Other related tasks include: a critical flow model a separator model and input to friction and heat transfer correlations. In 1982 his consulting work on RELAP 5 was discontinued and he become a professor at University of Colorado at Denver teaching in the Mechanical Engineering Department until his retirement in 2012.

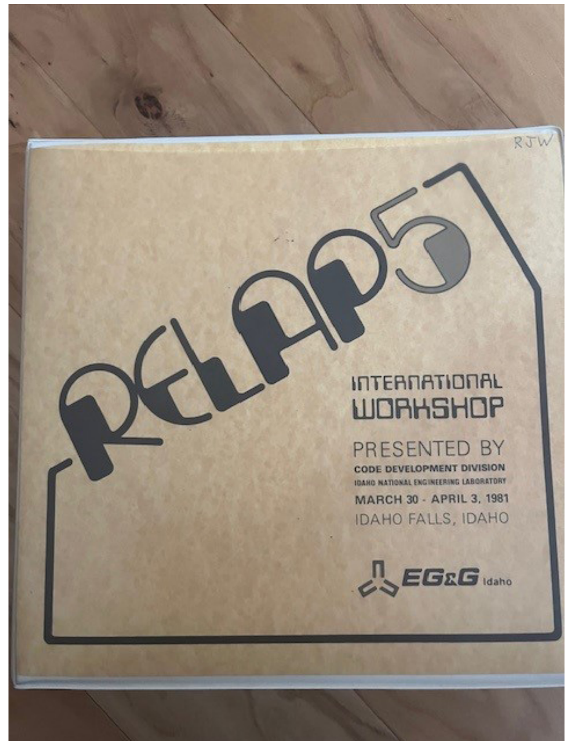
Dr. Trapp earned a B.S. in Mechanical Engineering (Long Beach State College Long Beach, 1965), a M.S. in Applied Mechanics (University of California at Berkeley 1966) and a PhD in Applied Mechanics with a Math minor (University of California at Berkeley 1971).

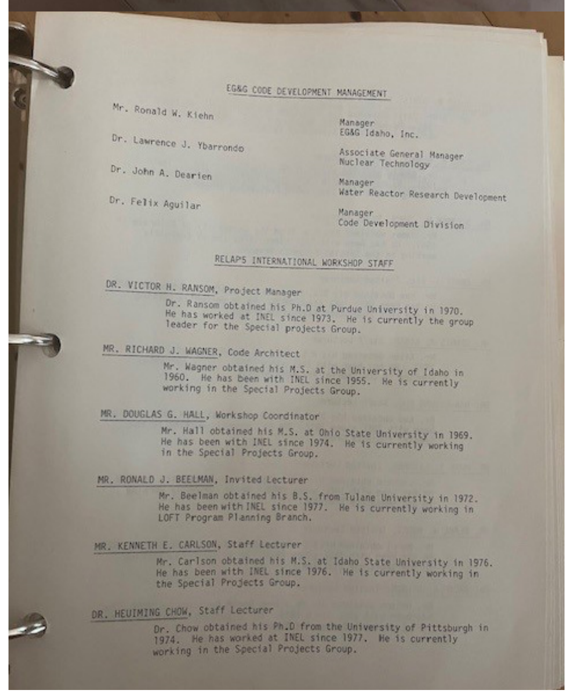
Mr. Richard Wagner
Code Architect and Physicist

Dick Wagner earned a B.S. in Physics at Seattle University in 1956. That same year he moved to Idaho Falls, started work at the National Reactor Testing Station, which eventually became INL. He married Maryann in 1957 and earned his MS in Physics from University of Idaho in 1960. In the late 1960's, he became the code architect in a group of engineers developing the RELAP1 computer program for predicting the containment pressure-temperature response to a LOCA for the US Atomic Energy Commission. In fact, Dick Wagner wrote the first line of the program, namely PROGRAM RELAP5.

Dick had many wise sayings, such as "I don't have time, in one lifetime, to understand all the mysteries of computing." Another was, "If the code makes it through input processing, it ought to run the entire transient." He always worked at least a half hour longer than was required each day, and he was one of the creators of the saying, "RELAP5 Do or Die!" You learned these things if you ever stopped by to ask him a question about the code. His answer always took one hour, and he answered about 20 other questions that he thought you should have asked!

It is said that an expert programmer knows and can write code that does everything that a computer language was designed to do. However, a master programmer can make a program do things that its computer language was never designed to do. Rumored to sleep with a Fortran book under his pillow, Dick was a programming Grandmaster. For





EG&G CODE DEVELOPMENT MANAGEMENT

Mr. Ronald W. Kiehn	Manager EG&G Idaho, Inc.
Dr. Lawrence J. Ybarrondo	Associate General Manager Nuclear Technology
Dr. John A. Dearien	Manager Water Reactor Research Development
Dr. Felix Aguilar	Manager Code Development Division

RELAP5 INTERNATIONAL WORKSHOP STAFF

DR. VICTOR H. RANSOM, Project Manager
Dr. Ransom obtained his Ph.D at Purdue University in 1970. He has worked at INEL since 1973. He is currently the group leader for the Special projects Group.

MR. RICHARD J. WAGNER, Code Architect
Mr. Wagner obtained his M.S. at the University of Idaho in 1960. He has been with INEL since 1955. He is currently working in the Special Projects Group.

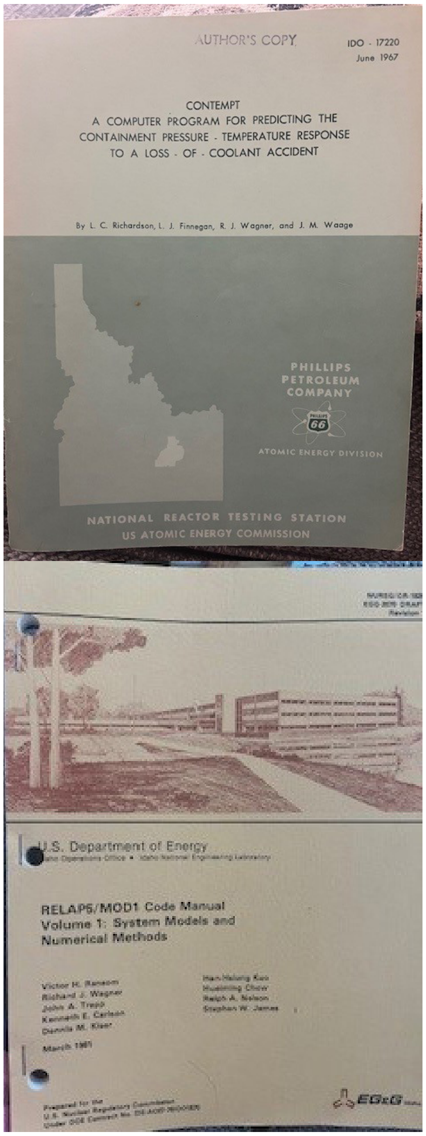
MR. DOUGLAS G. HALL, Workshop Coordinator
Mr. Hall obtained his M.S. at Ohio State University in 1969. He has been with INEL since 1974. He is currently working in the Special Projects Group.

MR. RONALD J. BEELMAN, Invited Lecturer
Mr. Beelman obtained his B.S. from Tulane University in 1972. He has been with INEL since 1977. He is currently working in LOFT Program Planning Branch.

MR. KENNETH E. CARLSON, Staff Lecturer
Mr. Carlson obtained his M.S. at Idaho State University in 1976. He has been with INEL since 1976. He is currently working in the Special Projects Group.

DR. HEUIMING CHOW, Staff Lecturer
Dr. Chow obtained his Ph.D from the University of Pittsburgh in 1974. He has worked at INEL since 1977. He is currently working in the Special Projects Group.

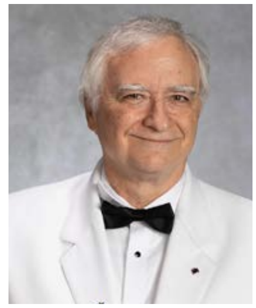
RELAP5 Workshop, 1981



RELAP books authored by Dr. Richard Wagner

example, he gave RELAP5 dynamic memory allocation/reduction, pointers, and derived types using FORTRAN66, two generations of standards before they were added to the language in Fortran90. He led the programming of RELAP5 until he retired on his 65th birthday, April 30, 1998. Code Architect of the first 19 years, Dick Wagner wrote or rewrote virtually all 250,000 original lines of RELAP5 code.

Thereafter, Dick joined Innovative System Software (ISS) and continued to work on the RELAP5 code for the rest of his career, which ended in his 80's. There is a saying, "You will never have to work a day in your life if you love your job." Dick loved his job.



Dr. George Mesina
Code Architect, Manager, Team Leader

George Mesina is the RELAP5-3D Team Leader and its Code Architect. He earned his Ph.D. in Applied Mathematics from the University of Pittsburgh in 1988 and has worked at the Idaho National Laboratory ever since as a Math/Computer Expert, Code Developer, Project Manager, and Acting Manager for TH & CFD for one year. George served as RELAP5-3D DOE-projects Project Manager and Developer from 1991 to 1997. When Dick Wagner retired in 1998, George replaced him as Code Architect. He became an INL Distinguished Scientist in 2015, and Acting Manager in 2023.

His projects and publications are in Computational Fluid Dynamics Statistical Analyses for ATR (Advanced Test Reactor) and Thermal Hydraulics. As a Department Fellow and the Code Architect of RELAP5-3D he has rewritten most of the now 350,000 lines of code in RELAP5-3D in Fortran90, Fortran 2003, and finally GNU Fortran 2008. He replaced the time advancement algorithm for accurate code coupling, augmented its ability to couple with other computer programs, significantly reduced its runtime through vector, parallel, and other optimizations, adapted it for nuclear power plant operator training simulators, worked on commercial grade dedications for licensing submittals to the NRC, added many other capabilities and features, fixed many user reported bugs; and improved code robustness and testing. Code architecture work is often unfunded, so much work was done late in the day, early in the morning, or on weekends. In fact, there is no holiday on which George was not at the lab, at least one time, working on the code. RELAP5-3D, do or die!

In one year as acting manager, George rebuilt the RELAP5 and CFD department from 3 to 12 fulltime researchers and developers, finished 5 contracts, started 3 more, and formed a consortium between

INL, Naval Nuclear Labs, and Information System Labs to collaborate for RELAP5-3D Development. He led the completion, release, and advertising of RELAP5-3D/Ver:4.5, organized and ran the 2023 International RELAP User Group Meeting (attended by a record 70 people), modernized the RELAP5-3D website, generated 225 new RELAP5 license requests a (262% increase in rate of requests), and boosted involvement and morale of everyone involved.

The 2012 INL Mentor of the Year, George also volunteers for professional and service societies. As State Deputy, George led the Idaho Knights of Columbus in 2022-24, achieving both the Circle of Honor and Pinnacle Awards. He chaired two ASME (American Society of Mechanical Engineers) Power Conferences during the challenging COVID years and was elected Chairman of the ASME Power Division in July 2024.

7.2.2. RELAP Developers

Dr. Richard Riemke

Richard Allan Riemke was born October 11, 1944, in Vallejo, CA. He received B.A. and M.A. degrees in Physiology and his PhD in Engineering Science from the University of California Berkeley. His first job after receiving his PhD was at the National Institutes of Health in Bethesda, Maryland. He wanted to move closer to home, so he started working in Idaho Falls, Idaho at the Idaho National Laboratory as a Nuclear Engineer.

When he started, Rich derived every numerical formulation in the RELAP5-3D manuals from the original governing or physical equations. His filing methods were computer precise, and he had near eidetic memory, able to tell you over the phone which filing cabinet, drawer, section, and file folder, and how far into the folder a particular derivation, user problem, or other document was. His reviews of reports, papers, and other documents were concise and exact.

He worked on RELAP5 for 30 years. Most days he came in an hour early and left an hour after quitting time. He scoured the literature and internet reports of RELAP5 code bug reports and was always setting up collections of tests and launching code runs for while he was away from the office so he could come in to find fresh calculations. Rich was methodical at debugging, tracking down the line of code, deriving the calculation or operation that it should have been, devising a solution in consultation with Dick Wagner or other teammates, and implementing it. Rich was a caring and helpful person; he would help any teammate work through their code issue when they came to him.

Rich worked on many different projects to improve the code. He was instrumental in developing the timestep backup and partial backup logic. He worked on most NRC-funded projects. His contributions to AP600, the Fortran 90 conversion, and the CRADA to develop the real time RELAP5 simulator coding were exemplary. Dedicated to making RELAP5 succeed, Rich came in every day, including weekends and holidays, except when he went on vacation to the beaches of the west coast. RELAP5 do or die!

Rich had an incredible variety of talents and interests. He played the piano & accordion, was a Lifeguard at the Rodeo Swimming Club, played trombone, was Student Conductor of the John Swett High School Band & Dance Band, was Student Body President, and graduated in 1962. He belonged to Alpha Sigma Phi Fraternity and worked for awhile during graduate school at Delta Delta Delta Sorority. Rich played trombone in the Cal Band and was Drum Major of the Band in 1965. The Cal Band performed on the Ed Sullivan Show. Ed brought Rich on the stage and introduced him. Rich was also in the U.S. Army Reserve Band as a trombone player for 5 years. Retiring in 2009, Rich moved to Huntington Beach, CA where he worked for Underwriters Laboratory, enjoyed listening to the Beach Boys, and he loved surfing. Rich passed away on March 7, 2019.

Mr. James E. Fisher

Mr. James Fisher has 28 years of experience in the analysis of complex systems including system performance modeling using a variety of computer codes and calculation methods and computer program development including 16 years of experience using the RELAP5 thermal-hydraulic systems computer code. His RELAP5 model development and analysis projects included the Westinghouse AP600 system model an advanced small modular reactor concept and a Booster Fuel model for a proposed ATR Gas Loop experiment. He taught modeling and analysis of the VVER-1000 design to engineers of various former Soviet Union countries. He developed the system model for the RBMK design which included code modification to install RBMK neutronics and worked with Russian engineers at Kursk NPP to teach them to use it to perform accident analysis. He implemented a Compressor model into the code for use with Gas Fast Reactor design. He received a master's degree in mechanical engineering from the University of Idaho in 1993 and a bachelor's degree in mechanical engineering from Colorado State University in 1978.

Dr. David Aumiller

Dr. David Aumiller received his Ph.D. in Nuclear Engineering from The Pennsylvania State University in 1996. He began working at the Bettis Atomic Power Laboratory as a senior engineer and rose through the ranks to the highest technical level as Consulting Engineer in 2021. His work has centered on thermal hydraulics of nuclear power plants, developing and improving the underlying theory, modeling, and computer programs to better design reactor systems. He has made huge contributions to improving the RELAP5 and COBRA programs. Along with Dr. Walter Weaver III, he co-authored the R5EXEC program that is used to couple RELAP5 and COBRA to other computer programs, such as Computational Fluid Dynamic codes, to solve more complex problems by splitting the portions of the reactor system into sections whereon each program is best at calculating the physics. He has published over 100 papers, reports, and journal articles on these topics and has won acclaim for his excellent work, including the Bechtel Excellent Technical Paper Award in 2002, the Bettis General Manager Awards in 2003 and 2005, and the Bettis Team Engineering Awards in 2011 and 2015. He has served the International RELAP5 User Group as Chairman from 2003 to 2005. Dave is always looking to bring others along in the field. He has mentored fellow researchers at Bettis, helped others through his service in the American Nuclear Society through presentations, chairmanships, conference organization, and leadership positions, eventually receiving the title of American Nuclear Society Fellow in 2020. He also teaches Nuclear Engineering at the University of Pittsburgh where he began as a lecturer in 2007 and became and has been an adjunct associate professor since 2008.

Mr. Douglas Barber

Mr. Barber is the Vice President & Manager of Nuclear Software Development at Information Systems Laboratories Inc. (ISL). He received an M.S. degree in Nuclear Engineering from Purdue University in 1997. He has been providing RELAP5 code maintenance support to the Nuclear Regulatory Commission (NRC) the Naval Nuclear Laboratory (NNL) and the Idaho National Laboratory (INL) for nearly 30 years.

Mr. Barber has been active in all aspects of neutronic and thermal-hydraulic software development for the last 30 years including code development verification & validation (V&V) assessment and configuration control. He has extensive experience working under ASME NQA-1-2008/09a and NQA-1-2015 compliant programs. Mr. Barber is an expert developer of TRACE PARCS and RELAP5 and is well-versed in modeling mass energy and momentum equations along with correlations and closure relations in TRACE and

RELAP5. He designed an interface module for the coupling of any system thermal-hydraulics code with any core neutronics code. This interface module design was utilized to incorporate the 3D neutronics solution based on PARCS code into the NRC codes RELAP5 and TRACE.

Mr. Barber has specialization in real-time reactor simulation numerical analysis and parallel programming. Mr. Barber was one of the original developers of PARCS in 1996 while a graduate student at Purdue University and has been involved with numerous updates to the code since that time. During his time at Purdue University, he designed a Preconditioned-Krylov Subspace linear solver for the Coarse Mesh Finite Difference problem in hexagonal geometry and adapted this solver for application on shared-memory multiprocessors. He implemented this advanced technique within the framework of the spatial kinetics code NESTLE including the NESTLE solver within RELAP5-3D. In addition, he served as a consultant to the INL and Science Applications International Corporation (SAIC) with assistance given in the areas of spatial kinetics and parallel programming using OpenMP.

Mr. Barber furthermore has substantial experience in parallel programming on shared and distributed memory multiprocessors using both threads (e.g. OpenMP and MPICH the high-performance shared-memory version of MPI) and message passing (e.g. MPI and PVM). He parallelized the nodal kinetics solver in RELAP5-3D using OpenMP. The biggest challenge in developing parallelized software is in the testing and debugging needed to quickly identify when parallelization errors are occurring. As part of this project Mr. Barber and Mr. Lance Larsen (ISL) developed a testing system that would compare single-threaded and multi-threaded calculations in real-time and identify specific subroutines and sections where results began to diverge between the two runs. This allowed the developers to then address any thread memory management or synchronization issues causing the divergence.

Mr. Barber has been responsible for RELAP5/Mod3.3 (NRC) code development maintenance configuration control and V&V since 2003. For over 18 years he has also been active in NNL contracts related to both the NUPAC/RELAP5-3D merge and the RELAP5-3D nodal kinetics upgrades. Mr. Barber is currently the project manager for the latest merge of RELAP5-3D and NUPAC for NNL. Under past NNL projects Mr. Barber was the principal investigator responsible for several nodal kinetics upgrades in RELAP5-3D. These updates included:

1. (Upgrading the steady-state solver to utilize a pre-conditioned BiCGSTAB solver in place of the old LSOR solver
2. Implementing a GMRES solver for the coarse mesh finite difference problem
3. Implementing the TPEN nodal method for hexagonal geometries
4. Implementing a rod cusping correction treatment
5. Extending the number of energy groups in the solution from 2 to 4
6. Implementing an algorithm for computing the separate reactivity effects from various feedback mechanisms
7. Redesigning the nodal kinetics memory structure
8. Implementing an asynchronous time step control algorithm to advance the thermal-hydraulics and neutronics independently
9. Parallelizing the nodal kinetics using OpenMP and
10. Implementing three parallel direct linear solvers (PARDISO MA41 and SuperLU_MT) for the finite difference equations.

Mr. Lance Larsen

Mr. Larsen is Principal Engineer and member of the Nuclear Software Development team at Information Systems Laboratories Inc. (ISL). He received an M.S. degree in Mechanical Engineering from Purdue University with an emphasis is nonlinear dynamical systems. He has assisted in the development of RELAP5-3D for the Naval Nuclear Laboratory (NNL) and Idaho National Laboratory (INL) for 15 years.

Mr. Larsen came to the nuclear industry with a background in software development that emphasized best practices for developing testable and maintainable code. He has worked to promote code development best practices that make code easier to understand debug maintain and test. Among the best practices he has promoted are indicating clearly the units for variables that have units as well as replacement of "magic" numbers (integers with a specific interpretation) with named constants that clarify the intent of the value. Such practices have had tangible benefits. For example, several years ago the water properties tables used a different set of interpolation routines than other fluids. Mr. Larsen updated RELAP5-3D such water used the same interpolation routines as other fluids. The interpolation code had much less testing than the water interpolation code and there were errors. To clarify the logic Mr. Larsen updated the code from using state calculations with magic numbers (i.e. 's(5) = s(4) + s(2)*s(3)') to state calculations with named constants (i.e. 's(h_) = s(u_) + s(p_)*s(v_)') where the intent of the calculation became clear – calculating enthalpy in this case. With the improved code he identified and fixed several interpolation errors.

Mr. Larsen has introduced innovative and powerful techniques for tackling difficult code challenges. For example, Doug Barber and Mr. Larsen were tasked with restoring the PARCS parallel computing capability which had not been maintained in several years. Due to the notorious difficulty of debugging parallel logic there was skepticism that this could be accomplished in any reasonable time frame. Mr. Larsen proposed and successfully developed a system that allowed two instances of a model one single threaded and one multithreaded to run side by side and communicate the inputs and outputs from specific routines. Results were compared automatically and the specific time step and location where differences occurred were identified and corrected with minimal effort. Implementation of an innovative strategy converted an intractable problem into a problem that was relatively simple to solve.

Dr. Glenn Roth

Dr. Roth received a PhD Mechanical Engineering from the University of Idaho in 2017. He joined ISL in 2014 and has worked on contracts with the NRC the DOE NuScale Power and BWXT. He has 20 years of experience in reactor analysis codes and is a specialist in the use of RELAP5-3D, TRACE, NUPAC (RELAP5), and KIPAC (TRAC).

Dr. Roth's Ph.D. research developed models for additional fluid fields in the RELAP5-3D system code. He selected models for droplet and bubble fields from available literature and developed the equations for those fields in a form that could be implemented into the RELAP5-3D solution scheme. The addition of these fields to the RELAP5-3D code would improve the fidelity of code predictions for two-phase flow regimes. Dr. Roth also researched suitable data sets for further model validation and development.

Dr. Roth was employed at Knolls Atomic Power Laboratory (KAPL) at the time that the Naval Laboratories selected NUPAC as the code of choice for reactor systems analysis, which is based on RELAP5-3D. While at KAPL, Dr. Roth was Cognizant Engineer for the NUPAC Code and worked with developers to troubleshoot and develop that code, including adding a new physics model and then evaluating the source code to verify the new model had been implemented correctly.

While employed at Idaho National Laboratory (INL) Dr. Roth updated the RELAP5-3D decay heat models to include the ANS 5.1 2005 standard. He verified the models against hand calculations and experimental results. He was part of the team that validated RELAP5-3D using LOFT test data for mPower.

Dr. Roth has worked on the NRC MONCC project in support of the development and maintenance of the NRC TRACE and RELAP nuclear system codes, implemented code updates and bug fixes for the RELAP5/Mod3.3 code, and has worked with users to resolve issues and provide training and technical support for RELAP5/Mod3.3.

While at ISL Dr. Roth has helped make updates to the RELAP5-3D code including updates to the CCFL and CHF models. Additionally, Dr. Roth was a key project member to allow dynamic switching between kinetics solution methods to improve calculation convergence. He has evaluated experimental data related to a dissolved gas model in RELAP5-3D and developed a plan to evaluate that model based on test data.

7.2.3. RELAP Analysts

Mr. Nolan Anderson

Nolan Anderson has worked with the RELAP5-3D code for over 18 years. He worked at INL for nearly 17 years as a software developer and the code custodian. He has been involved in many upgrades to the code in that time. He is currently the RELAP5-3D code custodian at Terrapower. He has a master's degree in Nuclear Engineering with an emphasis on thermal hydraulics.

Mr. Paul Bayless

Paul Bayless used RELAP5 for nearly his entire 39-year career at Idaho National Laboratory. In 1981 the code was used to help size flow instrumentation for a small-break loss-of-coolant experiment in the Loss-of-Fluid Test facility. About a year after that Paul built and used nuclear power plant models to simulate actual and potential transient events and accidents.

Paul analyzed potential severe accidents in light-water reactor power plants and was the first user of the combined SCDAP/RELAP5 code. This code was used to perform phenomenological analyses for the Advanced Test Reactor (ATR) Level 1 and 2 Probabilistic Risk Assessments. RELAP5 was later used in thermal-hydraulic analyses for the MAPLE isotope production reactor.

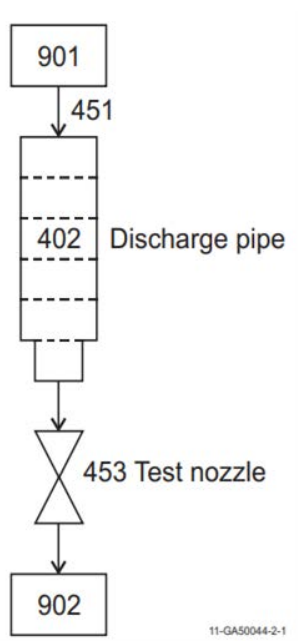
RELAP5 simulated accident scenarios in operating power plants and those scenarios were used to conduct realistic emergency drills in the Nuclear Regulatory Commission (NRC) headquarters and regional offices. Eventually RELAP5 output was provided directly to the NRC's emergency response computer representing what it would receive during an actual plant event.

Paul developed input models and performed transient analyses for the Westinghouse AP600 reactor in support of the NRC's licensing review of that design. He also supported efforts by Babcock and Wilcox and NuScale to qualify RELAP5-3D as a licensing tool for analyzing their small modular reactors.

Paul developed and taught the exercises for the 4-week long "Reactor Safety Analysis using RELAP5" training program developed for the Department of Energy's International Nuclear Safety Program which taught analysts from the former Soviet Union the basics of how to analyze their power reactors. He was involved in the assessment of the RELAP5/MOD3 code for applications to VVER and RBMK reactor designs. That initial training class evolved into the one-week basic and advanced RELAP5 training classes for which Paul developed a few lectures. Eventually Paul travelled to 11 foreign countries to provide RELAP5 training.

He was the project manager and lead analyst for the RELAP5-3D developmental assessment which is documented in Volume 5 of the code manual.

The last 15 years of Paul's career was heavily focused on gas-cooled reactors. For the Generation IV reactor program, he performed scoping thermal-hydraulic analyses for the prismatic core version of the very high temperature reactor (VHTR) using RELAP5-3D to investigate the steady state and transient behavior of the reactor and the containment cooling system. For the Advanced Reactor Technologies program, he performed analyses supporting the design and operation of the High Temperature Test Facility (HTTF) at Oregon State University and worked with Japan Atomic Energy Agency counterparts to develop RELAP5-3D models of their High Temperature Engineering Test Reactor for code assessment. He also performed thermal-hydraulic analyses for a gas-cooled test reactor design.



Marviken Nodalization Diagram

Mr. Cliff Davis

Mr. Cliff Davis used RELAP5 to perform thermal-hydraulic analyses of a wide range of nuclear systems and transients. These nuclear systems included candidate Generation IV reactors including reactors cooled by supercritical light water lead bismuth and carbon dioxide all major types of commercial U.S. reactors and other systems including Westinghouse Babcock and Wilcox and Combustion Engineering pressurized water reactors the AP600 the Advanced Test Reactor the Transient Test Reactor the production reactors at the Savannah River Site VVER-1000 reactors and related experimental systems.

His RELAP5 development activities included:

- Generation of fluid property files including sodium, carbon dioxide, molten salts, new helium xenon and helium-xenon.
- The noncondensable gases oxygen, carbon dioxide and carbon monoxide.
- Led the development of the polate utility that allows users and other codes to access the RELAP5 fluid property files.
- Added the Gnielinski heat transfer correlation into the code.
- Added several heat transfer correlations to support the application of the code to supercritical water applications.
- Extended Card 1 Option 11 to allow the code to simulate supercritical carbon dioxide.
- Developed component models that are included in the ATR's licensing version of the code.

Additionally, he consulted with Mitsubishi Heavy Industries in the development and defense of their Appendix K version of RELAP5. Aided mPower and NuScale in support of their licensing applications with RELAP5

Cliff received a B.S. in Mathematics (Probability and Statistics minor) in 1973 (Utah State University) and a M.S. in Mechanical Engineering in 1981 (University of Idaho).

7.2.4. RELAP Managers and Project Managers

Mr. Gary W Johnsen

Born on Thanksgiving Day 1944, Gary graduated from the Polytechnic Institute of Brooklyn (now NYU) in 1966 in Mechanical Engineering. He took a position with the Grumman Aerospace corporation working on the Apollo Program Lunar Module project in 1968. He relocated to Houston's Manned Spacecraft Center (now Johnson Space Center), where Gary worked on a series of final tests of the Lunar Module leading to the subsequent moon landings. In 1970, he was awarded the Presidential Medal of Freedom by President Richard Nixon as a member of the Apollo 13 Mission Operations Team.

In early 1971, he accepted a position with Martin Marietta Corp in Denver, Colorado, where he worked on the thermal performance analyses for the Skylab and Space Shuttle programs. In August 1974, the family moved to Idaho Falls, where Gary accepted a position with Aerojet Nuclear Co. at the National Reactor Testing Station, now the Idaho National Laboratory. It was here that he would embark on a long career in nuclear reactor safety analysis.

Throughout the 1970s, Gary was engaged in computer simulation and analysis of light water reactor accident behavior in support of the U.S. Nuclear Regulatory Commission and the Department of Energy, utilizing the RELAP4 computer code. During the 1980s he managed the planning and analysis of experiments in the Semiscale reactor simulation facility.

In the early 1990s, he joined the team working on the development of the RELAP5 computer code. He managed the RELAP5 department until 1997 and as the project manager until his retirement in 2007. Gary was a co-creator of the International RELAP5 User Group (IRUG).

Dr. Richard Schultz

Richard R. Schultz, Ph.D., P.E. has focused on the satisfactory completion of the work required to demonstrate the adequacy of the RELAP5-3D computer code to perform licensing calculations acceptable to regulatory agencies for both online and advanced nuclear reactor systems. The code must be demonstrated capable of adequately calculating the highly-ranked (and supporting medium-ranked) phenomena that directly influence the figures-of-merit, e.g., the peak fuel cladding temperature, that are linked to the subject plant margin of safety and/or acceptance for the most challenging operational and accident scenarios. These focus areas include:

1. Identification of the figures-of-merit and highly-ranked phenomena via PIRTs (phenomena identification and ranking tables);
2. Ensuring the models in the code adequately represent the physics that describe the behavior of these phenomena,
3. The scaling, design, and operation of separate-effects and integral-effects experimental facilities to generate the necessary data to both validate the code and adequately demonstrate that the data from the scaled facility are scalable to the prototype reactor;
4. Demonstrate that the code can adequately calculate the behavior and margin of safety of the figures-of-merit for the reactor; and
5. Provide the necessary documentation and support.

These focus areas form the basis for demonstrating the adequacy of RELAP5-3D to perform licensing calculations following the guidance provided by the U.S. Nuclear Regulatory Commission in their Regulatory Guide 1.203. Dr. Schultz has participated extensively in all of these activities and has been a user of RELAP5 since 1980.

Some of these projects generated the validation data used for the RELAP5 Developmental Assessment, Volume 3 of the manuals that is automatically generated with each public code release. The development and use of discretized models of nuclear systems to perform validation calculations, for the benefit of users, was documented in Volume 5 of the manuals by Don Fletcher and Dick. Dick was either a member or the Principal Investigator of the following projects:

- Marviken critical flow tests, 1979-1981
- ROSA_IV SBLOCA experimental program, 1983-1988
- RELAP5-3D Appendix K, 1999-2004
- ROSA-AP600 Test Planning, Analysis, & Support Project, 1999-1993;
- "BETHSY Experiment Evaluation & Code Assessment", 1990-94 PI Project.
- "International Code Applications and Assessment (ICAP)" Project centered on defining V&V matrices, analyses, and developmental studies for the RELAP5 and TRAC-BWR advanced TH codes, 1986-90 PI
- Methods development for analyzing the behavior of gas-cooled reactors, 2005-2014.

His earliest participation in PIRT studies was focused on the Westinghouse AP600 and AP1000 LWRs during the 1990s. Both of these designs have passive emergency core cooling systems. These studies were performed to provide the basis for scaling studies used to design the ROSA-AP600 1/30-scale, full-height test facility—located on site at what is now the Japan Atomic Energy Agency (formerly known as the Japan Atomic Energy Research Institute—JAERI) in Tokai-mura, Japan. Dr. Schultz was the U.S. NRC Principal Investigator (PI) of the effort to generate validation data—obtained by conducting 24 experiments at JAERI from 1994 to 199715--used for qualification of the RELAP5 software.

Dick Schultz serves the engineering community in professional society activities, especially twice being the conference chair of the International Conference of Nuclear Energy. He is a Founder of ASME Journal of Nuclear Engineering and Radiation Science, 2015 and a founding member of the ASME V&V30 Code Committee: V&V in Computational Simulation of Nuclear Systems Thermal Fluids Behavior. In 2008, he was elected an ASME Fellow. In 2012, he received INL's Individual Lifetime Achievement Award in Science and Technology (INL's highest award) and The George Westinghouse Gold Medal for eminent achievement in power engineering.

Dr. Piyush Sabharwall

Dr. Piyush Sabharwall is a Distinguished staff scientist in the Nuclear Science and Technology (NS&T) Directorate at Idaho National Laboratory (INL) with more than 20 years of research and development (R&D) experience. On June 24, 2024, he became the newest manager of the department containing the RELAP5-3D program. His Thermal Fluid Systems Methods and Analysis Department focuses on non-nuclear experimental test beds, thermal hydraulic design and analysis using RELAP5 and MOOSE based tools.

He is a recognized expert in the design and safety of nuclear systems, with specialized knowledge in thermal-hydraulics, heat transfer, compact heat exchanger design, and thermal energy systems. Dr. Sabharwall currently serves as the technical area lead on the U.S. Department of Energy–Office of Nuclear Energy's (DOE–NE's) Microreactor R&D Program (MRP), the Holtec Small Modular Reactor (SMR)-300 Advanced Reactor Demonstration Project and has led the development of a gas-cooled cartridge loop for the Versatile Test Reactor (VTR). In his current role as the department manager for Thermal Fluid Systems Methods and Analysis, he leads the thermal hydraulics research in the Nuclear Science and Technology directorate supporting and guiding development of system code such as RELAP5, MOOSE based tools such as SAM and Pronghorn and leads the development of thermal-hydraulic validation experiments assisting with accelerated development and deployment of advanced reactor technologies. He has consulted for industries, both nationally and internationally, building technical expertise with an additional focus on market studies and economic viability to rebuild U.S. nuclear industrial infrastructure and position the U.S. industry to continue its leadership in the global energy market.

Dr. Theron Marshall

Theron Marshall served as the manager of C130, the Thermal Fluid Systems Methods and Analysis Department, from 2021 to 2023. He began his association with RELAP5 as an analyst. He used RELAP5-3D for gas-cooled fast reactor (GCFR) designs in the early 2000's and the LEU (Low Enriched Uranium) conversions of research reactors in the mid-2020's. Both efforts proved a testament to the code's capabilities and robustness.

Theron has a Ph.D. in nuclear engineering and has participated in nuclear engineering research and design since graduating from Rensselaer Polytechnic Institute. He is currently a member of the INL technical staff.

Dr. James Wolf

Jim Wolf was one of the founding members of the International RELAP5-3D User Group and a driving force behind the RELAP5 real-time CRADA that developed the simulator version called RELAP5/RT. He managed RELAP5-3D for fifteen years in the time span from 1998 to 2021. He oversaw the conversion to Fortran90, PVM-coupling, integer time-stepping, code verification, and adaptation to many non-aqueous fluids. Jim was the organizer of International RELAP User Group and the annual INL-Bettis meeting for 20 years.

Ms. Amy Francisco

Amy is the Integrated Data & Analysis manager at Naval Nuclear Laboratory. She assumed this role in 2024. Amy is responsible for leading the Data Science Data Engineering and Integrated Reactor Engineering teams. Her subdivision is responsible for driving Digital Transformation and integrated reactor engineering initiatives at NNL.

Prior to this role Amy was the Engineering Quality manager in the Quality Performance & Improvement department where she was responsible for maintaining and providing oversight of the NNL Quality Assurance Program.

Amy developed the two-phase pump theory and coding for RELAP5-3D. She was also the manager in charge of the INL contract for NNL for two years.

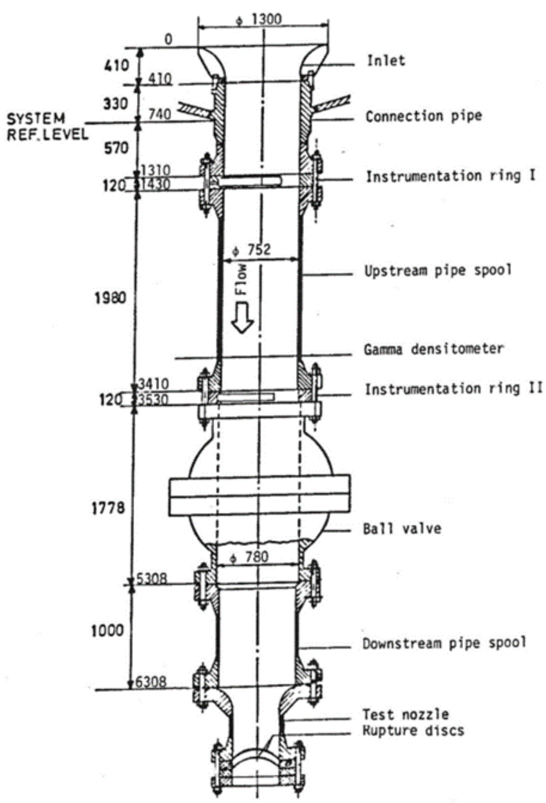
Amy joined the company in 1998 as an Intern. She held positions of increasing responsibility in quality reactor technology fleet support code development plant analysis and reactor and plant safety analysis. She also advised the NNL Reactor Safety Council and the Nuclear Safety Advisory Council.

Amy has a bachelor's degree in chemical engineering from the University of Pittsburgh and completed the Naval Nuclear Laboratory's Reactor Engineering School. Amy serves as the NNL Dissenting Opinion Support Team lead working to improve NNL culture surrounding conflict resolution.

7.2.5. The RELAP5 Team

Dr. Paolo Balestra

Dr. Paolo Balestra obtained his master's degree in energy engineering (2012) and his Ph.D. in energy and environment with focus on nuclear engineering (2017) at the "La Sapienza" University of Rome. He is currently the Methods Lead for the Advanced Reactor Technology-Gas Cooled Reactor (ART-GCR) DoE Program at INL. He is also work package manager of INL's High Temperature Gas-cooled Reactor (HTGR) area of the multiphysics section of the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program. His research efforts are focused on two areas: the analysis and optimization of the new generation of advanced reactor design and the development and validation of dedicated advanced reactor designs simulation tools.



Marviken Critical Flow Diagram

Dr. Victor Coppo-Leite

Dr. Victor Coppo Leite is a researcher at Idaho National Laboratory who earned a Ph.D. in Nuclear Engineering (2023) from Pennsylvania State University. He also holds a Bachelor of Science and a Master of Science in nuclear engineering from the Federal University of Rio de Janeiro. His interests include incorporating machine learning methods into thermal hydraulic problems and multi-physics modeling. Currently he applies his expertise to contribute to the development of RELAP5-3D while also engaging in activities to maintain the code and conduct analyses.

Mr. Brandon Cox

Brandon Cox is an engineer and software developer in the Thermal Fluid System Methods & Analysis Department at Idaho National Laboratory. He is currently a RELAP5-3D developer and analyst. In this capacity he participates in various RELAP5-3D updates and development activities and corrects user errors. As the RELAP5-3D code custodian he supports licensing requests prepares new RELAP5-3D versions and completes verification and validation testing. He also performs engineering calculations and analyses using RELAP5-3D and CFD.

He graduated with a B.S. of Chemical Engineering (2019) from Brigham Young University. He has experience in developing engineering software and computational fluid dynamics (CFD). He has experience using Python Fortran Linux and OpenFOAM. Outside of work he enjoys spending time with family and many outdoor activities including hiking hunting fishing and photography.

Dr. Ramiro Freile

Dr. Ramiro Freile is a computational scientist in the Thermal-Hydraulics Experiment and Modeling & Simulation at Idaho National Laboratory (INL). His background is in turbulent computational fluid dynamics (CFD) and multiphysics and he is actively involved in the development of simulation codes based on the open-source MOOSE framework and performing TH analysis in advanced reactors using these tools. He has a PhD in Nuclear Engineering from the University of Texas A&M College Station and worked at INL as a graduate fellow for about 2 years prior to joining INL as staff in 2024. He has performed extensive research in areas including turbulence modeling CFD multi-phase solidification modeling Lattice Boltzmann simulations modeling and multi-physics simulations for advanced reactors.

Dr. Robert Kile

Dr. Robert "Robby" Kile earned a B.S. in Nuclear Engineering (2018) from Purdue University and a M.Sc. (2020) and Ph.D. (2023) in Nuclear Engineering from the University of Tennessee Knoxville. He works at Idaho National Laboratory in the Advanced Reactor Technology and Design department as a nuclear engineer. His research interests include neutronics and thermal hydraulics analysis of advanced reactors and code validation for advanced reactor applications. Dr. Kile is involved in the OECD/NEA Thermal hydraulic code validation benchmark for high temperature gas-cooled reactors using HTTF data as the coordinator for the Depressurized Conduction Cooldown Problem. He is a member of the American Nuclear Society (ANS) and a member of the executive committee for ANS's Nuclear Installations Safety Division.

Dr. Henry Makowitz

Dr. Henry Makowitz is a classically trained theoretical physicist and computational scientist as well as an experienced research scientist in a number of basic and applied research disciplines.

He is currently a Consultant to the Idaho National Laboratory RELAP5-3D Group. At INL he is developing a Four Pressure theoretical model suitable for future RELAP5-3D implementation and enhancing RELAP5-3D code and manual documentation. He has written several INL documents including a formal INL Report a paper submitted to NS&E and a US DOE Proposal currently under review by US DOE NNSA NE and ER organizations on the Four Pressure INL Efforts.

Dr. Makowitz received a B.S. in Physics (1973) and a M.Sc. in Mechanical Engineering (1978) at State University of New York-Albany and a Ph.D. in Theoretical Physics (1988) from the University of Texas-Austin.

His RELAP5 experience includes mostly parallel processing development for the Mod2 Code at Cray Research Inc. and the INEL/INEEL. Developed a complete RELAP5-Mod2 implementation suitable for parallel processing on Cray X-MP Y-MP and C-90 series machines. Obtained 10x speedups on 16 cpu C-90 machines. Published extensively about this work and presented my results at National and International Meetings of various scientific societies including the M&C of the ANS.

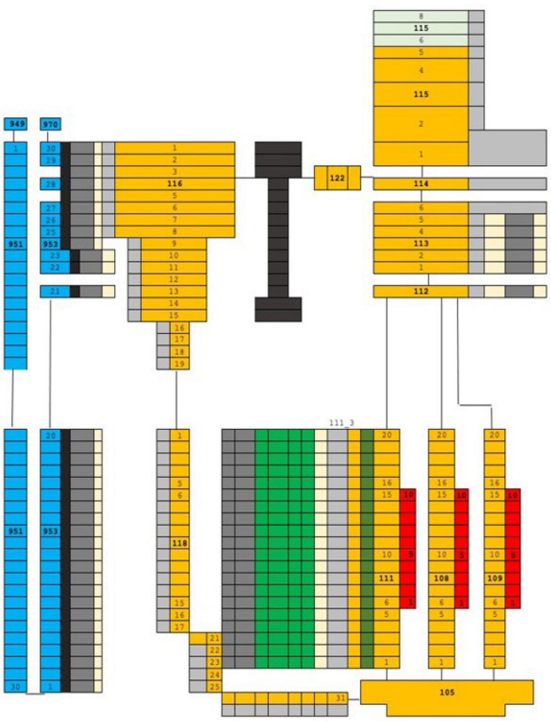
Additionally, he studied the Ill-Posedness issue in the theory from the point of view of the Special Theory of Relativity and authored an INEL report and several ANS Meeting short Papers and Abstracts. Concluded that the Ill Posed issue was due to a violation of space and time transformation laws per Galilean Relativity vs Special Relativity.

Dr. Sinan Okyay

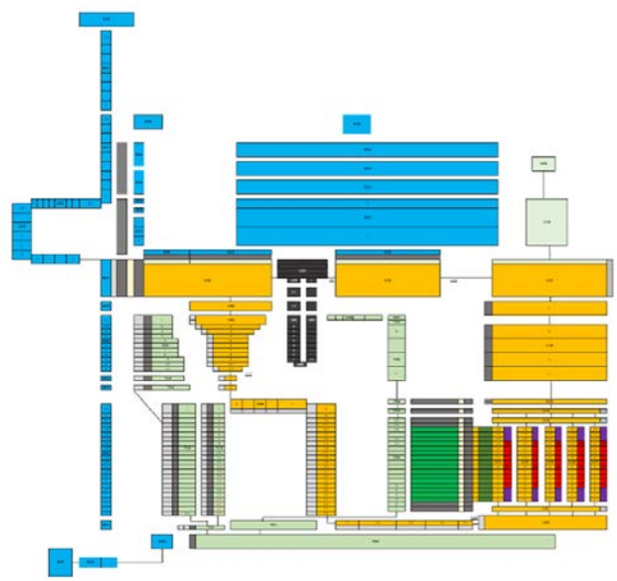
Sinan Okyay is a graduate fellow at Idaho National Laboratory. He is a Ph.D. candidate in Nuclear Engineering Department of The Pennsylvania State University. He holds a Bachelor of Science in the same field from Hacettepe University. His interests include computational fluid dynamics (CFD) high-fidelity thermal-hydraulic simulations and numerical simulations of natural convection systems. Currently he is working on high-fidelity modeling of the reactor cavity cooling system for a high-temperature gas-cooled reactor.

Dr. Carlo Parisi

Dr. Carlo Parisi is a scientist with over 20 years of experience in academics and government laboratories. He received a M.Sc. (2004) and Ph.D. (2008) in Nuclear Engineering at University of Pisa Italy. His area of expertise includes thermal hydraulics system design and analysis neutronics. Since October 2015 he is a scientist at the Idaho National Laboratory USA. His current activities include thermal hydraulic design and analysis for several projects: MARVEL Microreactor, Thermal Test Reactor Capabilities, TREAT Sodium Loop Experiment, and Light Water Reactor Sustainability. He is a member of the INL RELAP5-3D code team where he provides technical contribution to the code assessment application and development. He is member of the American Nuclear Society since 2004.



PCAT RELAP5-3D Nodalization Diagram



MARVEL RELAP5-3D Nodalization Diagram

Ms. Connie Stevens

Ms. Connie Stevens works as the RELAP5-3D project coordinator. She received a Master of Legal Studies from the University of Utah S.J. Quinney College of Law (2023) and a B.Sc. in Applied Management (2022) from Brigham Young University-Idaho. Connie's work is currently focused on building and strengthening relationships with domestic and international RELAP5-3D license holders, leading the organization of the annual International RELAP5 User Group (IRUG) meeting, and providing program management support to the department manager and RELAP5 team.

Dr. Mauricio Tano-Retamales

Dr. Mauricio Tano-Retamales is a Staff Scientist in the Thermal Fluids Systems and Methods Analysis group at Idaho National Laboratory (INL). He currently leads efforts for high- and intermediate-fidelity modeling and simulation of multiphysics phenomena in advanced nuclear reactors which include the development of models coupling between neutronics thermal-hydraulics thermomechanics salt chemistry and corrosion. Additionally, Dr. Tano-Retamales is an active developer of INL's coarse-mesh computational fluid dynamics (CFD) code Pronghorn and INL's MOOSE-based subchannel code for advanced nuclear reactors. Dr. Tano-Retamales leads the coupling between RELAP5-3D and CFD codes and performs nuclear reactor analyses using RELAP5-3D.

Dr. Ishita Trivedi

Dr. Ishita Trivedi works as a computational scientist in the Advanced Reactor Design & Technology at Idaho National Laboratory (INL). She received her PhD in Nuclear Engineering from North Carolina State University focusing on Uncertainty Quantification and Propagation Methods for Safety Analysis of Lead-cooled Fast Reactors. Ishita also has a M.Sc. and B.Sc. in Nuclear Engineering from North Carolina State University and Pennsylvania State University respectively. At INL Ishita is involved in several advanced reactor multi-physics analysis projects including leading the MARVEL criticality benchmark development nuclear thermal propulsion reactor modeling and simulation among other projects. Ishita also works in software quality assurance validation & verification of INL applications.

Dr. Jan Vermaak

Jan Vermaak joined INL in 2022 as a Senior Nuclear Multiphysics Engineer. He has 14 years of diverse experience in reactor engineering ranging from nuclear safety analysis to managing reactor engineering at the Texas A&M University TRIGA reactor. He holds a bachelor's degree in mechanical engineering and a master's and PhD in nuclear engineering. Jan's current work is focused on code development and analysis in the fields of neutronics computational fluid dynamics (CFD) thermal-hydraulics mechanical stress and nuclear fuel performance.

45th ANNIVERSARY RELAP5-3D
Reactor Excursion & Leak Analysis Program

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Idaho National Laboratory

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