INL/EXT-19-56151
Revision 1

Light Water Reactor Sustainability Program

Assessment of verification and validation status - RELAP5-3D and RAVEN

October 2019

U.S. Department of Energy

Office of Nuclear Energy

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October 2019

Prepared for the
U.S. Department of Energy
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EXECUTIVE SUMMARY

The United States nuclear industry is facing a strong challenge to ensure maximum safety while enhancing economic benefit. Safety is a key parameter to all aspects related to light water reactor (LWR) nuclear power plants (NPPs), especially cost savings. Since the goal is to extend the lifetimes of these NPPs, the traditional deterministic safety concept may not guarantee a current economic asset. The Light Water Reactor Sustainability (LWRS) Program has been promoting a wide range of research and development (R&D) in this field to maximize the safety, economics, and performance of these NPPs through improved scientific understanding.

One of the best practices to achieve this goal is to identify and optimize safety margins, which can lead to cost reduction. To do this, under the LWRS framework, the Risk-Informed Systems Analysis (RISA) Pathway will focus on the optimization of safety margin and minimization of uncertainties to ensure both safety and economics at the highest level. The RISA Pathway will provide enhanced capabilities for analyzing and characterizing LWR systems performance by developing and demonstrating methods, tools, and data to enable risk-informed margins management (RIMM).

The goals of the RISA Pathway are twofold: (1) deploy the risk-informed tools and methods that enable better representation of safety margins and factors that contribute to cost and safety; and (2) conduct advanced risk assessment applications with industry to support margin management strategies that enable more cost-effective plant operation. The tools and methods provided by the RISA Pathway will support effective margin management for both active and passive safety systems, structures, and components (SSC) of an NPP.

The tools and methods used in the RISA Pathway should have high confidence and highest technical maturity for and implementation to industry at its current setting. They should also have a capability to support risk-informed decision making for both probabilistic and deterministic elements of safety. The RISA Pathway will, therefore, perform a comprehensive assessment of verification and validation (V&V) status of RISA Toolkit to enhance credibility RISA Toolkit which be used by industry.

This report summarizes assessment technical maturity of RISA thermal-hydraulics accident analysis code RELAP5-3D and multi-purpose probabilistic risk analysis tool RAVEN including, V&V status, specific information of the tool such as capability and features, quality assurance program, developer/independent V&V record, separation/integral tests history, user documents, and feedback.

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ACRONYMS

3D	three-dimensional
AOO	Anticipated Operation Occurrence
ATF	Accident Tolerant Fuel
BEPU	Best Estimate Plus Uncertainty
BWR	Boiling Water Reactor
DBA	Design Basis Accident
DNC	Dynamic Natural Convection
DOE	U.S. Department of Energy
EPRI	Electric Power Research Institute
ER	Equipment Reliability
ESFAS	Engineered Safety Feature Actuation System
FLEX	Diverse and Flexible Coping Strategy
FY	Fiscal Year
GOTHIC	Generation of Thermal-Hydraulic Information for Containment
I&C	Instrumentation and Controls
INL	Idaho National Laboratory
IR	Incident Report
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MAAP	Modular Accident Analysis Program
MOOSE	Multiphysics Object-Oriented Simulation Environment
MP-BEPU	Multi-Physics Best Estimate Plus Uncertainty
NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Program
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
PDF	Probability Density Function
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
R&D	Research and Development
RAVEN	Risk Analysis in a Virtual Environment
RCIC	Reactor Core Isolation Cooling

RD&D	Research, Development, and Demonstration
RIA	Reactivity Initiated Accident
RIMM	Risk-Informed Margin Management
RI-MP-BEPU	Risk-Informed Multi-Physics Best Estimate Plus Uncertainty
RISA	Risk-Informed Systems Analysis
RISC	Risk-Informed Safety Categorization
RPS	Reactor Protection System
SLR	Second License Renewal
SNL	Sandia National Laboratories
SSC	System, Structure, and Component
TTEXOB	Terry Turbine Expanded Operating Band

1. INTRODUCTION

1.1 Background

The risk-informed tools and methods, so-called RISA toolkit, to be used in industry needs high level of verification, validation, and uncertainty quantification to give maximum credibility for safe nuclear power plant operation. Since the tools and methods being used in RISA Pathway needs to have highest technical maturity and could be used in industry immediately, the comprehensive assessment of verification and validation (V&V) status level is one of most important tasks for successful industry deployment of risk-informed tools and methods [1].

Most of tools have developer and/or user V&V program to improve quality of certain tool. However even many of V&V programs are reported some of the tools still need to confirm its technical maturity based on philosophy of the risk-informed, thus, application for Probabilistic Risk Assessment (PRA). The RISA Pathway will therefore aim following goals to assure RISA toolkit quality:

- Define requirements based on risk-informed concept
- Investigate and review development and V&V status for technical maturity assessment.
- Identify technical gap and propose improvement to meet RISA toolkit requirements

First part of the work will collect and summarize available information of selected RISA Toolkit. This will include list of documents and accessibility of the V&V records for each version of the software, if necessary. The selected toolkit will then evaluate its maturity level. Based on risk-informed concept, capability and/or applicability of PRA method will be the main requirement as RISA toolkit. The requirements will be set by using Requirement Traceability Matrix (RTM). The RTM captures the requirements from user and developer of the project or software. The high-level requirements will be set based on need as RISA toolkit. The importance of each requirements will be evaluated by Phenomena Identification and Ranking Technology (PIRT). The PIRT is a systematic way of gathering information from experts on a specific subject, and ranking the importance of the information, in order to meet some decision-making objective to determine what has highest priority for research on that subject. Finally, degree of the maturity will be measured by Technology Readiness Level (TRL). Developed by National Aeronautics and Space Administration (NASA), the TRL is a method for estimating the maturity of the technologies during the development and acquisition phase of certain technology. A total of nine level are set for RISA Pathway from low level (1~4) to high (5~9) which represents from research and development (R&D) status to ready for industrial use.

The industry application pilot demonstration projects (pilot project in-short) are main features of the RISA Pathway in order to give clear vision on risk-informed margin recovery to U.S. nuclear industry and decision makers. The pilot projects focus on specific scope of phenomena, components, and simulation capabilities needed to address the given issue area. As a part of these applications, refinement of the associated methods and tools would continue at a reduced level of effort compared to the effort associated with the RISA toolkit development. However, not all toolkit suitable for use in RISA pilot project, thus, certain tool may need additional development. The RISA Pathway will therefore identify technical gap of target toolkit and will propose additional development to meet requirements as RISA toolkit.

As the development and capabilities of the RISA toolkit progresses, the pathway will collaborate with industry to determine how to transition the RISA toolkit to a user-supported community of practice. The assessment of V&V status project will support smooth industry deployment of RISA toolkit including planning for lifecycle software management issues such as training, software quality assurance, and development support.

1.2 RISA Industry Pilot Demonstration Project

On May 15, 2018, Idaho National Laboratory (INL) organized a special workshop with delegations from major U.S. nuclear utilities to discuss and develop the “Pilot Demonstration Projects” under the RISA Pathway. Eight projects were identified based on comprehensive analysis on rising issues from the U.S. nuclear industry. Table 1 shows how each pilot demonstration project relates to each RISA research and development (R&D) focus area. These are the most relevant industry topics that can potentially impact plant operations in a significant way making them interesting and relevant applications for the RISA toolkit. The RISA Pathway will continue to communicate with various U.S. nuclear industries to collect issues and develop additional pilot demonstration projects.

Table 1-1 Pilot demonstration projects related to RISA R&D focus areas.

RD&D Focus Areas	Pilot Demonstration Projects
Enhanced resilient NPP concepts	RISA-Enhanced Resilient Plant Systems Enhanced Operation Strategies for System Components.
Cost and risk categorization applications	Risk-Informed Asset Management Plant Health Management.
Margin recovery and operation cost reduction	Enhanced Fire Probabilistic Risk Assessment (PRA) Modernization of Design Basis Accidents Analysis with Application on Fuel Burnup Extension Digital Instrumentation and Control (I&C) Risk Assessment Plant Reload Process Optimization

1.3 List of RISA toolkit

Based on current and potential RISA pilot projects, Figure 1-1 shows the list of computational software proposed to be used and deployed to the industry for risk-informed margin management. Most of tools are completely developed and currently using in various industry. However, some tools are still under development and maybe needs additional V&V activities.

A brief description of potential RISA toolkit are as follows [1].

- **BISON:** BISON is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including LWR fuel rods, tristructural isotropic (TRISO) particle fuel and metallic rod and plate fuel. It is an advanced fuel performance code being developed at Idaho National Laboratory (INL) and offers distinctive advantages over FRAPCON/FRAPTRAN, such as three-dimensional (3D) simulation capability, etc. BISON solves the fully-coupled equations of thermomechanics and species diffusion, for either one-dimensional (1D) spherical, two-dimensional (2D) axisymmetric, or 3D geometries.
- **CFAST:** Developed by National Institute of Standards and Technology (NIST), the Consolidated Model of Fire and Smoke Transport (CFAST) is a computer program that fire investigators, safety officials, engineers, architects, and builders can use to simulate the impact of past or potential fires and smoke in a specific building environment. CFAST is a two-zone fire model used to calculate the evolving distribution of smoke, fire gases, and temperature throughout compartments of a building during a fire. The CFAST package includes NIST’s Smokeview program, which visualizes with colored, 3D animations, the results of the CFAST simulation of a

specific fire's temperatures, various gas concentrations and growth and movement of smoke layers across multi-room structures.

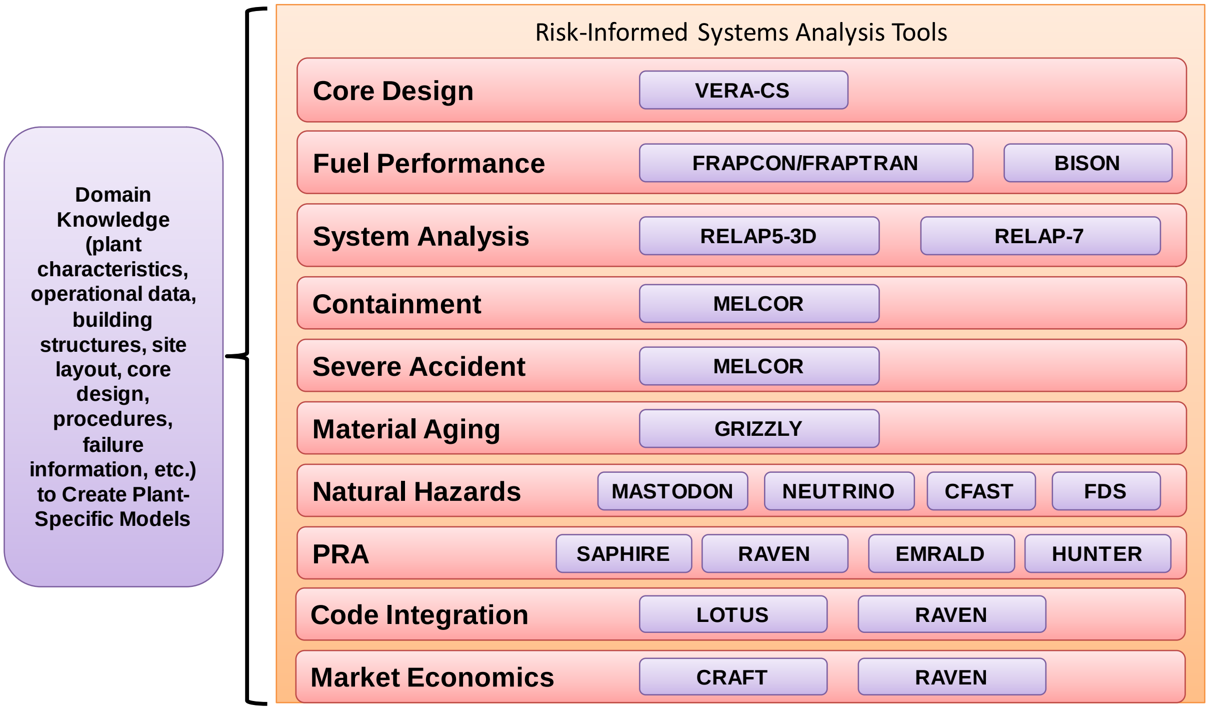

Figure 1-1 Current software modules used to perform RISA-specific analyses.

- **CRAFT:** CRAFT is a stochastic analysis framework that has been designed to evaluate the risk of SSCs of complex systems, including NPPs. The risk is evaluated from both a financial and a safety perspective by explicitly considering aging of SSCs and their impact on the overall plant risk. CRAFT applications range from plant asset management to plant risk diagnostics and prognostic.
- **EMRALD:** EMRALD is a state-based discrete event simulation tool that can calculate system failure probabilities, couple multiple simulations, and perform dynamic PRA. A key part of the EMRALD tool is to develop an object-oriented model that is flexible enough to support the varied dynamic simulation models (e.g., fails to operate, fails on demand). By having a state-based approach, it can integrate different hazards into a single comprehensive model. For example, a single model can include fire-, flooding-, transient-, and seismic-initiating events. Each of these events becomes a trigger into the state-based approach that tells the model to make a transition based upon the specific initiator.
- **FDS:** NIST developed the computational fluid dynamics Fire Dynamics Simulator (FDS) code to perform computational fire modeling and simulation. The code has been extensively validated for the types of fire scenarios encountered in both standard buildings, as well as nuclear environments. FDS facilitates the simulation of combustion, including fire migration, of an arbitrary number of materials in geometrically complex environments.
- **FRAPCON/FRAPTRAN:** FRAPCON/FRATRAN is a suite of codes developed by Pacific Northwest National Laboratory (PNNL) for the NRC for the purposes of performing fuel performance analyses under steady state (FRAPCON) and transient (FRAPTRAN) conditions. FRAPCON is a computer code that calculates the steady-state response of LWR fuel rods. The

code calculates the temperature, pressure, and deformation of a fuel rod as functions of time-dependent fuel rod power and coolant boundary conditions. FRAPTRAN calculates the transient performance of LWR fuel rods during reactor transients and hypothetical accidents such as LOCAs, anticipated transients without scram, and reactivity-initiated accidents. FRAPTRAN calculates the temperature and deformation history of a fuel rod as a function of time-dependent fuel rod power and coolant boundary conditions.

- **GRIZZLY:** GRIZZLY is a simulation code being developed to simulate the progression of aging mechanisms and SSCs in LWRs and to assess their ability to safely perform their intended engineering functions after being subjected to aging. GRIZZLY is ultimately planned to have capabilities for modeling a variety of structures, but current development is focused on reactor pressure vessels (RPVs) and concrete structures because of the essential functions and extreme difficulty of mitigating degradation or replacement of those components. For RPVs, GRIZZLY has a modern and flexible architecture for multidimensional engineering fracture mechanics analysis, which allows it to compute the probability of fracture in the presence of a population of pre-existing flaws that can serve as fracture initiation sites under a given transient event. It also has a set of models being developed to predict microstructure evolution under irradiation, which will be used to provide improved predictive models of embrittlement that can be applied for long-term operation scenarios. For concrete structures, GRIZZLY has coupled physics models to predict expansive mechanisms, including alkali-silica reaction and radiation-induced volumetric expansion, and their effects on the mechanical response of the structure, including fracture and damage.

- **HUNTER:** HUNTER is a flexible hybrid approach that functions as a framework for dynamic modeling, including a simplified model of human cognition—a virtual operator—that produces relevant outputs, such as the human error probability (HEP), time spent on task, or task decisions based on relevant plant evolutions. HUNTER is the human reliability analysis counterpart to the RAVEN framework used for dynamic PRA. Although both RAVEN and HUNTER are under various stages of development, there has been a successfully integrated and implemented RAVEN-HUNTER initial demonstration. The demonstration centers on a station blackout scenario, using complexity as the sole virtual operator performance-shaping factor (PSF). The implementation of RAVEN-HUNTER can be readily scaled to other nuclear power plant scenarios of interest and will include additional PSFs in the future.

- **LOTUS:** LOTUS is a Multi-Physics Best Estimate Plus Uncertainty (MP-BEPU) analysis framework being developed at INL, which establishes the automation interfaces among the various disciplines in NPP systems analysis, such that uncertainties can be propagated consistently in multi-physics simulations. LOTUS integrates existing computer codes, as well as the advanced computer codes still being developed under various U.S. Department of Energy (DOE) programs, to provide feedback and guide development of advanced tools. Regardless of the specific codes used to model the physics involved, the methodology developed in LOTUS is a paradigm shift in managing the uncertainties and assessing risks. LOTUS uses a ‘plug-and-play’ approach where the codes are simply modules ‘under the hood’ that provide the input-output relationships for a specific discipline. The focus shifts on managing the data stream at a system level. LOTUS is essentially a workflow engine with capability to drive physics simulators, model complex systems, and provide risk assessments.
- **MASTODON:** MASTODON is a tool that will have the capability to perform stochastic Non-linear Soil-Structure Interaction (NLSSI) in a risk framework coupled with virtual NPP. These NLSSI simulations will include structural dynamics, time integration, dynamic porous media flow, hysteretic nonlinear soil constitutive models (i.e., elasticity, yield functions, plastic flow directions, and hardening softening laws), hysteretic nonlinear structural constitutive models, and geometric nonlinearities at the foundation (i.e., gapping and sliding) [3].

- **MELCOR:** MELCOR is a computational code developed by Sandia National Laboratories (SNL) for the NRC, DOE, and the International Cooperative Severe Accident Research Program (CSARP). MELCOR simulates the response of LWRs during severe accidents. Given a set of initiating events and operator actions, MELCOR predicts the plant's response as the accident progresses. MELCOR also includes containment transient analysis capabilities to model thermal hydraulic phenomena (within a lumped-parameter framework) for existing containment designs for boiling water reactors (BWRs) and PWRs.

- **Neutrino:** Neutrino is a mesh-free, smooth particle hydrodynamics-based solver developed by Centroid Lab, which also uses advanced boundary handling and adaptive time stepping. Neutrino is an accurate fluid solver and is being used to simulate coastal inundation, river flooding, and other flooding scenarios. Neutrino code can model friction and adhesion between solid/fluid boundaries and various adhesive hydrodynamic forces between fluid/fluid particles [4].

- **RAVEN:** RAVEN is a flexible and multi-purpose uncertainty quantification, regression analysis, probabilistic risk assessment, data analysis and model optimization framework, designed to perform parametric and stochastic analyses based on the response of complex systems codes. It can communicate directly with the system codes described above and below (e.g., RELAP5-3D, Neutrino, BISON, MAAP, etc.), which are currently used to perform plant safety analyses. Depending on the tasks to be accomplished and on the probabilistic characterization of the problem, RAVEN perturbs (e.g., Monte-Carlo, Latin hypercube, reliability surface search) the response of the system under consideration by altering its own parameters. The data generated by the sampling process is analyzed using classical statistical and more advanced data mining approaches. RAVEN also manages the parallel dispatching (i.e. both on desktop/workstation and large High-Performance Computing machines) of the software representing the physical model. RAVEN heavily relies on artificial intelligence algorithms to construct surrogate models of complex physical systems in order to perform uncertainty quantification, reliability analysis (limit state surface) and parametric studies.

- **RELAP5-3D:** The RELAP5-3D code has been developed for best-estimate transient simulation of LWR coolant systems during postulated accidents. Specific applications of the code have included simulations of transients in LWR systems, such as LOCA, Anticipated Transients without Scram (ATWS), and operational transients, such as loss of feed water, loss of offsite power, station blackout, and turbine trip. RELAP5-3D, the latest in the series of RELAP5 codes, is a highly generic code that, in addition to calculating the behavior of the reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of vapor, liquid, non-condensable gases and nonvolatile solutes.

- **RELAP-7:** The RELAP-7 code is the next generation nuclear reactor system safety analysis code being developed at INL. The code is based on INL's modern scientific software development framework, Multi-Physics Object Oriented Simulation Environment (MOOSE). The overall design goal of RELAP-7 is to take advantage of the previous thirty years of advancements in computer architecture, software design, numerical integration methods, and physical models. The result will be a reactor system analysis capability that retains and improves upon RELAP5-3D's capabilities and extends the analysis capability for all reactor system simulation scenarios.

- **SAPHIRE:** SAPHIRE is a software application developed for performing a complete PRA using a personal computer (PC) running the Microsoft Windows operating system. SAPHIRE enables users to supply basic event data, create and solve fault and event trees, perform uncertainty analyses and generate reports. For NPP PRAs, SAPHIRE can be used to model a plant's response to initiating events, quantify core damage frequencies, and identify important contributors to core damage (Level 1 PRA). The program can also be used to evaluate containment failure and release models for severe accident conditions given that core damage has occurred (Level 2 PRA). In addition, SAPHIRE can be used to analyze both internal and external events and, in a limited manner, to quantify the frequency of release consequences (Level 3 PRA).

- **VERA-CS:** VERA-CS is a core simulator tool developed by the Consortium on Advanced Simulation of LWRs (CASL) and includes coupled neutronics, thermal-hydraulics, and fuel temperature components with an isotopic depletion capability. The neutronics capability is based on MPACT, a 3D whole core transport code. The thermal-hydraulics and fuel temperature models are provided by the COBRA-TF (CTF) subchannel code. The isotopic depletion is performed using the ORIGEN code system.

The industry codes such as CAFTA (i.e., a PRA tool), Modular Accident Analysis Program (MAAP) (i.e., a systems analysis tool), and GOTHIC (i.e., a containment response tool) will be used in the pilot demonstration projects:

- **CAFTA**: Developed by the Electric Power Research Institute (EPRI), CAFTA is an integrated tool to perform PRA, incorporating linking event tree/fault tree methodology. The code is a fault tree analysis tool, utilizing the full power of today's PCs and providing the ability to effectively model and analyze complex systems. As fault tree analysis assumes a greater importance in many industries, the need to develop models in a logical and efficient manner has increased. CAFTA code addresses this need by providing a set of interactive editors, databases, and model evaluation tools. This interactive environment promotes the smooth flow of information throughout the model development, quantification, and results interpretation process.
- **MAAP**: MAAP is a fast-running computer code that simulates the response of LWR and heavy water reactor (HWR) moderated NPPs for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate both LOCA and non-LOCA transients for PRA applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).
- **GOTHIC**: GOTHIC is a versatile, general purpose thermal-hydraulics software package, which solves the conservation equations for mass, momentum, and energy for multi-component, multi-phase compressible flow in lumped parameter and/or multi-dimensional (i.e., 1D, 2D, or full 3D) geometries. The ability to combine these different nodalization options in a single model allows GOTHIC to provide computationally efficient solutions for multi-scale applications.

1.4 RISA Toolkit Deployment Plan

The RISA Toolkits are a set of computer software that will be used through the industry application pilot demonstrations and deployed to U.S. nuclear industry to support risk-informed margin management (RIMM) analysis. Since not all software has been fully verified and validated, the RISA Pathway will perform appropriate degree of software Verification and Validation (V&V) in order to give clear understanding to future RISA toolkit users in US nuclear industry. The industry will engage to the selected pilot demonstration projects. The RISA Pathway will continuously communicate with industry to develop issues and to collect feedback. The RISA toolkit deployment will have the following four process steps:

1. Select tools and methods.
2. Confirm verification and validation status.
3. Pilot demonstrations using selected tools and methods.
4. Industry deployment and feedback.

In order to meet above four deployment steps. The RISA Pathway proposes maximum five years project plan as shown in Figure 1-2. For the first year, the RISA Pathway will focus on the initiation of selected pilot demonstration projects and will deliver preliminary results on the case studies. The feedback on selected pilot demonstration projects will be communicated with industry through a specific RISA Pathway working group. Based on the preliminary studies, the pilot demonstration projects will be extended to full scale analysis during next two years. The validation and verification of associated the RISA toolkit will be done in this period. For the last two years of the project, the RISA Pathway will develop optimized methods of R&D results implementation to industry as well as long-term support plan by research institutes for U.S. nuclear industry to provide sustainable benefit from risk-informed margin management. It is noted that the project timeline could be different to the technical maturity and development status of using the tool and method.

1.4.1 Selection of RISA Toolkit

The tools and methods used in the RISA Pathway should have high-confidence and enough technical maturity and ability to cover a wide RIMM area range. Advanced technologies should be applied to the RISA Toolkits, such as multi-physics and multi-scale analysis, and cutting-edge computational proficiency as well as capability of uncertainty control if necessary. The toolkit should also have the capability to support risk-informed decision-making for both probabilistic and deterministic elements of safety. Current RISA toolkit includes various computer simulation tools that can cover wide range of work scope. Many of tools are currently available in related industry, and well validated with mature technology level. However, there are still many tools under development or needing to be verified and/or validated to confirm the technical maturity and suitability for the RISA Pathway framework.

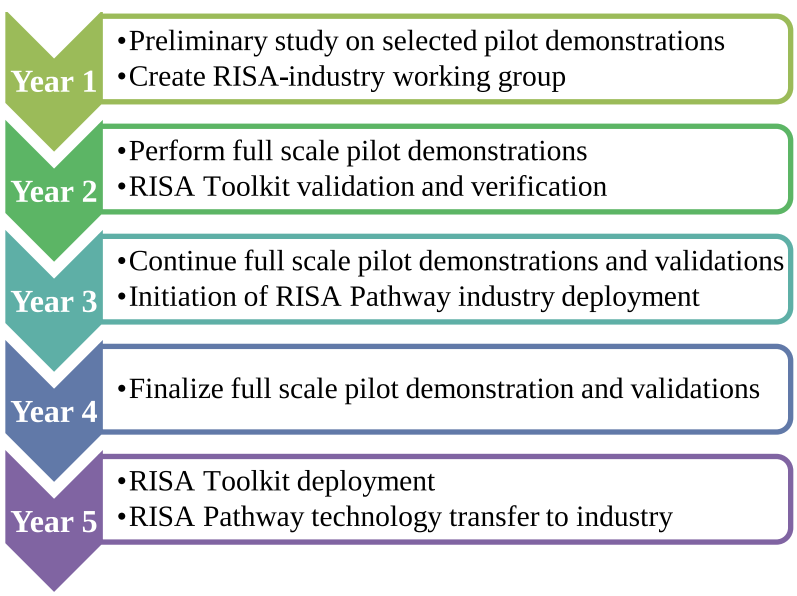

Figure 1-2 Notional 5-year plan of a RISA Pathway Industry Application

1.4.2 Verification and Validation (V&V) of the RISA Toolkit

In order to provide confidence during industry deployment, the selected tools and methods should be address quality-assurance levels appropriate for industry use. Well-known methods such as V&V and uncertainty quantification of the produced result will enhance credibility of the selected tools and enable industry to use them with confidence. The selected RISA Toolkits will be examined to confirm V&V

status to show its Technology Readiness Level (TRL) and to assure quality of the outcomes. The RISA Pathway will deliver the annual report of V&V examination and confirmation data for the selected RISA toolkit. The deliverable will include specific information of the tool such as capability and features, quality assurance program, developer/independent V&V record, separation/integral tests history, user documents, and feedback.

1.4.3 Pilot demonstration using RISA Toolkit

As of FY-2019, a total of eight RISA pilot demonstration projects are proposed during comprehensive discussion with U.S. nuclear industries and related institutes. Each project has its own selected tools and methods to show optimum RIMM, which aims to enhance both safety and economics. The RISA Pathway will maintain strong engagement with the U.S. nuclear industry to perform each pilot demonstration project. Direct participation of industry will provide better understanding on arising issue and can facilitate promoting innovative solutions, as well as smooth deployment of RISA toolkit in the future.

1.4.4 RISA Toolkit Industry Deployment and Feedback

Successful deployment of the RISA Toolkits to the U.S. nuclear industry is the one main goal of the entire RISA Pathway. The industry has been involved in the RISA Pathway from its initiation and will support the pilot demonstration project by using the selected RISA toolkit. During the pilot demonstration project phase, industry will have enough time to experience selected tools and methods. The RISA Pathway will organize its annual meeting with leading U.S. nuclear industries to develop additional pilot demonstration projects, collect feedback to upgrade on-going projects, and discuss an effective RISA toolkit implementation strategy. Licensing and regulation issues will be also addressed. The result of this meeting will be published and shared for long-term maintenance of the RISA toolkit.

2. METHODS FOR TECHNICAL MATURITY EVALUATION

Toolkit for RISA Pathway mainly represents related computer software to be used for pilot demonstration projects as listed in Section 1.3. Since the simulation appears to be the only feasible option to quantitively capture realistic aspects of the multi-physics behavior of the complex system, the software user should clearly understand features, characteristics and any potential issues of selected tool. The RISA Pathway focuses on use of modeling and simulation tool for risk-informed approach, thus, probabilistic risk assessment (PRA). This means selected software should have own PRA capability or could be coupled with other PRA tool to apply for risk-informed margin management of the nuclear power plant.

The technical maturity evaluation of the RISA toolkit will therefore focus on PRA capability and/or applicability. The toolkit should be constructed easy enough for industry users and continuous user support program by developer/manufacturer needs to be maintained. If the toolkit is still under development, the developer must have responsibility for further maintenance once development is finalized. Update or significant change of the software structure is not likely occurred. Version control, V&V history, license maintenance, quality assurance programs and customer service are also required for sustainable use of the software. The RISA Pathway will summarize above information on RISA toolkit to facilitate deployment of the tools to industry.

The requirements for the RISA toolkit are proposed by Requirement Traceability Matrix (RTM). Since 2015, the RISA Pathway has been using RTM to evaluate development status and V&V proposal for RELAP-7, the next generation nuclear system safety analysis code [2]. This method has been showing efficient analysis result by capturing necessary requirement for developers by reflecting needs from the users. The RTM will be then evaluated by Phenomena Identification and Ranking Technology (PIRT) to categorize importance of the requirements. The RISA Pathway has been using PIRT method to review technical maturity NEUTRINO, smooth particle hydrodynamics computational fluid dynamics software [3]. Three levels of ranking, high-medium-low, will be used for each requirement. Finally, Technology Readiness Level (TRL) method will give maturity level of each requirement. The RISA Toolkit maturity assessment matrix is proposed in APPENDIX A.

2.1 Requirement Traceability Matrix (RTM)

The Requirements Traceability Matrix (RTM) is created to associate specific requirements with the work products that satisfy them. Tests are also associated the requirements on which they are based so documented proof exists that the product has met the requirement. The matrix provides unique identifiers for each requirement and ensures completeness that lower level requirements come from higher level requirements. This matrix provides a tangible item that can be examined by the customer and the provider alike. Information that may not be clearly explained in descriptive text will be directly traceable forwards and backwards from a requirement or from an associated test. Traceability is used to manage change and provides a basis for test planning. It is a tool to ensure that the software process has been completed from initial definition to completion of a product. Use of an RTM provides a software quality check point. It is however noted that RISA toolkit V&V status assessment will not use traceability matrix but will only specify requirements for target software.

As shown in Table 2-1, the RISA Pathway reviewed RELAP-7 next generation thermal-hydraulic code to support its development and V&V activities [2]. Three types of requirements were used for RELAP-7 assessment: General requirements, Specific requirement and Technology/V&V requirements. General requirements are its basic features and capability based on developer's purpose of the tool. For example, REALP-7 is designed to simulate various accidents of nuclear power plants, hence, the general requirements of RELAP-7 were mostly on simulation capability of loss-of-coolant accident (LOCA) and/or non-LOCA based accident scenario and other related phenomena. The specific requirements are mainly unresolved legacy issues of the field of interest which the reasons are complexity of the technology and/or challenges in research and development. The legacy issues are categorized into physical phenomena, numerical discretization, code and modeling accuracy, computer science, validation, etc. The technology/V&V requirements cover numerical architecture, software design, graphical interface, physical phenomena, V&V record, regression test, QA program, user feedback service, etc.

Table 2-1 Example of general requirements RTM (ref)

Req #	Category	Requirement Specification	Modification Date (mm/dd/yyyy)	Code V&V RTM No.	RELAP-7 Status
GR-1	Reactor Types & System Designs	Capability of simulating various LWR designs such as PWR and BWR	9/30/2015	All test cases falling into PWR or BWR category in Code V&V RTM.	Sample Test-45 for BWR (HEM model) Sample Test-152 for PWR (TMI loop)
GR-2		Capability of simulating various PWR designs (i.e., Westinghouse, Combustion Engineering, and Babcock&Wilcox)	9/30/2015	All test cases falling into PWR or PWR (B&W) category in Code V&V RTM.	PWR cores are tested but not compared for different manufactures
GR-3		Capability of simulating various containment design influencing LOCA simulation	9/30/2015	-	Not tested
⋮					
GR-9	T/H System Safety Analysis (Design- and Licencing-Basis Transients/Accidents)	Capability of simulating excessive heat transfer events (non-LOCA)	9/30/2015	VR-87, 88	Not tested
GR-10		Capability of simulating loss of heat transfer events (non-LOCA)	9/30/2015	VR-80, 85, 86, 90-92, 95-97, 100, 101, 104, 105, 108, 110-113, 116-119, 121, 124, 125, 128, 129, 131	Not tested
GR-11		Capability of simulating loss of flow events (non-LOCA)	9/30/2015	VR-86, 104, 118, 119, 131	Not tested
⋮					

The requirements for RISA Toolkit mainly will focus on higher technical maturity and capability or applicability of PRA. Following requirements are proposed for RISA Toolkit. The requirements could be changed/added based on type of the tool.

Development level

- RISA Toolkit should have highest development level that can be immediately used for industrial application. If the tool is still under development, the development goal should clearly indicate commercialization plan. Use of under-developed or lower technical maturity tool will decrease credibility of risk-informed analysis outcome.

Use of proven technology

- Some of newly developed tool uses cutting edge technology to build the software. However, such forefront technology may not always applicable to existing or aged industry system. The candidate of RISA Toolkit should be applicable to entire existing U.S. nuclear power plants.

PRA capability/applicability

- Since the main goal of the risk-informed analysis is to apply PRA to the safety margin management, the RISA Toolkit should have its own PRA capability or applicable to existing PRA tool.

Documentation

- RISA Toolkit should have recognizable list of documents which user can access easily, including user manual, theory manual, development plan, QA plan, V&V plan and results, etc.

System requirements

- Users will equip diverse computer and operating systems, and RISA Toolkit should be operated in different computer systems. Hence, the system requirements and related verification test result should be documented properly.

Easy installation

- RISA Toolkit should be installed easily without developer's support. Facility of installation will be reviewed.

Graphic user interface (GUI)

- GUI will provide head-up information to users. Any type of graphical out-put generator will be listed.

Version control

- For every step of software update, the version history of the tool should be maintained for users. Related documents and manuals should also be updated.

V&V history

- In high-level definition, verification is to check the tool is correctly built while validation means to confirm the outcome of verified code is correct. Since V&V result is one of most important parameters for the quality of code outcome, the history of V&V activity by developer and third party should be documented. This includes regression test list for code verification.

QA program

- Quality assurance program will provide standard of RISA Toolkit quality. This will include ASME Nuclear Quality Assurance (NQA-1) certification program to support nuclear industry for high quality products and services.

Web page

- Dedicated web page should be maintained in order to communicate with users.

User support

- The code developer should provide feedback on the problems reported from users.

Training program

- For easy deployment to industry, training program by developer or RISA Pathway will be a good solution.

License

- The developer should clearly indicate details of licensing.

2.2 Phenomena Identification and Ranking Technique (PIRT)

The PIRT (Phenomena Identification and Ranking Technique) is a method to systematically gather information from experts on a specific issue (e.g., NPP accident) and rank the importance of the information. The ultimate goal is to support decision-making such as determining which information should have high priority for research on that subject. The PIRT was first developed in the late 1980s and has been successfully applied in nuclear technology such as nuclear reactor safety analysis. This is also an essential sub-process for BEPU (Best Estimate Plus Uncertainty) analysis such as CSAU (Code Scaling, Applicability, and Uncertainty), the BEPU methodology acknowledged by the NRC. From a perspective of BEPU analysis, the importance of PIRT comes from the fact that the information obtained from the

PIRT (i.e., relative importance of phenomena) is used to determine the code uncertainty input parameters and their realistic boundaries. In general, experts determine the relative importance of phenomena by considering their influence on the relevant plant safety metrics (e.g., peak cladding temperature). Figure 2-1 [3]. Figure 2-1 shows the standard PIRT proposed for nuclear industry application.

For the safety analysis of NPP, the PIRT process allows us to identify and prioritize the phenomena relevant to the specific NPP accidents so that we can properly analyze them. An example of the PIRT application was performed by RISA Pathway for NEUTRINO, the flooding hazard analysis computer software. Developed by Neutrino Dynamics, the NEUTRINO is a general-purpose simulation and visualization computational fluid dynamics code which uses Smoothed Particle Hydrodynamics (SPH) mesh-free particle-based Lagrangian computational solver [4]. RISA Pathway performed a comprehensive assessment of the NEUTRINO code development and V&V status by using "high-level PIRT" method. In establishing the high-level PIRT, the characteristics of flooding hazards were first classified based on the specific hazard modes and associated physics. Shown in Table 2-2, the development status was easily evaluated by using PIRT ranking method and will give clear understanding to users. For RISA Toolkit V&V status assessment, three level ranking methods will be applied to requirements set in Section 2.1 by RTM:

High : Dominant impactful requirement. Require high level of technical maturity.

Medium: Moderate impactful requirement. Require medium level of technical maturity.

Low: Small or no impactful requirement. Satisfy with basic level of technical maturity.

Table 2-2 Ranking of importance and NEUTRINO code development status for flooding hazard analysis

Hazard type	Major phenomena	Importance	NEUTRINO code	
			Capability	Validation
Water rising	Water level / wetting area	High	High	High
Pressure and wave	Velocity profile (wave propagation and dissipation)	High	High	High
	Vortex (turbulence)	Low	Medium	Medium
	Fluid-solid impact (impact forces, spray)	High	Medium	Low
Debris migration	Buoyancy	Medium	High	High
	Fluid-solid interaction (debris travelling)	High	Medium	Low
	Solid-solid interaction (collision, force impact)	High	Low	Low

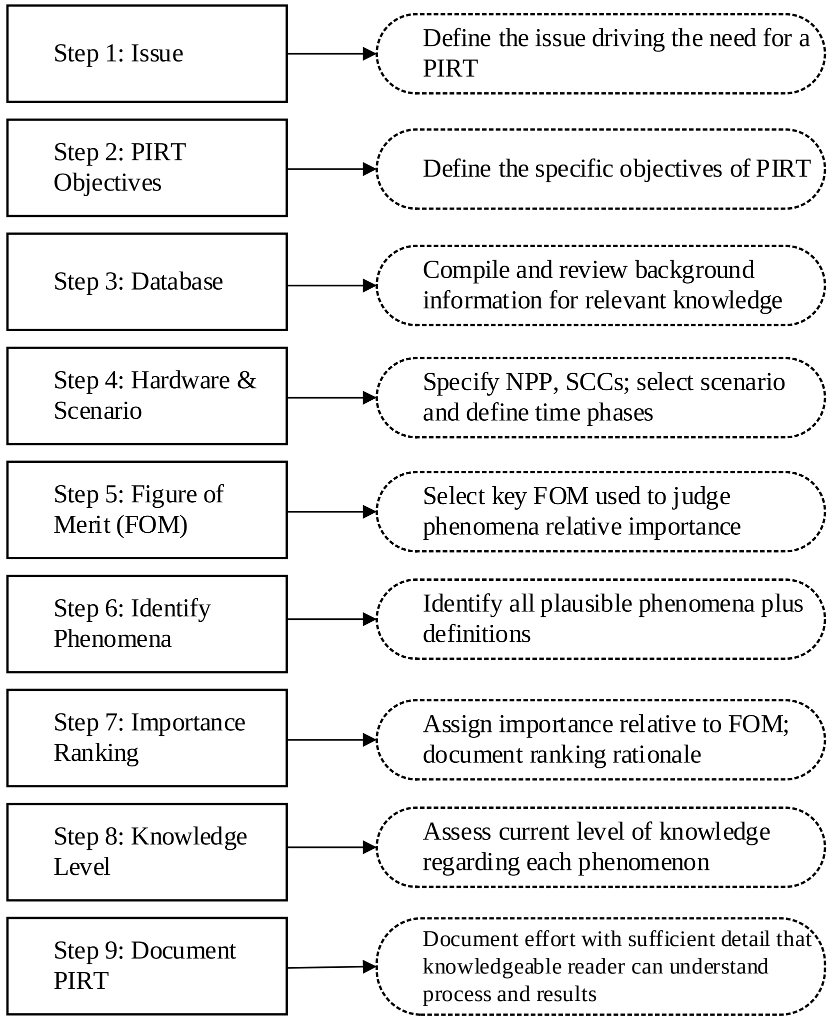

Figure 2-1 Standard process of PIRT proposed for nuclear industry application [3]

2.3 Technology Readiness Level (TRL)

Technology Readiness Levels (TRL) are a type of measurement system used to assess the maturity level of a particular technology. Each technology project is evaluated against the parameters for each technology level and is then assigned a TRL rating based on the projects progress. There are nine technology readiness levels as shown in Figure 2-2. TRL 1 is the lowest and TRL 9 is the highest. When a technology is at TRL 1, scientific research is at the beginning and those results are being translated into future research and development. TRL 2 occurs once the basic principles have been studied and practical applications can be applied to those initial findings. TRL 2 technology is very speculative, as there is little to no experimental proof of concept for the technology. When active research and design begin, a technology is elevated to TRL 3. Generally, both analytical and laboratory studies are required at this level to see if a technology is viable and ready to proceed further through the development process. Often during TRL 3, a proof-of-concept model is constructed.

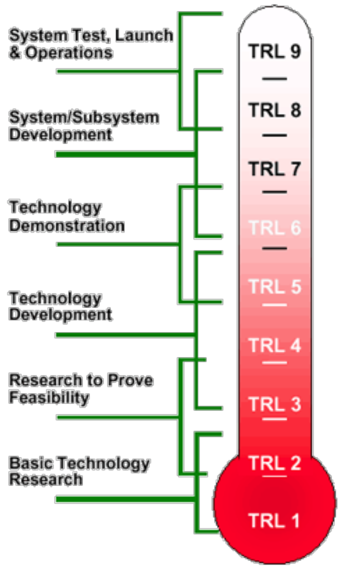

Figure 2-2 Schematic diagram of Technology Readiness Level (TRL)

Once the proof-of-concept technology is ready, the technology advances to TRL 4. During TRL 4, multiple component pieces are tested with one another. TRL 5 is a continuation of TRL 4, however, a technology that is at 5 is identified as a breadboard technology and must undergo more rigorous testing than technology that is only at TRL 4. Simulations should be run in environments that are as close to realistic as possible. Once the testing of TRL 5 is complete, a technology may advance to TRL 6. A TRL 6 technology has a fully functional prototype or representational model. TRL 7 technology requires that the working model or prototype be demonstrated in a space environment. TRL 8 technology has been tested and "flight qualified" and it's ready for implementation into an already existing technology or technology system. Once a technology has been "flight proven" during a successful mission, it can be called TRL 9.

The U.S. Department of Energy proposed Technology Readiness Assessments (TRAs) and developing Technology Maturation Plans (TMPs) to assist individuals and teams that will be involved in conducting program and project management for the acquisition of capital assets [5]. TRAs and TMPs activities are a tool to assist in identifying technology risks and enable the correct quantification of scope, cost and schedule impacts in the project. Recent application of TRL in nuclear industry was to review maturity of advanced nuclear fuels and material development [6]. The purpose of the research was to evaluate existing technologies and identify R&D needs while developing advanced fuels and materials. Similar to the RTM list, required parameters for critical issues were defined and measured its TRL. The level of TRL was also again defined based on the goal of advanced fuel and material development.

Based on general TRL, specific TRL for RISA Toolkit is proposed in Table 2-3. The RISA Pathway aims deploy higher TRL toolkit to industry which will be TRL 7 or higher. For the RISA toolkit, the industry pilot demonstration project is the best practice to upgrade TRL 6 or lower toolkit to TRL 7 or higher. Figure 2-3 shows schematic overview of TRL increase of RISA Toolkit through pilot project.

Figure 2-3 Schematic diagram of TRL increase through RISA pilot project

Table 2-3 Description of Technology Readiness Level (TRL) for RISA Toolkit

Level of Tech. Development	Technology Readiness Level	TRL Definition	Application to RISA Toolkit
System Operation	TRL 9	Actual system operated over the full range of expected conditions.	The toolkit technology is in its final form and routinely use under the full range of industrial purpose.
System Development	TRL 8	Actual system completed and qualified through test and demonstration.	The toolkit has been proven to operate in its final form and under expected conditions. In almost all cases, this TRL represents the complete of R&D. Entire planned and proposed V&V of the toolkit are finalized by developer/user.
	TRL 7	Full-scale, similar (prototypical) system demonstrated in relevant environment	Full-scale demonstration of actual/prototypical technology/toolkit in relevant operation environment. Full-scale experiment and validation are performed. Development of the toolkit is virtually complete.
Technology Demonstration	TRL 6	Engineering/pilot-scale, similar (prototypical) system validation in relevant environment	Engineering-scale models or prototypes are tested in a relevant environment. Experiment and validation in engineering/pilot-scale environment including scaling effect testing which can support operational system design. This level represents completion of technology development for operational demonstration and prototype toolkit is ready for test. The prototype toolkit will have capability of performing all the functions that will be required from actual operational system. The operating environment for the testing should closely represent the actual operating environment. Major difference between TRL 5 is the scale-up from laboratory to engineering size.
Technology Development	TRL5	Laboratory scale, similar system validation in relevant environment	The basis of technology is fully integrated into the toolkit and ready for larger scale demonstration and validation. This level will include results from the laboratory scale test, analysis of the differences between the laboratory and actual operating system/environment, and analysis of experiments/demonstrations results for actual operating system/environment application. The major difference between TRL 4 and 5 is the increase of the toolkit fidelity and environment to the actual application. Verification is complete and the toolkit development level is close to prototypical.
	TRL 4	Component and/or system validation in laboratory environment	The basis of technology for toolkit are partly integrated and can be apply for component level demonstration. This is relatively "low fidelity" compared with the actual level of toolkit completion level. The expected maturity of this level includes the integrated experiments and validation, examination of scaling effect and actual application. Verification and regression test could be included. TRL 4-6 represent the bridge from scientific research to engineering. TRL 4 is the first step in determining whether the basic modeling will work in the toolkit.
Research to Prove Feasibility	TRL 3	Analytical and experimental critical function and/or characteristic proof of concept	Actual R&D is started for toolkit development. This includes analytical studies and laboratory-scale studies to validate the phenomena of separate technology. This level will have results of laboratory tests performed to measure parameters of interest and comparison to analytical predictions for critical toolkit functions. At TRL 3, actual R&D progresses to experiments and verifications. Validation could be done for part of the toolkit development, but system level validation is not yet initiated.

Basic Technology Research	TRL 2	Technology concept and/or application formulated	Progressed from TRL 1, technical options may be developed in TRL 2. However, still no activity was performed to prove assumptions and concept. Literature studies will outline the toolkit development concept. Most of activity in this level is analytical research and paper studies to understand goal of the R&D. Related experiments and V&V works could be designed during this level.
	TRL 1	Basic principles observed and reported	This is the lowest level of technology readiness. Scientific research begins to be translated into applied R&D. Available information includes published research or other references that identify the principles that underline the technology. No actual R&D started.

3. V&V STATUS ASSESSMENT FOR RELAP5-3D

3.1 Overview

RELAP5-3D (Reactor Excursion and Leak Analysis Program) is a computer simulation software dedicated to the nuclear power plant operational transient and accident thermal-hydraulics analysis. Developed at Idaho National Laboratory (INL) and originally funded by U.S. Atomic Energy Commission (current U.S. NRC), the RELAP5-3D is the state-of-the art for reactor safety analysis, reactor design, simulator training of operators, and nuclear industrial facility licensing.

History of the RELAP5-3D development is well summarized in “A History of RELAP Computer Codes” written by G. L. Mesina [7]. Early development was done in 1960’s by homogeneous equilibrium model (HEM) fluid flow analysis code FLASH-1 which uses three control volumes to model loss-of-coolant accident (LOCA) analysis. As a revised version of FLASH-1, RELAP1 (or RELAPSE-1) was written in FORTRAN IV to calculate pressures, temperatures, flow, mass inventories, reactivities, and power for PWRs during a reactivity event or LOCA. Release in 1968, RELAP2 could resolve BWR system behavior and ran twice faster than RELAP1. From RELAP3, maximum 20 control volumes, components and expanded heat transfer model could be simulated. Built by ANSI FORTRAN 1966, RELAP4/MOD1 was release in 1973. This version could handle up to 100 control volumes and true one-dimensional (1-D) flow for large-break LOCA (LB-LOCA) scenarios.

First version of RELAP5 was developed in 1979 as RELAP5/MOD0. Main improvement from previous RELAP series is change from one-fluid HEM to a two-fluid model with a different set of governing equations for the liquid and gas phases as well as significant improvement on capability of small-break LOCA analysis. Another important update was change to FORTRAN 77. This RELAP5 has been updated until RELAP5/MOD3.2.

By support from U.S. Department of Energy (DOE), RELAP5-3D was developed in mid-1990’s by INL. In the late 1990s, the International RELAP5 Users Group (IRUG) was organized to support international RELAP5-3D users and to disseminate code worldwide. Notable features of the RELAP5-3D are full three-dimensional hydrodynamics with rectangular, cylindrical and spherical geometries and introduce of the RELAP5-3D Graphical User Interface (RGUI). RELAP5/Ver. 2.0 was the first expansion to the Windows operating system. RELAP5-3D/Ver. 2.4 was the final FORTRAN 77 code version. Latter versions are built with Fortran 90/95/2003. RELAP5-3D/Ver. 3.0 was the beta-test release of the first fully Fortran 95 version. As of 2019, RELAP5-3D/Ver. 4.4.2 is the most recent release and the most robust, verified, and validated product of the RELAP5 series.

3.2 Features

Main purpose of the RELAP5-3D code is for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems. RELAP5-3D also has three-dimensional thermal hydraulics and neutron kinetic modeling capabilities. The multi-dimensional component in RELAP5-3D was developed to allow the user to accurately model the multi-dimensional flow behavior that can be exhibited in any component or region of a nuclear reactor coolant system. There is also two dimensional conductive and radiative heat transfer capability and modeling of plant trips and control systems. The features of the RELAP5-3D are very well stated at user manual Volume 1 [8].

RELAP5-3D allows for the simulation of the full range of reactor transients and postulated accidents, including:

- Trips and controls
- Component models (pumps, valves, separators, branches, etc.)
- Operational transients
- Startup and shutdown
- Maneuvers (e.g. change in power level, starting/tripping pump)
- Small and large break Loss Of Coolant Accidents (LOCA)
- Anticipated Transient Without Scram (ATWS)
- Loss of offsite power
- Loss of feedwater
- Loss of flow
- Light Water Reactors (PWR, BWR, APWR, ABWR, etc.)
- Heavy Water Reactors (e.g. CANDU reactor)
- Other types of the reactor (e.g. SMR, GenIV, etc)

3.2.1 Hydrodynamic Model

RELAP5-3D is a transient, two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain non-condensable components in the vapor/gas phase and/or a soluble component in the liquid phase. The multi-dimensional component in RELAP5-3D was developed to allow the user to more accurately model the multi-dimensional flow behavior that can be exhibited in any component or region of an LWR system. Typically, this will be the lower plenum, core, upper plenum and downcomer regions of an LWR. However, the model is general, and is not restricted to use in the reactor vessel. The component defines a one, two, or three-dimensional array of volumes and the internal junctions connecting them. The geometry can be either Cartesian (x, y, z) or cylindrical (r, q, z). An orthogonal, three-dimensional grid is defined by mesh interval input data in each of the three coordinate directions

3.2.2 Heat Structures Model

Heat structures provided in RELAP5-3D permit calculation of heat transferred across solid boundaries of hydrodynamic volumes. Modeling capabilities of heat structures are general and include fuel pins or plates with nuclear or electrical heating, heat transfer across steam generator tubes, and heat transfer from pipe and vessel walls. Temperature-dependent and space-dependent thermal conductivities and volumetric heat capacities are provided in tabular or functional form either from built-in or user-supplied data. There is also a radiative/conductive enclosure model, for which the user may supply/view conductance factors.

3.2.3 Trip System

The trip system consists of the evaluation of logical statements. Each trip statement is a simple logical statement that has a true or false result and an associated variable. Two types of trip statements are provided--variable and logical trips. Since logical trips involve variable trips and other logical trips, complex logical expressions can be constructed from simple logical statements. Both types of trips can be latched or unlatched. A latched trip, once set true, is no longer tested and remains true for the remainder of the problem or until reset at a restart. An unlatched trip is evaluated every time step.

3.2.4 Control System

RELAP5-3D allows the user to model a control system typically used in hydrodynamic systems, including other phenomena described by algebraic and ordinary differential equations. Each control system component defines a variable as a specific function of time-advanced quantities; this permit control variables to be developed from components that perform simple, basic operations.

3.2.5 Reactor Kinetics

There are two options that include a point reactor kinetics model and a multi-dimensional neutron kinetics model. A flexible neutron cross section model and a control rod model is available for detailed modeling of the reactor core. The decay heat model developed as part of the reactor point kinetics model and has been modified to compute decay power for point kinetics and multi-dimensional neutron kinetics models.

3.3 Verification and Validation Status

3.3.1 Verification of RELAP5-3D

From the very early stage of the RELAP5-3D development, the verification tests have been conducted to run the code features correctly on Linux or Windows system. In 2013, the verification procedure was upgraded to overcome previous deficiencies of the code coverage, detail of comparison, running time, and testing fidelity of RELAP5-3D restart and backup capabilities [9]. In addition, the sequential verification was performed in 2016 to checks coding against specifications only when originally written and then applies regression testing to compare code calculations from consecutive updates or versions on a set of test cases to ensure that the performance does not change [10]. List of verification suites are listed in Appendix B. RELAP5-3D code also has automatic regression test suites during installation in Appendix C.

The process test of RELAP5-3D verification includes null testing, restart testing and backup testing. For the operational mode testing RELAP5-3D runs either standalone or coupled to another program with specific control mode.

Null testing This verification test is to check identical calculation results from two different version of the RELAP5-3D. This is one of most important fundamental to code development that to verify code modification does not impact to result for same input problem.

Restart testing Restart is to continue running from certain point or end of the original calculation. RELAP5-3D records all calculation parameters for the entire simulation time frame. The purpose of restart testing is to ensure restarted run produces same calculations as the original run.

Backup testing RELAP5-3D has time-step backup feature to reset variables in case of improper solution has obtained. Backup testing checks that the code can correctly repeat any given advancement at the same time-step size. This test checks that the code still produces same calculations with a forced backup.

Operational mode testing Operation mode can run verification suite for coupling above three tests. Specific coupled test suites are marked with cpl_ .

3.3.2 Validation of RELAP5-3D

For RELAP5-3D, the special form of validation testing called “Developmental Assessment” is performed and documented in “RELAP5-3D Code Manual Volume III: Developmental Assessment” with high detail [11]. RELAP5-3D validation test suites includes phenomenological, separate effects, and integral effects cases to investigate how well selected code models perform for each version of the code.

Entire validation suites are performed with both semi- and nearly implicit numerical method for the benchmark test. The validation result evaluated as “excellent” or “reasonable” which are considered acceptable. Evaluated as “minimal” or “insufficient” indicate that additional work on the code models may be needed. The calculations were run in 64-bit mode on a personal computer using the Linux operating system. Comparison test with previous version and between different computer operation systems (Linux and Windows) are also performed. The list of validation test cases and the evaluation results are summarized in Appendix D.

Total of 18 phenomenological test cases and 23 separate effect models are validated. The phenomenological cases are generally simple problems that test one or two code models which the problems often have analytical solutions. These includes investigate core behavior, including two-phase level behavior, critical heat flux (CHF), and heat transfer (subcooled, nucleate, post-CHF, reflood). There are several cases that address critical flow. Other cases address countercurrent flow limitation (CCFL) in pipe and down-comer geometries, and phenomena associated with the pressurizer, steam generator, pump, and jet pump components.

The integral effects cases use data from large experimental facilities. These cases are generally of greater interest because they provide an indication of how well the code performs overall in modeling transients with large number of phenomena. Nine cases addressing eight experiments were included in the assessment. Test comprises two small break loss-of-coolant accident (LOCA) experiments, two large break LOCA experiments (the Loss-of-Fluid Test [LOFT] experiment L2-5 is modeled with both 1-D and 3-D components), and four loop natural circulation tests from the semi-scale facility.

As a conclusion of the “developer assessment” validation, most of the cases showed essentially no differences in results between calculations using either the semi- or nearly-implicit solution scheme. However, there appear to be errors in the nearly-implicit solution scheme associated with the multi-dimensional hydrodynamic component. Until these issues are resolved, users should account for this in determining how best to apply the code to specific facility simulations

3.4 Technical Maturity Assessment

Following information describes RELAP5-3D technical maturity assessment results based on the requirements as the RISA Toolkit.

3.4.1 Development level

RELAP5-3D is one of highest developed computer software in entire nuclear industry. This software has been using widely in the world for licensing, validating and educational purpose. Most of important nuclear thermal-hydraulics analysis could be covered. Some of versions have been modified by users to use in specific purpose, e.g., MARS computer code for light water reactor transient analysis. Latest version (RELAP5-3D/Ver. 4.4.2) was released in June 2018. Main update features of version 4.4.2 are updating consistent fluid implementation for H2ON, H2O95N fluid properties, run timing study capability, LiPb transport properties, fix reverse flow error Jet-mixer component, improvement of variation allocation and deallocation, updates for Groeneveld 2006 CHF correlation, update of CCFL model, and fix of Gnielinski heat transfer correlation for pure air. Current update activities and additional development needs are:

- RELAP5-3D Version 4.0 is built and tested with level 11.1 of the Intel FORTRAN and C-language compilers. Recently RELAP5-3D is re-coding to GFortran to imply in INL developed MOOSE (Multiphysics Object Oriented Simulation Environment). Verification tests of RELAP5-3D GFortran version will be completed in near future.

- RELAP5-3D has been using for supporting accident tolerance fuel (ATF) development. It is however identified that analysis of ATF cladding coating characteristics is not fully supported.
- GenIV type reactors are not fully supported by RELAP5-3D. Additional development is needed for liquid metal (Sodium, Lead-Bismuth, Malten Salt) properties and their thermal-hydraulics behavior
- To use in risk-informed analysis, RELAP5-3D must have or tightly coupled with other Best Estimate Plus Uncertainty (BEPU) and/or PRA capable RISA Toolkit. Different to conservative method, best estimate methodology means best estimate computer code and uncertainty analysis. The uncertainty analysis refers identification and quantification of important input parameters as well as combined impact to result. Uncertainty quantification (UQ) method is to calculate probabilistic information from simulation by using specified input parameter Probability Distributions Functions (PDFs). Another uncertainty analysis method is sampling method which calculates system performance distribution to evaluate most impactful input parameter to the result. Additional development is needed for RELAP5-3D to be used for risk-informed analysis.

3.4.2 Use of proven technology

Since RELAP5-3D is written in FORTRAN 95 and works stable in 32- and 64-bit computers, no specific issue were so far identified. From developer's verification report [11], a 64-bit machine allows integer, floating point, and logical data to be addressed in 8-byte (equivalently 64-bit) words of memory, can address 4-byte data efficiently, and allows 128-bit floating point operations. 32-bit platforms allow 4- and 8-byte data and operations also but have smaller native memory addresses (2 to 4 Gigabytes) and require operating system aid to address additional memory.

The RELAP5-3D distribution uses the FORTRAN 95 intrinsic library that eliminates machine dependencies of earlier versions. Further adaptability and portability could be achieved by adding binary files in machine independent formats. The fluids properties files are written in the eXtended Data Representation (XDR) format, while the plot files can be written in XDR or ASCII, although the option for machine dependent binary still exists. For the code coupling capability, available in some RELAP5-3D source code products only, the coupling software is Parallel Virtual Machine (PVM) version 3.4.4.

The fundamental physics and code structure of RELAP5-3D uses well known and validated technologies. For mathematical models, six equation model of finite difference method is used for two-phase fluid flow which includes liquid mass, vapor mass, liquid momentum, vapor momentum, liquid energy, and vapor energy equations. Closure relations in RELAP5-3D uses finite difference method for multi-dimensional heat transfer equations and an analytical nodal method for neutron kinetics. For numerical discretization both semi- and nearly-implicit schemes are used.

Since RELAP5-3D is well validated and already widely used in nuclear industry no specific issues are observed to be used immediately in the nuclear industry.

3.4.3 PRA capability/applicability

RELAP5-3D does not have embedded PRA capability. However, many of researches to couple RELAP5-3D and versatile PRA tool, RAVEN, has been supported by RISA Pathway. The Reactor Analysis and Virtualcontrol ENvironment (RAVEN) has been developed in INL which is a generic software framework that performs parametric and probabilistic analysis based on the response of complex system codes. RAVEN has highest versatility of coupling with other computer codes. Boiling Water Reactor (BWR) nuclear power plant black out accident scenario was studied to assess risk-informed method capability to assess impact of power uprate [12]. This study was extended to evaluate plant resiliency of accident tolerant fuel (ATF) and diverse and flexible coping strategies (FLEX) equipped PWR station black out and LOCA scenarios [13]. Recent RISA Pathway activity to develop dynamic

PRA model by coupling RELAP5-3D and RAVEN as stochastic tool showed good capability of use of RELAP5-3D for risk-informed analysis [14].

3.4.4 Documentation

Currently, INL is responsible on maintaining and developing RELAP5-3D. A set of manuals are released for major upgrade of the code. However, the manuals are only available to RELAP5-3D license holders. Released in 2018, following manuals are included for RELAP5-3D/Ver. 4.4.

Volume I: Code Structure, System Models and Solution Methods

Volume II: User's Guide and Input Requirements,

Volume II Appendix A: RELAP5-3D Input Data Requirements

Volume III: Developmental Assessment

Volume IV: Models and Correlations

Volume V: User's Guidelines

3.4.5 System requirements

RELAP5-3D/Ver. 4.4 has been tested in Dell cluster supercomputers with OpenSUSE 11.4 operating system, SGI cluster with OpenSUSE Enterprise Server 10, SUN and HP workstations with Open SUSE 11.4 Linux, and Personal Computers running both Windows XP and Windows 7. It is possible to build RELAP5-3D on other platforms with other compilers (not supported by development team).

RELAP5-3D does not officially support Mac OsX.

Comparison study between Linux and Windows are performed and documented [17].

3.4.6 Easy installation

RELAP5-3D development team provides installation guide with software package in CD/DVD once license is purchased. This guide includes system requirements, step by step installation process in various Linux/Unix and Windows system, RELAP5-3D running command and options, and independent fluid property library tool POLATE.

Installation of RELAP5-3D is comparably easy. Support from development team is also available.

3.4.7 Graphic user interface (GUI)

Currently, no official GUI is available for RELAP5-3D. However, independent from RELAP5-3D development team, SNAP (Symbolic Nuclear Analysis Package) is available for GUI [15]. SNAP consists of a suite of integrated applications designed to simplify the process of performing engineering analysis. SNAP currently includes support the CONTAIN, COBRA, FRAPCON-3, MELCOR, PARCS, RADTRAD, RELAP5 and TRACE analysis codes as a separate plug-in.

AptPlot, a free plotting tool, has capability of accessing RELAP5-3D plot data file. AptPlot is a WYSIWYG 2D plotting tool designed for creating production quality plots of numerical data and performing data analysis [16].

3.4.8 Version control

RELAP5-3D development team maintains released version history and all versions of the code. Table 3-1 is list of RELAP5-3D versions and release date. It is noted that versions before 4.0.3 are no longer available for release.

RELAP5-3D development team also provides release note and updated manuals with new version.

Comparison study between latest (4.4.2) and previous (4.3.4) versions [16] and between Linux and Windows [17] computer operating system was also performed.

3.4.9 V&V history

Records and history of RELAP5-3D V&V activities are very well documented and maintained by development team. The main validation activities are routinely updated with new version of the code and the result is documented in manual Volume III with the name of "Developmental Assessment". More than 50 validation phenomenological, separate and integral effect test cases are simulated. However, verification and regression tests are not officially listed and described in developer provided manual.

Table 3-1 RELAP5-3D version history

Version	Release date
RELAP5-3D Version 4.4.2	June 25, 2018
RELAP5-3D Version 4.3.4	October 9, 2015
RELAP5-3D Version 4.2.1	June 30, 2014
RELAP5-3D Version 4.1.3	October 1, 2013
RELAP5-3D Version 4.0.3	July 12, 2012
RELAP5-3D/Ver: 3.0.0 Beta	November 29, 2010
RELAP5-3D/Ver:2.4	October 05, 2006
RELAP5-3D/Ver:2.2	October 30, 2003
RELAP5-3D/Ver:2.0.3	August 21, 2002
RELAP5-3D/Ver:1.3.5	March 14, 2001
RELAP5-3D/Ver:1.2.2	June 26, 2000
RELAP5-3D/Ver:1.2.0	May 05, 2000
RELAP5-3D/Ver:1.1.72	October 28, 1999
RELAP5-3D/Ver:1.1.7	August 4, 1999
RELAP5-3D/Ver:1.1.0	November 23, 1998
RELAP5-3D/Ver:1.0.08	September 24, 1998
RELAP5-3D/Ver:1.0.05	September 19, 1997
RELAP5-3D/Ver:1.0.0	July 6, 1997

3.4.10 QA program

RELAP5-3D is maintained under a strict code-configuration system that provides a historical record of the changes in the code. Changes are made using a version control system that allows separate identification of improvements made to each successive version of the code. Modifications and improvements to the coding are reviewed and checked as part of a formal quality program for software. In addition, the theory and implementation of code improvements are validated through assessment calculations that compare the code-predicted results to idealized test cases or experimental results.

RELAP5-3D is compliant with the INL's Nuclear Quality Assurance (NQA-1) software quality assurance plan. This covers U.S. NQA-1 complaints. The related documents include "RELAP5-3D Development: Asset Maintenance Plan", "RELAP5-3D Development Software Management: Software Quality Assurance Plan" and "RELAP5-3D Development Software Configuration Management Plan". These documents are not available to public.

3.4.11 Web page

Under the guidance of INL, RELAP5-3D has its own webpage: http://relap53d.inl.gov/. The webpage includes brief information of the code, user group information, seminar and training notices, version release date, newsletters, user support programs and contact information.

3.4.12 User support

The INL is current main development team of the RELAP5-3D. INL has formed and maintaining International RELAP5 Users Group (IRUG) to identify and support improvement of code to worldwide users. IRUG members could be individual, organization or group that holds RELAP5-3D license. Annual IRUG meeting is also hosted by INL to share technical information and report RELAP5-3D latest status. Membership in the IRUG is open to any individual, organization or group that has an interest in using and advancing the RELAP5-3D code.

For the user feedback problems, user may submit issues and problems at RELAP5-3D webpage. The resolved issues are routinely reported at IRUG meeting.

3.4.13 Training program

RELAP5-3D development team regularly provides various training program. Most common training program is provided during IRUG meeting. Three days of training workshop provides fundamental to intermediate level of REALP5-3D training.

On-site training courses are normally conducted at the facilities of the sponsoring organization and may include students from other organizations. RELAP5-3D development team can also provide dedicated training program for specific institute or organizations.

The cost would depend on the length of the course and the specific training topics to be covered.

Training videos are also available with the DVD form.

3.4.14 License

RELAP5–3D is subject to export controls. Membership in IRUG or any non–U.S. citizens or organizations must have prior approval from the United States Department of Energy. Export control restrictions may limit or preclude the use of the code for certain countries, agencies, or companies. However, the commercial use will not extend to providing the code or modified versions to third parties without prior arrangement with INL.

Before Battelle Energy Alliance, LLC (BEA) can send a draft license agreement for RELAP5-3D, it must ensure that sending the code will not violate U.S. export control regulations. BEA must clear request organization, user and any other potential user of this code within your organization from an export control perspective. Various levels of participation are listed below. A request for source code requires that you provide justification why executable code will not meet your immediate needs. Once the export control review is complete, BEA will send you a draft license agreement.

Four different levels of membership available in license. Each has a different level of benefits, services, and membership fee as described below. All licensees receive an updated version of the code when a new version is released. Following fee schedule is effective for all license renewals starting June 1, 2018.

Members

A full member organization is the highest level of participation possible in the IRUG. Members receive the RELAP5–3D code in source form. Multiple copy use is allowed. Two levels of membership are available: Regular and Super User. Regular Member organizations receive up to 40 hours of staff assistance in areas such as model noding, code usage recommendations, debugging, and interpretations of results from INL RELAP5 technical experts. Super Users receive up to 100 hours of staff assistance.

- Annual License Fee (Regular User): $30,000 (domestic), $58,000 (foreign)
- Annual License Fee (Super User): $67,000 (domestic), $83,000 (foreign)

Multi–Use Participants

Multi–use participants are organizations that require use of the code but do not need or desire all the benefits of a full member. Participants receive the RELAP5–3D code in object form. Multiple copy use is allowed.

- Annual License Fee: $10,000 (domestic), $18,000 (foreign)

Single–Use Participants

Single–use participants are restricted to use RELAP5–3D on a single computer, one user at a time and receive the RELAP5–3D code in object form.

- Annual License Fee: $7,000 (domestic), $13,000 (foreign)

University Participants

University Participants can acquire a no fee license to RELAP5–3D with the restriction that the code executable only be used for educational purposes at the university. They receive the code in object form but do not receive any staff assistance.

3.5 Conclusion and Remark

RELAP5-3D is one of most matured nuclear safety analysis computer software and fully validated through worldwide use and licensing. As a RISA Toolkit, RELAP5-3D technology has been also proved by coupling with RAVEN PRA application. RELAP5-3D technology maturity assessment result are shown in Table 3-2. In terms of RISA Pathway activity, following remarks are addressed.

- Verification is needed for RELAP5-3D GFortran version.
- Lack of accident tolerance fuel cladding coating behavior analysis feature.
- Development is needed for risk-informed multi-physics BEPU capability.
- Code maintenance funding is unstable. This issue should be resolved to meet LWRS philosophy.
- Significant decrease of code developers and maintenance human resource.

Table 3-2 RELAP5-3D technology maturity assessment result

Requirements	Importance	Description	Technology Readiness Level (TRL)
Development level	High	RELAP5-3D is fully developed and validated. However, direct risk-informed application is not capable. Stable funding and continuous supply of human resources will be most important parameter for sustainable use and support.	8
Use of proven technology	High	Technologies used in RELAP5-3D are proven and reliable. No specific issue was found.	9
PRA capability/applicability	High	RELAP5-3D does not have PRA capability. Coupling with other PRA tools are still under development.	4
Documentation	Medium	Set of manual includes theories, code using method and validation record.	9
System requirements	Low	Different operating systems with various version are tested for full operation. Version Windows 10 is under-development. Comparison study between Linux and Windows are performed and documented. Not compatible with Mac OsX.	7
Easy installation	Medium	Installation method is well described in software package.	9
Graphic user interface (GUI)	Low	No official GUI is available. However, no issue was found by using non-GUI environment.	3
Version control	Low	New version includes all features of previous version and updates. Version history is well controlled by development team. Since RELAP5-3D development level is highest, no significant difference was reported between the versions.	9
V&V history	High	Validation by developer is well documented. However, list of regression and verification test is not listed.	7
QA program	High	RELAP5-3D follows development company’s QA program which also compliant with NQA-1.	9
Web page	Medium	Webpage has basic information and regularly updated as needed.	9
User support	High	Annual international RELAP5-3D user group (IRUG) meeting. However, user feedback response time is slow due to limited resources	7
Training program	Medium	Regular training program is organized by development team. Special training program could be also organized as needed.	9
License	Medium	License information is clear. Export control is needed for non-US customers.	9

4. RAVEN

4.1 Overview

Developed by INL, the RAVEN (Risk Analysis and Virtual Environment) software is a multi-purpose versatile tool for uncertainty quantification, regression analysis, probabilistic risk assessment, data analysis and model optimization. The main goal of the code is to perform parametric and stochastic analyses based on the response of complex systems codes. The code has been showing strong capability of coupling with other RISA Toolkit such as RELAP5-3D, NEUTRINO, BISON, MAAP, etc. RAVEN investigates the system response as well as the input space using standard sampling schemes (e.g. Monte Carlo, Latin hypercube, reliability surface search), but its strength is focused toward system feature discovery, such as limit surfaces using supervised learning techniques (artificial intelligence algorithms). The generated data can be analyzed using advanced statistical and clustering techniques both for time dependent and steady state problems. RAVEN can optimize large parameter spaces using its optimization algorithms.

The development of RAVEN started in 2012 with in the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program based on MOOSE (Multiphysics Object-Oriented Simulation Environment) framework. The initial development was aimed at providing GUI, logic control and dynamic risk analysis capabilities for newly developing thermohydraulic code RELAP-7 in INL. Currently, RAVEN is independent from MOOSE but still compatible with other software in MOOSE framework.

RAVEN was then improved by support from LWRS Risk Informed Safety Margin Characterization (RISMC, current RISA) Pathway to provide numerical solutions of probabilistic safety margin assessment. RAVEN can manage parallel dispatching of the software representing the physical model for advanced data mining applications. RAVEN also can apply for the artificial intelligence algorithms analysis to construct surrogate models of complex physical systems in order to perform uncertainty quantification, reliability analysis (limit state surface)and parametric studies.

4.2 Features

The main purpose of the RAVEN in RISA Pathway is to support system analysis code for risk-informed analysis. The software was designed for easy access and approach for any type of users from different science and engineering domain. Every aspect of RAVEN was driven by this singular principle from the build system to the application programming interface (API) to the software development cycle and input syntax. The development philosophy is to construct a capability of analysis to calculation flow at run-time, interpreting the user-defined instructions and assembling the different analysis tasks following a user specified scheme. Python was used to maximize flexibility and accelerate development. The major features of the RAVEN are as follows. Table 4-1 shows list of features and capabilities of RAVEN [18].

- Sampling of physical models for uncertainty quantification, dynamic PRA and reliability analysis
- Generation and use of surrogate models (artificial intelligence) and synthetic time-series
- Ensemble modeling (calculation flow embedding)
- Parameter optimization
- Data processing and analysis for time dependent and steady state statistical estimation, relational analysis, data mining, clustering and dimensionality reduction

The structure of RAVEN code consists “Distributions”, “Samplers”, “Optimizers”, “Models”, “Data Objects”, “Databases”, “Out stream”, “Steps”, and “Job Handlers”.

Table 4-1 List of RAVEN capability features

Area	List of features and capability
Forward propagation of uncertainties	• Monte Carlo sampling • Grid sampling • Stratified Sampling • Factorial design • Response surface design • Generalized Polynomial Chaos (gPC) with sparse grid collocation (SGC) • Generalized Polynomial Chaos (gPC) with sparse grid collocation (SGC) using the High Dimensional Model Representation expansion (HDMR) • General combination of the above sampling strategies
Advanced sampling methods	• Moment driven adaptive gPC using SGC • Sobol index driven HDMR integrated using SGC over gPC basis • Adaptive sampling for limit surface finding (surrogate and multi grid-based accelerations) • Dynamic event tree-based sampling (Dynamic Event Trees, Hybrid Dynamic Event Trees, Adaptive Dynamic Event Trees, Adaptive Hybrid Dynamic Event Trees)
Model capabilities	• Generic interface with external codes • Custom code interfaces (for MAAP, DYMOLA, RELAP5-3D, MOOSE, etc.) • Custom ad-hoc external models (build in python internally to RAVEN)
Creation and use of surrogate models	• Support Vector Machine • Gaussian process models • Linear models • Decision trees • Naive Bayes • Neighbors classifiers and regressors • Multi-dimensional interpolators • High dimension model reduction (HDMR) • Morse-Smale complex • Dynamic Mode Decomposition
Data analysis capabilities	• Data regression, clustering and dimensionality reduction • Custom generic post-processors • Time-dependent data analysis (statistics, clustering and time warping metrics) • Data plotting and visualization
Data management	• Data importing and exporting • Databases and high performance in-memory data storage

4.2.1 Distributions

RAVEN provides various options in one- or multi (N)-dimensional options to character probability distribution (or density) function (PDF). The options for one-dimensional (1-D) continuous distributions are: Beta, Exponential, Gamma, Laplace, Logistic, LogNormal, LogUniform, Normal, Triangular, Unifrom, Weibull and Custom one-dimensional distribution. For one-dimensional discrete distributions RAVEN has Bernoulli, Binomal, Geometric, Poisson, Categorical and Markov Categorical distributions. In case of multiple initial values are not independent but statistically correlated, multi-dimensional (N-D) option provides normal, weight and spline distribution.

4.2.2 Samplers

The "sampler" is the perturbation packages for RAVEN code. Three packages are available for target system input space: forward, dynamic event tree (DET) and adaptive sampling methods.

Forward Sampler The "forward sampler" performs sampling of the input space without exploiting. The "forward sampler" includes Monte Carlo (MC), Stratified, Grid Based, Sparse Grid Collocation, Sobol Decomposition, Response Surface Design of Experiment, Factorial Design of Experiment, Ensemble Forward Sampling strategy, Custom Sampling strategy.

Dynamic Even Tree (DET) Sampler The "dynamic event tree (DET) sampler" allows the scenario path can be selected by coupled (system) code through probabilistic environment. This method can have benefit to understand complex phenomena uncertainties. Four different DET samplers are available: Dynamic Event Tree (DET), Hybrid Dynamic Event Tree (HDET), Adaptive Dynamic Event Tree (ADET), and Adaptive Hybrid Dynamic Event Tree (AHDET). The ADET and the AHDET methodologies represent a hybrid between the DET/HDET and adaptive sampling approaches.

Adaptive Sampler The "adaptive sampler" is an alternative version of classical "forward" sampling method which basically reduces number of simulations by reduced order model (ROM) for large size of input/output parameter calculation cases. Currently, RAVEN provides following "adaptive samplers" : Limit Surface Search, Adaptive Dynamic Event Tree, Adaptive Hybrid Dynamic Event Tree, Adaptive Sparse Grid, and Adaptive Sobol Decomposition.

4.2.3 Optimizers

The "optimizer" defines method of optimization for the controllable input space and parameters to minimize or maximize the target system function. This function is an active learning process which can accelerate optimization time. Different to "sampler", the "optimizer" does not require sampling over distribution. Gradient Based Optimizer is available in RAVEN code.

4.2.4 Models

The "model" represents API application of phenomenon, component and system modeling. The "model" provides connection between input and output space. It also can be used for data mining by extracting key features. Currently RAVEN provides following six models for API application.

Code The "Code" model is an implementation of the model API that allows communicating with external codes. The communication between RAVEN and any driven code is performed through the implementation of interfaces directly operated by the RAVEN framework. For the coupling with other code, a Python interface is used to interpret the information coming from RAVEN and translates them to the input of the driven code without additional RAVEN modification. RAVEN also can optimize computation resources in cases of coupled with parallel and/or multi-thread code. Currently, model APIs are already implemented to couple with RELAP5-3D, RELAP-7, SASS, Modelica, and any MOOSE-based application.

External Model The “External model” allows the user to create own model in a Python file (imported, at run-time, in the RAVEN framework), its own model (e.g. set of equations representing a physical model, connection to another code, control logic, etc.). This model will be interpreted/used by the framework and, at run-time, will become part of RAVEN itself.

Reduced Order Model (ROM) The ROM (also called Surrogate Model) is a mathematical representation of a system, used to predict a selected output space of a physical system. The “training” is a process that uses sampling of the physical model to improve the prediction capability (capability to predict the status of the system given a realization of the input space) of the ROM. More specifically, in RAVEN the Reduced Order Model is trained to emulate a high-fidelity numerical representation (system codes) of the physical system: Two general characteristics of these models can be generally assumed (even if exceptions are possible). The higher the number of realizations in the training sets, the higher is the accuracy of the prediction performed by the reduced order model; and The smaller the size of the input domain with respect to the variability of the system response, the more likely the surrogate model will be able to represent the system output space.

Hybrid Models The “Hybrid model” is to combine ROM and any other models such as “Code” and “External model”. The ROMs will be “trained” based on the results from the high-fidelity model. The accuracy of the ROMs will be evaluated based on the cross-validation scores, and the validity of the ROMs will be determined via some local validation metrics. After these ROMs are trained, the “Hybrid model” can decide which of the model will be executed based on the accuracy and validity of the ROMs.

Ensemble Models The “Ensemble model” is to create a chain of models (whose execution order is determined by the Input/Output relationships among them). If the relationships among the models evolve in a non-linear system, a Picard’s Iteration scheme is employed.

Post-Processors The “Post-Processor” model represents the container of all the data analysis capabilities in the RAVEN code. This model is aimed to process the data (for example, derived from the sampling of a physical code) in order to identify representative Figure of Merits. Post-Processors allows performing statistical and regression/correlation analysis, data mining and clustering, reliability evaluation, topological decomposition, etc.

4.2.5 Data Objects and Databases

“Data objects” and “Databases” manages RAVEN generated data. The “Data object” is a set of data generated during RAVEN execution. These data objects can be used as input or output for specific model. Currently RAVEN has three different data type: Point set; History set; and Data set. RAVEN also can access directly to an external HDF5 format database for storing and retrieving necessary data. This “Database” can organize data in parallel or hierarchical.

4.2.6 Out stream system

“Out stream” is result export system in RAVEN by print out in text-base (XML, CSV, etc) or plotting through MatPlotLib library.

4.2.7 Steps

“Steps” provides a standardized way for the user to combine the entities reported above for the construction of any analysis. The “Step” is the core of the calculation flow of RAVEN and is the only system that is aware of any component of the simulation.

4.2.8 Job Handler

“Job Handler” coordinates and regulates the dispatch of jobs in RAVEN. It monitors and handles parallelism in the driven models interact with computing environment.

4.2.9 Coupling with different code

Flexible coupling capability is one of most strong features of RAVEN. Started from RELAP-7, many of codes in the field of thermal-hydraulics, safety analysis, neutronics, PRA, and MOOSE based-applications can be coupled, even with the code generated with different program languages such as FORTRAN, C++, Python, etc. Coupling interface is “GenericCode” to access input files and produces output in CSV format. Detail of coupling method and interface for specific code is documented in [18]. Current available coupling interface is as follows:

- RELAP-7 : Next generation nuclear system safety analysis code
- RALAP5-3D : Reactor excursion and leak analysis program. Coupling available with RELAP5/MOD3 as well.
- MOOSE based applications and vector post processor : Multiphysics object-oriented simulation environment applications including BISON
- Open Modelica : Open source complex physical system modeling simulation language
- Dymola : Modeling and simulation environment based on Modelica modeling language
- Cubit : Mech generator for MOOSE based applications
- Rattlesnake : MOOSE based radiation transport solver
- MAAP5 : Modular accident analysis program to simulate nuclear power plant accident.
- MAMMOTH : MOOSE based general reactor physics application
- MELCOR : Nuclear power plant accident progression modeling code
- SCALE : Reactor core physics and safety analysis code
- COBRA TF : Reactor subchannel thermal-hydraulics simulation code
- SAPHIRE : Systems analysis programs for hands-on integrated reliability evaluations for probabilistic risk and reliability tool.
- PHISICS : Time-dependent neutronics calculation code
- PHISICS/RELAP5-3D : Multi-coupling interface for both PHISICS and RELAP5-3D
- RAVEN : Allows execute RAVEN input file for driving with slave RAVEN calculation. Example of use could be optimization of surrogate model parameters in RAVEN then train and validate through slave RAVEN runs.

4.3 Verification and Validation Status

Since RAVEN is designed as non-physics application, no specific activities were performed for validation with existing experiments. While installation, RAVEN automatically performs a regression tests to confirm code is ready to perform correctly. A total of 733 regression tests are available in RAVEN framework. The test suits are continuously added as needed. The detail list of the tests can be found in reference [19]. The RAVEN regression test is consisting of three different type of tests: Requirement tests; Analytical tests; and Verification tests.

4.3.1 Requirement tests

Requirement test is to confirm required code development features, which are defined from developer and stakeholders, are correctly performing. A total of 35 regression tests are performed for following requirements.

- *6 Minimum requirements* including computer system type, memory, hard drive, compilers, computer language and version control
- *6 Functional requirements* including framework, input/output control, execution control, etc.
- *11 Usability requirements* including risk evaluation, analysis and mitigation features
- *9 Performance requirements* including infrastructure support for external code coupling, database control, computational memory control, etc.
- *2 System interface requirements* including external code coupling methods

RAVEN automatically generates RTM (Requirements Traceability Matrix) for above requirements during regression test. It is noted that these requirements are currently not available to public due to INL's software QA policy.

4.3.2 Analytical tests

The analytical test is to perform a benchmark with RAVEN and given analytical solutions during regression test. A total of 37 analytical tests is performed for following 12 analytical solution benchmark cases. Details of the test and solutions for benchmark is included in the reference [20].

- Projectile (vacuum, gravity) test
- Attenuation test
- Tensor polynomial (first-order) test
- Stochastic collocation with gamma distribution test
- Global Sobol sensitivity: Ishigami test
- Sobol G-Function test
- Risk importance measuring test
- Parabolas test
- Fourier test
- Optimization function test
- Processor load test

4.3.3 Verification tests

The verification test is a comparison test between previous and latest version of RAVEN. A total of 613 verification tests are documented. The test suits are divided into following areas and perform regression test for entire features RAVEN.

- 46 tests from RAVEN workshop
- 18 tests for CashFlow plug-in
- 18 tests for crow swig classes unit test
- 531 tests for main RAVEN features

4.4 Technical Maturity Assessment

Following information describes RAVEN technical maturity assessment results based on the requirements as the RISA Toolkit.

4.4.1 Development level

Initial stage of RAVEN development was aiming to provide graphic user interface, control logics and post-processing for RELAP-7 thermal-hydraulics code [21]. The generic mathematical framework for probabilistic risk assessment including was also implemented and tested with simplified PWR accident scenario. From 2013, more features on PRA capabilities are added, updated and demonstrated through scenario-based studies. Flexible coupling capability with external code is one of strong advantage in RAVEN code. Starting from RELAP-7, RELAP5-3D was successfully coupled and demonstrated for both classical and dynamic PRA. Other nuclear industry related codes and MOOSE based applications were also easily coupled and demonstrated. The available coupling interface RISA Toolkit are: BISON, RELAP5-3D, RELAP-7, MAAP, MELCOR, GRIZZLY, MASTODON, and SAPHIRE. Coupling with GOTHIC and NEUTRINO is under development.

Currently RAVEN is developing by INL. The code became open-source software and development is managed in GitHub repository hosting services. Through GitHub system, RAVEN developer and user can communicate, get feedback and release developer's version. Though the development was initiated in 2012, the first official release of RAVEN v.1.0 was May 2018. In May 2019, version 1.1 was released with following major updates:

- Full set of software QA program is prepared for RAVEN which is compliant with NQA-1 Level 3. However, the detail of QA documents is ready to apply for Level 2.
- Added probabilistic risk assessment plug-in which allows dynamic PRA and uncertainty quantification
- New interface to couple with additional external codes: COBRA TF, SAPHIRE, SCALE, PHISICS, MELCOR and RAVEN (RAVEN able to drive a RAVEN inner analysis)
- Python 3.x became default environment and related regression tests added.
- Update on Windows system installation procedure.
- Upgrade RAVEN yemplate input system for simplified analysis flow creation.
- Updated User Guide documentation
- Update on "DataObject" system to support unstructured data sets
- Additional features added: ARMA enhancements with signal clustering, Segmented ROMs capability, Multi-collinearity detection in statistical analysis and addition of the computation of standard errors, and Limit Surface performance improvement

Multi-physics analysis requires highest computational resources. Recent coupling work with RELAP5-3D already shown limitation in effective use of available INL supercomputing system by occupying thousands of CPUs. Parallel dispatching method significantly improve use of computation resources and will economize infrastructures of supercomputing host services.

Currently, reliability analysis of RAVEN uses limit surface model. However, evolution of risk in uncertainty space is not fully modeled which is important for fine risk-informed analysis to support decision making. Risk-weighted optimization method will support modeling of risk evolution in limit surface model and enhance applicability to RISA Pathway purpose.

4.4.2 Use of proven technology

RAVEN is generated with Python computer programming language. Python has advantage of developing mathematical models, graphic user interfaces and applications for use in various computer systems such as Linux, Windows and Mac OS. RAVEN needs specific library environment "Conda" Python library. As of 2019, Python program language has been widely used in all the area of science, engineering, graphics, software design, gaming, etc. The technology has been fully proven and easy to control.

The mathematical algorithms and models applied in RAVEN are fully validated and proved. The classical stochastic and statistic models are used. Uncertainty quantification and dynamic PRA needs sampling methods. RAVEN has both forward and adaptive sampling method are available along with various options inside. Other features such as reduce order modeling, statistical analysis and data mining also has various options which are mathematical matured technology.

4.4.3 PRA capability/applicability

Though the initial goal of the development was different, RAVEN has been evolved to the PRA tool. It has been showing strong capabilities on uncertainty quantification, reliability analysis, PRA, data mining, model optimization, reduced order modelling, etc. To use in dynamic PRA, RAVEN has equipped various sampling technologies. For the forward samplers, Monte-Carlo, grid-based, stratified and sparse grid collocation options are available. The samplers for the adaptive method are limit surface, surface search, adaptive dynamic event tree, adaptive hybrid dynamic event tree, adaptive sparse grid, and adaptive Sobol decomposition.

4.4.4 Documentation

INL has responsibility on maintaining and developing RAVEN. The set of manuals and test documents are available to the public. The manual is also included in software package which could be generated by Python and LaTex script. The manual used to update along with developer’s version release. The latest updated version was released in October 2019. The list of publicly available manuals and its contents is:

- RAVEN User Manual : Installation and running guide, input parameters, options, functions, coupling interfaces, plug-in generation, etc.
- RAVEN Theory Manual : Theoretical basis and algorithms implemented in the code.
- RAVEN User Guide : Tutorials and sample runs for all the features
- RAVEN Regression Test Description : Description of entire regression tests
- RAVEN Analytical Test Documentation : Benchmark data for analytical regression tests

The manuals could be generated during the installation in any OS type system.

Since RAVEN development was started as supporting software of RELAP-7, the features and goals of development are not same as starting of the project. The manual and related documents are then prepared by developers and updated as needed which partly not well balanced and stated. To use widely as RISA Toolkit, thorough revision and documentation expert proof reading will be needed.

4.4.5 System requirements

RAVEN is compatible with Linux (Ubuntu 16.04 and Fedora 22 or higher), Microsoft Windows (64-bit Windows 7, 10, and Windows Server 2012 R2 Standard) and Mac OsX (10.10.3 or higher). Python 3 and Conda library is necessary for RAVEN. Python 2 is compatible, but development is based on Python 3.

4.4.6 Easy installation

The default installation method is direct cloning dispatch from GitHub repository. The RAVEN webpage in GitHub as well as user manual contains detail step by step installation method for each computer operation systems. The result of installation tests and issues are recorded in https://github.com/idaholab/raven/wiki/install-testing-results. The advanced installation methods for developers are also available.

Installation of RAVEN is comparably easy. Support from development team is also available.

4.4.7 Graphic user interface (GUI)

Currently, no official GUI is available for RAVEN. Initially, MOOSE based GUI program PEACOCK has been developed to use with RELAP-7 interface which is currently not available.

The flow chart type GUI will support understanding complex modelling of RAVEN analysis and will facilitate new users' approach. In terms of RISA Pathway perspective, RAVEN GUI will increase usability of PRA to traditional system analysis code and will ease industrial deployment of risk-informed approaches.

For the plotting the output, RAVEN includes Python library MatPlotLib which allows producing two-/three-dimensional graph. More information is available at section 14.2 of user manual [18].

4.4.8 Version control

Two official stable version was released for RAVEN as shown in Table 4-2.

Table 4-2 RAVEN version history

Version	Release date
RAVEN v.1.1	May 10, 2019
RAVEN v.1.0	May 14, 2018

The developer's version is also available between v.1.0 and v.1.1. Through GitHub user may check available update and install latest changes.

RAVEN development team also provides release note and updated manuals with new version.

Comparison test is performed with new modification is added to developer's version in https://github.com/idaholab/raven/pulls.

The comparison study between latest (v.1.1) and previous (v.1.0) was performed and recorded as "log file". No specific report is provided for version comparison test.

4.4.9 V&V history

The detail of regression tests which could be the verification of RAVEN code are well documented and updated as necessary. The regression test suite includes analytical benchmark data which allows user can directly compare with analytical solution and RAVEN result.

4.4.10 QA program

RAVEN and RAVEN Plug-ins are strictly managed by INL's Software Quality Assurance (SQA) program. The program includes software development, maintenance and retirement plan as well as standardized method of capturing software requirements and changes through requirement traceability matrix (RTM). Changes are made using a version control system that allows separate identification of improvements made to each successive version of the code. Modifications and improvements to the coding are reviewed and checked as part of a formal quality program for software.

RAVEN is INL's Nuclear Quality Assurance (NQA-1) software quality assurance plan. This covers U.S. NQA-1 Level 3 (commercial grade – no safety) standard. The related documents include "RAVEN and RAVEN Plug-ins Software Quality Assurance and Maintenance and Operations Plan", "RAVEN

Software Design Description", and "RAVEN Software Requirements Specification and Traceability Matrix". These documents are not available to public.

4.4.11 Web page

Similar to INL developed MOOSE based applications, RAVEN is hosted by GitHub repository services (https://github.com/idaholab/raven). The GitHub is widely using for computer software development platform. General user is free of charge. The developer's page needs paid account. INL maintains its GitHub repository to support developing various open source software. The RAVEN GitHub page includes general information, install guide, developers' communication, bug and issue report, released versions, manuals, other documents, training and workshop information, etc. The RAVEN GitHub webpage is open to public.

The conventional style webpage https://raven.inl.gov/ is also available.

Update is needed to unify both web pages.

4.4.12 User support

The RAVEN development team maintains interactive "issue report" area in GitHub RAVEN webpage https://github.com/idaholab/raven/issues. To report the issue user should sign up GitHub first. The issue report has own format to provide better communication with users. The reported issue will be "open" then "closed" once the problem is solved. The resolved results are addressed as defects and corrected to developer's version. No specific document is published for resolved issues, but the resolution result is communicated with submitter and entire user group.

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Issue Description
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##### What did you expect to see happen?

##### What did you see instead?

##### Do you have a suggested fix for the development team?

##### Please attach the input file(s) that generate this error.  The simpler the input, the faster we can find the issue.

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For Change Control Board: Issue Review
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This review should occur before any development is performed as a response to this issue.
- [ ] 1. Is it tagged with a type: defect or task?
- [ ] 2. Is it tagged with a priority: critical, normal or minor?
- [ ] 3. If it will impact requirements or requirements tests, is it tagged with requirements?
- [ ] 4. If it is a defect, can it cause wrong results for users? If so an email needs to be sent to the users.
- [ ] 5. Is a rationale provided? (Such as explaining why the improvement is needed or why current code is wrong.)

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For Change Control Board: Issue Closure
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This review should occur when the issue is imminently going to be closed.
- [ ] 1. If the issue is a defect, is the defect fixed?
- [ ] 2. If the issue is a defect, is the defect tested for in the regression test system?  (If not explain why not.)
- [ ] 3. If the issue can impact users, has an email to the users group been written (the email should specify if the defect impacts stable or master)?
- [ ] 4. If the issue is a defect, does it impact the latest release branch? If yes, is there any issue tagged with release (create if needed)?
- [ ] 5. If the issue is being closed without a pull request, has an explanation of why it is being closed been provided?
```

It is however sustainable resource is necessary for continuous support from INL RAVEN development group.

4.4.13 Training program

RAVEN development team organizes training program as needed. In 2018, training workshop was organized along with RELAP5-3D user group meeting and training.

On-site training courses are normally conducted at the facilities of the sponsoring organization and may include students from other organizations. RAVEN development team can also provide dedicated training program for specific institute or organizations. The cost would depend on the length of the course and the specific training topics to be covered.

The presentation slides from training course are also available at https://github.com/idaholab/raven/wiki/workshop.

4.4.14 License

RAVEN is open source and free of charge for download and use. However, INL requires "Contributor License Agreement (CLA)" to maintain intellectual property for contribution from any person or entity. Detail of CLA is shown in Appendix E.

4.5 Conclusion and Remark

RAVEN has been widely used for uncertainty quantification, PRA, sensitivity study and data mining. It specially has strong capability of coupling with various computer software using in nuclear industry. Open source policy, plotting capability and output management is also notable features. As a RISA Toolkit, RAVEN has been proved by generating various dynamic PRA and coupling capability with RELAP5-3D. RAVEN technology maturity assessment result is shown in Table 4-3. In terms of RISA Pathway activity, following remarks are addressed.

- Improvement is necessary for use of high computational resource (CPU) for multi-physics simulation with RELAP5-3D. This is one of urgent issue as RISA Toolkit
- Risk-weight optimization will help improving PRA capability and accuracy
- For entire manual, thorough update and proof reading should be done
- Flow chart type GUI will facilitate for RISA Toolkit industrial deployment
- Coupling interface is needed for VERA-CS, FRANCON/FRAPTRAN, CFAST, FDS and GOTHIC RISA Toolkit.
- More industrial use and verification is needed.

Table 4-3 RAVEN technology maturity assessment result

Requirements	**Importance**	**Description**	**Technology Readiness Level (TRL)**
Development level	High	Fundamental development for RAVEN technology is mostly finalized. Coupling with RELAP5-3D is fully demonstrated. Coupling interface with another RISA Toolkit (VERA-CS, FRAPCON/FRAPTRAN, CFAST, FDS, GOTHIC) is still missing. Computational capacity improvement and additional model needs to be upgraded. More industrial use and V&V is needed.	7
Use of proven technology	High	Technologies used in RAVEN. No specific issue was found.	9
PRA capability/applicability	High	RAVEN has been showing good result for classical and dynamic PRA by coupling with other codes. Use of risk-weighted optimization method will improve applicability to RISA concept.	8
Documentation	Medium	Set of manual includes theories, user guide, and verification activities. However, thorough revision and proof reading is necessary.	6
System requirements	Low	Different operating systems (Linux, Windows and Mac OsX) with various version are tested for full operation. Operating system comparison study was not performed.	7
Easy installation	Medium	Installation method is well described in software package.	9
Graphic user interface (GUI)	Medium	No official GUI is currently available. Plotting capability is included. However, since RAVEN aims coupling with another codes, flow chart type GUI will facilitate understanding complex modeling.	5
Version control	Medium	New version includes all features of previous version and updates. Developer’s version is also available. No specific report is provided for version comparison study.	7
V&V history	Low	No validation activity is needed for RAVEN. Verification is done through regression tests.	9
QA program	High	RAVEN follows development company’s QA program which also compliant with NQA-1.	9
Web page	High	RAVEN is open source and web page is important for dissemination and management. Both GitHub and conventional style web pages are existing. However, general information should be updated.	7
User support	High	GitHub web page is main method for user support. Reported issues management is well controlled. However, sustainable resource is necessary for continuous support to INL RAVEN team	7
Training program	Medium	Training program is organized by development team as needed.	8
License	Medium	RAVEN is open source. Contributor license agreement is needed for participating development.	9

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APPENDIX A. RISA TOOLKIT TECHNOLOGY MATURITY ASSESSMENT MATRIX

Requirements	Importance	Description	Technology Readiness Level (TRL)
Development level			
Use of proven technology			
PRA capability/applicability			
Documentation			
System requirements			
Easy installation			
Graphic user interface (GUI)			
Version control			
V&V history			
QA program			
Web page			
User support			
Training program			
License			

APPENDIX B. RELAP5-3D VERIFICATION TEST SUITES

Input file	Description
2phspump	Tests two-phase pump head degradation as a function of void fraction alone and as a function of void fraction and pressure.
3dflow	Simulates 3-D flow of single-phase liquid, single-phase vapor, or two-phase flow in a 3x3x3 Cartesian grid with either 1-D or 3-D momentum equations.
ans	Tests decay heat options with the point kinetics model, fission power types, fission product types available with each ANS standard, and the G-factor contribution to the decay heat.
bornm	Tracks a square wave in boron concentration through a constant area pipe with and without Godunov method.
crit	Tests the metal water reaction model for steam flowing past the right surface of a cylindrical heat structure.
cyl3	Tests Ransom-Trapp and Henry-Fauske critical flow models for a range of stagnation conditions including subcooled, two-phase, and superheated in a small horizontal pipe. Also tests cases with no choking allowed and homogeneous flow.
decay	Tests reactor kinetics and decay heat calculation for long time period.
det	Four loop PWR with small break LOCA in one loop.
det_new	Upgraded four loop PWR with small break LOCA in one loop.
duklem	Tests the CCFL model using Dukler-Smith air-water countercurrent flow data. Wallis, Kutateladze, and Bankoff correlations are tested.
ecomix	Models a portion of the cold leg of a typical PWR during ECC injection.
edhtrkm	Edward's pipe simulates a rapid blowdown of a pipe. Includes extras: reactor kinetics, heat structure cosine temperature problems, and all control variables types, but shaft. Cases use fluids: h2o, d2o, h2on, h2o95, hen, and an air/water mixture.
eflag	Simulates blowdown of one vessel into another to check the effect of the e-flag on the thermodynamic state in the downstream vessel.
enclss	Steady-state calculation of a graphite stack using the heat conduction enclosure model.
fric	Tests various single-phase wall and junction friction models. Cases include turbulent flow with and without heated wall effect, laminar flow with and without shape factors, abrupt area change options, and user input equations for wall and form friction.
fwhtr	Represents a tube-in-shell feedwater heater.
fwpress	Flexible wall air test case (1atm, 1sec).
Fwstiff	Flexible wall air test case (1sec).
gota27	Simulates rod-to-rod radiation in a 64-rod bundle in low-pressure steam using radiation enclosure model.
hse	Simulates two-phase flow through a horizontal tee with offtakes coming off the top, bottom, or side face of the horizontal pipe.
ht_expl_fluid	Simple model of pipe and heat structure fluid heat transfer calculation with explicit time step
ht_imp_fluid	Simple model of pipe and heat structure fluid heat transfer calculation with implicit time step
httable	Simple model of a pipe and heat structure exercising structure BC related to heat flux and heat transfer coefficient.
httest	Simple model of a pipe and heat structure that varies IC and BC to achieve various heat transfer regimes for heat transfer packages 1, 111, and 134. Also tests the non-equilibrium volume option
hxco2m	Models a once-through heat exchanger with PbBi on the shell side and supercritical carbon dioxide inside the tubes. Tests the normal and alternate heat structure-fluid coupling models in steady-state.
jetjun	Simulates insurges and outsurges of liquid into a pressurizer with and without the jet junction model.
jetpump	Tests jet pump performance over a range of suction and driveline flows.
l31acc	Represents the accumulator response during a slow depressurization during LOFT L3-1.

I2-5emA	Tests Appendix K options during a LOFT Experiment L2-5, which simulates a loss-of coolant accident initiated by a large break.
movetest	Hydro base test
neptunus20m	Models pressurizer insurge/outsurge experiment with spray.
pack	Vertical fill problem tests water packing model when subcooled liquid is injected into superheated steam from below. Uses semi- and nearly-implicit timesteps.
pitch	Tests an inertial check valve with movement.
radialm	Models pure radial, symmetric flow problem in a 2D hollow cylinder. There is no azimuthal flow.
rcpr	Tests the performance of a recompressing compressor in a supercritical CO2 cycle.
refbunm	Tests two-phase flow and heat transfer with horizontal and vertical bundles that exercise the Groeneveld and PG CHF correlations and correlations for narrow, rectangular channels.
reflecht	Flecht-seaset developmental assessment case
regime	Tests the standard horizontal and vertical flow regimes by adjusting flow boundary conditions through a simple pipe. Both the pre-CHF and post-CHF regimes are tested for the vertical pipe.
rigidbodym	Models pure azimuthal, symmetric flow problem in a 2D hollow cylinder. There is no radial flow.
rthetam	Models flow in a 2D hollow cylinder with symmetric flow in both the radial and azimuthal flow directions.
rtsampnm	Based on typpwr, tests radio-nuclide transport model and the axial heat source options using nodal kinetics.
rtsamppm	Based on typpwr with uses point kinetics, tests various axial heat source options, including those from tables, control variables, and reactor kinetics. Tests also the radio-nuclide transport model.
slab3	Tests the metal water reaction model for steam flowing past the right surface of a rectangular heat structure.
sphere3	Tests the metal water reaction model for steam flowing past the right surface of a spherical heat structure.
state	Tests various fluid states, including subcooled liquid, two-phase, superheated vapor, high pressure liquid, high-temperature vapor, and supercritical, for h2o, h2on, d2o, and new helium.
tdvtdj	Time dependent junction and volume control variable test
todcnd	Models heat transfer from hot wall with the reflood and two-dimensional heat conduction models.
turbine9	Multi-stage steam turbine with moisture separation. All four types of turbines are tested.
typ12002	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds.
typ_kindt	TYPPWR input model with nodal kinetics, Krylov solver, and independent kinetics timestep.
valve	Models opening and closing of all valves, except relief.
varvol2	Uses the variable volume model and a general table to vary the fluid volume of a single liquid-filled volume.
do_nothing	Tests if zero flow and zero heat transfer are maintained in a rectangular solid of 3x5 volumes. Constructed of 5 volume pipes connected by multiple junctions.
iter1	Test generation of boiling curve as heat flux varies in a pipe.
nothing_trans	Tests if zero flow and zero heat transfer are maintained in a rectangular solid of 3x3x5 volumes. Constructed of 5 volume pipes connected by multiple junctions.
pvmcore	Tests ability of RELAP5-3D to run the vessel interior of the modified Christensen model
pvmcs	Edward's pipe problem adapted to test control system
pvmnonc	Parallel pipes tests multiple connections to TDV and multiple noncondensables
pvmpt	A version of TYPPWR (test 40) that tests point kinetics
cpl_det	A simplified version of TYPPWR that tests the detector model with point kinetics
cpl_det_new	Same as cpl_det with modified weighting factors and attenuation coefficients.
cpl_new_sa	A version of TYPPWR that tests the detector model with nodal kinetics
cpl_pvm_core	Christensen model domain decomposed into two semi-implicitly coupled regions, one with the center of the pipe representing the core, the other with the upper and lower portions.

cpl_pvmcs	Edward's pipe problem adapted to test control system coupling
cpl_pvmeda	Edward's pipe problem split in half to test asynchronous coupling
cpl_pvmedca	Edward's pipe problem split in half to test asynchronous explicit conserving coupling
cpl_pvmedcs	Edward's pipe problem split in half to test synchronous explicit coupling
cpl_pvmnd	A version of TYPPWR that tests nodal kinetics coupling
cpl_pvmnonc	Parallel pipes tests multiple connections to a coupling TDV and multiple noncondensables
cpl_pvmpt	A version of TYPPWR that tests point kinetics coupling

APPENDIX C. RELAP5-3D REGRESSION TEST SUITES

Input file*	Description
2ppumpmod.i	Tests two-phase pump head multiplier versus void fraction test.
2D1Flex.i	3-D volume with flexible inner pipe wall at radius 1.
3dflow.i, 3dflown.i	Simulates 3-D flow of single-phase liquid, single-phase vapor, or two-phase flow in a 3x3x3 Cartesian grid with either 1-D or 3-D momentum equations.
ans05.i, ans79.i, ans94.i	Tests decay heat options with the point kinetics model, fission power types, fission product types available with three ANS standards (ANS79-1, ANS94-1 and ANS05-1), and the G-factor contribution to the decay heat.
arteryFlex.i	Sending pulse down an artery
cstest1.i, cstest2.i	Simple loop with a steam generator, reactor, pump and pressurizer. Two tests: steady-state; and transient.
cyl3.i	Tests Ransom-Trapp and Henry-Fauske critical flow models for a range of stagnation conditions including subcooled, two-phase, and superheated in a small horizontal pipe. Also tests cases with no choking allowed and homogeneous flow.
ed_71.i	Edward's pipe problem test for constant temperature extrapolation for liquid super heat case
edair.i	Edward's pipe problem test with air
edallvrf.i	Edward's pipe problem test with cartesian neutronics case and 199-backall feature
edboron.i	Edward's pipe problem test with boron
edhtng1.i	Edward's pipe problem test with rectangular geometry space kinetics
edhtng2.i	Edward's pipe problem test with rectangular geometry space dependent reactor kinetics at steady-state
edhtng4.i	Edward's pipe problem test with rectangular grid neutronics transient case
edhtng5.i	Edward's pipe problem test with rectangular grid neutronics steady-state case. User feedback option
edhtng7.i	Edward's pipe problem test with rectangular grid neutronics steady-state case
edhtng8.i	Edward's pipe problem test with rectangular grid neutronics steady-state case. Different time option with edhtng7.i
edhtrkd.i	Edward's pipe simulates a rapid blowdown of a pipe. Coolant for heavy water
edhtrk.i	Edward's pipe simulates a rapid blowdown of a pipe. Coolant for light water
edhtrkm.i	Edward's pipe simulates a rapid blowdown of a pipe. Machine-dependent binary test.
edhtrkn.i	Edward's pipe simulates a rapid blowdown of a pipe. Coolant for light water. Nearly-implicit scheme
edhtrt.i	Edward's pipe simulates a rapid blowdown of a pipe with point kinetics tabular data
edstack.i	Stacked deck containing edhtrk, edrst, and edstrip
enclss.i	Steady-state calculation of a graphite stack using the heat conduction enclosure model.
fldrn2.i	Water fill into steam problem
fwhtr.i	Represents a tube-in-shell feedwater heater.
gota27	Simulates rod-to-rod radiation in a 64-rod bundle in low-pressure steam using radiation enclosure model.
hex2d1.i	IAEA benchmark hexagonal test case. Steady-state and user feedback option.
hex2d.i	IAEA benchmark hexagonal test case. Edward's pipe with hexagonal grid space kinetics. Steady-state.
hex2dk_dt0.i	IAEA benchmark hexagonal test case. Hexagonal mesh with Krylov solver. Steady-state. Uses kinetics time step control.
hex2dk.i	IAEA benchmark hexagonal test case. Hexagonal mesh with Krylov solver. Steady-state.
hstest.i	Typical four loop PWR with a small break LOCA.
Jetpmp.i	Tests jet pump performance over a range of suction and driveline flows.

k3200nk.i	RBMK-100 Kursk NPP nodal kinetics model.
lsemi8x8x16.i	LOFT 12-5 post test analysis case.
marpzd4.i	EHWR-NPR Pressurizer model to simulate behavior of gas driven pressurizer.
mulrst1.i, mulrst2.i	Edward's pipe with multiple restarts file
nc.i	Test non-condensible properties for the CO2, CO and O2
neptunus20.i	Models pressurizer insurge/outsurge experiment with spray.
outsurge.i	Pressurizer outsurge test
plotcsv2strip.i	Edward's pipe test plot-output file format with CSV.
ptkin.i	Point kinetics test
pump2.i	Two loops with pump test case.
refbun.i	Tests two-phase flow and heat transfer with horizontal and vertical bundles that exercise the Groeneveld and PG CHF correlations and correlations for narrow, rectangular channels.
reflood.i	Reflood test at 5 pipe cells with a hot wall
rencheck.i	Renodalization test
rk.i	Reactor kinetics test
rpump.i	Recompressing compressor model test
rtsampn.i	Radio-nuclide case 1 nodal kinetics. Nearly with default solver
rtsampp.i	Radio-nuclide case 4 point kinetics. Nearly with default solver
scw.i	Supercritical water reactor test
slab3.i	Metal water reaction test
slabl.i	Metal water reaction on left surface test
sphere.i	Metal water reaction test with spherical heat structure.
sschf2.i	Steady state CHF test with over-ride on CHF control
ssctrl3.i	Steady state solution control test with over-ride
sstrip1.i	Steady state mode trip test with over-ride of steady state trip
todcnd.i	Models heat transfer from hot wall with the reflood and two-dimensional heat conduction models.
turbine9.i	Multi-stage steam turbine with moisture separation. All four types of turbines are tested.
typ1200295n.i, typ1200n295n.i	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds. Coolant H2O95N.
typ12002.i, typ1200n2.i	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds. Coolant H2O.
typ12002nj.i	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds. Coolant H2O. Use New Jacobian method.
typKryNemSS95n.i	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds. Coolant H2O95N. Use Krylov solver.
typKryNemSS.i	Models small-break LOCA in a typical pressurized water reactor for 1200 seconds. Use Krylov solver.
typpwr.i	Standard small-break LOCA in a typical pressurized water reactor model.
typpwr2.i	Models small-break LOCA in a typical pressurized water reactor. The maximum time step (0.10 sec) is the value used in the simulator models by DS&S. The break junction uses the abrupt area change model. The core heat slabs follow current mesh practices. Use detector input data.
typpwr3d2.i	Models small-break LOCA in a typical pressurized water reactor. The maximum time step (0.10 sec) is the value used in the simulator models by DS&S. The break junction uses the abrupt area change model. The core heat slabs follow current mesh practices.

typpwr952.i	Models small-break LOCA in a typical pressurized water reactor. The maximum time step (0.10 sec) is the value used in the simulator models by DS&S. The break junction uses the abrupt area change model. The core heat slabs follow current mesh practices. Use H2O95.
typpwrn2.i	Models small-break LOCA in a typical pressurized water reactor with nearly implicit method
typpwrr295n.i	Models small-break LOCA in a typical pressurized water reactor. The maximum time step (0.10 sec) is the value used in the simulator models by DS&S. The break junction uses the abrupt area change model. The core heat slabs follow current mesh practices. With LSOR solver. Use H2O95.
typpwrr2.i	Models small-break LOCA in a typical pressurized water reactor. The maximum time step (0.10 sec) is the value used in the simulator models by DS&S. The break junction uses the abrupt area change model. The core heat slabs follow current mesh practices. With LSOR solver.

*Restart files are not listed

APPENDIX D. RELAP5-3D VALIDATION TEST SUITES AND RESULTS

Phenomenological validation tests

Test case	Models validated	Description and evaluation
Water faucet	Hydro numerics, gravity, momentum equation	The water faucet is a simple problem with water fall inside of a a vertical pipe. Fluid velocity and liquid fraction are correctly predicted over the entire pipe length. Results are identical for both semi- and nearly-implicit calculations. Evaluation level was <u>excellent.</u>
Water over steam (1-D)	Gravitational head, liquid level	The water over steam problem is a vertical pipe that starts with saturated water at the top and saturated steam at the bottom then the water falls to the bottom. The code calculations using a one-dimensional pipe component were <u>reasonable</u> using both semi- and nearly-implicit methods.
Water over steam (3-D)	Gravitational head, liquid level	With three-dimensional water over steam test model, the results from semi-implicit method was evaluated as <u>reasonable</u>. Calculation with nearly-implicit calculation was evaluated as <u>minimal</u>.
Fill and drain	Level tracking	The fill-drain problem is to assess liquid level tracking. A vertical pipe is filled with water then drained. Evaluation result was <u>excellent</u> for both semi-implicit and nearly-implicit method calculations.
Bubbling steam through liquid	Entrainment, two-phase level	The bubbling steam through liquid case injects steam with increasing velocity at the bottom of a water column to investigate liquid entrainment and two-phase level. Without the mixture level tracking model, fairly uniform void fraction profile was maintained below the two-phase level. The evaluation result was <u>reasonable</u>. Using mixture level tracking model, the calculated level results agreed qualitatively with the expected behavior, as levels were established as the steam mass flow rate was increased incrementally. <u>Minimal</u> level was given for semi-implicit solution method since an unusual void profile was calculated later in the transient. For the nearly-implicit method, the performance was <u>reasonable</u> until about 800 seconds, then became <u>insufficient</u> as a large error in the system mass appeared.
Manometer	Noncondensibles, wall friction, liquid level, oscillations	The manometer problem models water flowing between two vertical columns. Both semi- and nearly-implicit calculations showed identical and evaluated as <u>excellent</u>. The code accurately predicted the amplitude and period of the oscillatory parameters. However, the calculation results were evaluated as <u>minimal</u> level without mixture level tracking model due to low amplitude with long period.
Gravity wave (1-D and 3-D)	Horizontal stratification, force term	The gravity wave problem is a horizontal pipe that has water at a slightly increasing height along the length, causing a sloshing flow back and forth. The evaluation result was <u>reasonable</u> for both one- and three-dimensional components by generating an oscillating wave at nearly the analytic solution speed. The semi-implicit solution scheme exhibited a small instability in the pressure solution later in time that was not present with the nearly-implicit solution scheme.
Pryor pressure comparison	Water packing	The Pryor pressure problem is a steam-filled horizontal pipe that is filled from one end with water to investigate the water packing phenomenon. <u>Reasonable</u> results were shown in both semi- and nearly-implicit methods. Several pressure spikes were observed due to the presence of over-condensation and the resulting water-packing of the volumes.
Core power	Decay power	The core power case used three individual problems to assess the 1979 decay heat model. Both semi- and nearly-implicit calculations were found to be in <u>excellent</u> agreement with the standard.

Point kinetics ramp	Point kinetic	The point kinetics ramp problem assessed the point kinetics model response to a reactivity insertion. Assessment judgments of excellent were assigned for both the semi- and nearly-implicit calculations.
Pure radial symmetric flow (3-D)	3-D momentum equations	Three problems were used to test the numerics for the multi-dimensional hydraulic component. For the pure radial flow case, both calculations were in excellent agreement with the exact solution for the radial velocity and pressure distributions.
Rigid body rotation (3-D)	3-D momentum equations	For the rigid body rotation case, the semi-implicit calculation agreed with the exact solution for the radial and azimuthal velocities. Although the calculated radial pressure distribution did not exactly match the exact solution, the results were judged to be in excellent agreement. The nearly-implicit calculation did not agree with the exact solutions, particularly for the pressure distribution. The results were also not symmetric in the azimuthal direction, which indicates that there is an error in the nearly-implicit numerical scheme for the multi-dimensional component.
R-theta symmetric flow (3-D)	3-D momentum equations	For the r-theta symmetric flow case, the semi-implicit calculation was in excellent agreement with the exact solution for the radial velocity, azimuthal velocity, and radial pressure distributions. The nearly-implicit calculation did not agree with the exact solutions, particularly for the pressure distribution. The results were also not symmetric in the azimuthal direction, which indicates that there is an error in the nearly-implicit numerical scheme for the multi-dimensional component. The velocity comparisons were evaluated as reasonable, while the pressure prediction was minimal.
Conduction enclosure	Conduction enclosure	Three cases addressed the conduction enclosure model; as pure heat transfer tests, only the semi-implicit solution scheme was used for the comparisons. The conduction enclosure problem was a two-dimensional, steady-state heat conduction problem. Excellent agreement between the analytical solution and the code calculation, with a relatively small number of nodes being able to give an approximation for the steady state temperature of the plate with less than 1% error throughout the domain.
Conduction enclosure: 1-D transient model	Conduction enclosure	The conduction enclosure one-dimensional transient simulated axial conduction in a cylinder. Overall, the agreement between the calculation and the analytical solution was excellent.
Conduction enclosure: 2-D transient model	Conduction enclosure	Transient cooling of a two-dimension plate was simulated by the final conduction enclosure problem. Excellent evaluation results were shown by comparison with the analytical solution. At one hour, the maximum error was less than 0.9%, and by three hours the error dropped to less than 0.07%.
Cladding oxidation	Metal-water reaction	The cladding oxidation case simulated a chemical reaction between the coolant and structural material in cylindrical, rectangular, and spherical heat structures. The agreement between the calculations and the analytical solutions was excellent.

Separate effect validation tests

Test case	Models validated	Description and evaluation
Edwards-O'Brien blowdown test	Vapor generation, flashing, critical flow, pressure wave propagation	The objective of Edwards pipe problem tests basic rapid blowdown phenomena in a simple straight pipe geometry. The single-phase choked flow initiated at the break location then the pipe undergoes rapid depressurization and propagation of a pressure wave along the pipe. As the pipe rapidly depressurizes, flashing occurs along the pipe, resulting in two-phase break flow until the pipe is depressurized and essentially empty. Both the semi- and nearly-implicit calculated pressure results were in reasonable agreement with the test data. Although measured break flow data were not available for this test, the fact that the pressure was well-calculated indicates that the break flow was likely well-calculated also.
Marviken critical flow test 21	Subcooled critical flow, saturated liquid critical flow	Four critical flow tests performed in the Marviken facility were used to investigate various aspects of the critical flow model. Test 21 assessed the subcooled choked flow model. Both the semi- and nearly-implicit solution scheme calculations were in reasonable agreement with the data overall, although during the first 15 s the agreement was evaluated as excellent.
Marviken critical flow test 22	Subcooled choking model, flashing, two-phase level	Test 22 addressed subcooled and saturated choked flow. Both the semi- and nearly-implicit solution scheme calculations were in reasonable agreement with the data. The mass flow rate was somewhat over predicted early in the transient, and over predicted later, with an overall assessment evaluation as reasonable. As a consequence of the mass flow rate prediction, the vessel pressure was under predicted in the early portion of the transient and over predicted later, but again the overall evaluation result was reasonable. Excellent level was given for predicting the temperature profile in the test vessel and reasonable for the mixture density prediction.
Marviken critical flow test 24	Subcooled choking model, flashing, two-phase level	Test 24 also addressed subcooled and saturated choked flow. Results using the semi- and nearly-implicit solution schemes were very similar. RELAP5-3D under predicted the vessel pressure at the early stage of transient then turned to over-prediction. Prediction for vessel pressure was evaluated as reasonable. The calculations were in excellent agreement during subcooled flow, but under predicted the mass flow rate somewhat with saturated conditions; the overall mass flow rate was evaluated as reasonable. Prediction of the temperature profile in the test vessel was reasonable. For the mixture density prediction, it was reasonable until 24 seconds then became minimal compare to the data.
Marviken jet impingement test 11	Saturated vapor critical flow, interfacial drag in bubbly/slug, pool boiling, void profile	Marviken JIT 11 assessed high quality vapor choked flow. The calculated results for both the semi- and nearly-implicit solution schemes were very similar and were judged to be in excellent agreement with the test data. With the specified pressure boundary condition, the code did an excellent job of matching the critical flow data.
Mobby-Dick air-water	Critical flow	The final critical flow assessment used a Moby Dick air-water experiment to investigate two-phase, two-component flow. Both semi- and nearly-implicit calculations were evaluated as minimal predict the pressure drop near the choking plane in the presence of an air-water mixture because of a significant under prediction of the two-phase pressure drop upstream of the choking plane. The unavailability of detailed information on the facility or experiment resulted in larger uncertainties for this test than most others.
Christensen test 15	Subcooled boiling heat transfer, void profile	Christensen Test 15 assessed the interphase mass transfer and wall heat flux partitioning models. The calculations were evaluated as excellent agreement with the measured data, correctly predicting the void fractions over the entire length of the test section.
GE level swell, 1ft, Test 1004-3	Vapor generation, interphase drag, two-phase level	Two GE level swell experiments were used to evaluate the performance of the vapor generation, interphase drag, and two-phase level models. For the 1-ft test, both the semi- and nearly-implicit calculations were in reasonable

		agreement with the experiment data. All major trends and phenomena were correctly modeled. Void fractions in the liquid region tended to be slightly over predicted, while the void fraction in the vicinity of the mixture level was generally low. The code correctly reproduced the initial level swell. Users should be aware of the possibility of localized unphysical flow regime and void fraction inversions in the region around the mixture level.
GE level swell, 4ft, Test 5801-15	Vapor generation, interphase drag, two-phase level	For the 4-ft test, both calculated results were judged to be in reasonable agreement with the experimental data. The code tended to predict a sharper mixture level for most of this transient, with a lower void below the level and a higher void above. However, the major trends were correctly predicted. A limited sensitivity study indicated that the results were not significantly different whether the mixture level tracking model was used or not.
Bennett heated tube tests 5358, 5294, and 5394	Non-equilibrium heat transfer, CHF, subcooled boiling, steam cooling	Bennett heated tube Tests 5358, 5294, and 5394 assessed the CHF model in RELAP5-3D. Both the semi- and nearly-implicit calculations were reasonable in predicting the CHF position even though they predicted the position slightly earlier than the measured data in the cases of the low and high mass fluxes and slightly later in the case of the intermediate mass flux.
ORNL THTF tests 3.07.9B, 3.07.9N, and 3.07.9W	CHF, film boiling, steam cooling	The ORNL THTF tests assessed the CHF and film boiling heat transfer models in RELAP5-3D. In general, the CHF and film boiling heat transfer predicted by both solution schemes were in reasonable agreement with the experiment data for the four ORNL THTF tests. For the 3.07.9W test case, the predicted rod temperatures were in reasonable agreement with the experiment data, and the code did a reasonable for predicting CHF. For the 3.07.9B and 3.07.9N cases, the CHF was reasonably predicted but the rod temperatures were only in minimal agreement with the measured data, being over predicted in 3.07.9B and under predicted in 3.07.9N.
ORNL THTF test 3.09.10	Void profile in rod bundles, radiation heat transfer	For the ORNL 3.09.10I test, the predicted void fraction and gas temperature were in excellent agreement with the experiment data, and the rod surface temperature was in reasonable agreement. Grid spacers were only accounted for in the RELAP5-3D model through the CHF correlation. The grid spacers appeared to have a significant effect on the rod surface temperatures in the ORNL tests, leading to discontinuities in the temperature response.
Royal Institute of Technology tube test 261	CHF	RIT Tube Test 216 was used to assess the default CHF and PG-CHF ("power" form) models in RELAP5-3D. Both the semi- and nearly-implicit calculations did a reasonable for predicting the CHF position, even though it predicted the position slightly later in the case of the default CHF model and earlier in the case of the PG-CHF model than the measured data.
FLECHT-SEASET Test 31504	Reflood model (low reflood rate), two-phase level, natural circulation, subcooled boiling, steam cooling, quench front, interphase evaporation, entrainment, CCFL, condensation heat transfer	FLECHT SEASET Test 31504 investigated the performance of the reflood model at a low flooding rate. Both semi- and nearly implicit calculations were judged to be in reasonable agreement with the measured data. Predicted rod surface temperatures in the lower half of the rod bundle were in excellent agreement with the data. Above the core midplane, the code adequately predicted the initial cladding temperature rise and peak temperature, but under predicted the cool down and rod quench behavior. For the most part, the code tended to under predict vapor temperatures in the early part of the transient and over predict vapor temperatures in the latter part of the reflood transient. The under prediction of the cool-down and quench behavior of the upper half of the core, coupled with the under prediction of vapor temperatures in the early part of the transient and the over prediction of vapor temperatures in the latter part of the transient, indicates a weakness in the current reflood model that needs to be addressed. Overall, measured and calculated void fractions were generally in good agreement. The code predicted mass inventory and distribution were in excellent agreement during the first 70 s of the transient, but after 70 s, the calculated mass inventory was under predicted by about 10%.
FLECHT-SEASET Test 31701	Reflood model (high reflood rate), two-phase level, steam cooling, entrainment, CCFL,	FLECHT SEASET Test 31701 investigated the performance of the reflood model at a high flooding rate. Both semi- and nearly-implicit calculations were judged to be in reasonable agreement with the measured data. Predicted rod surface temperatures in the lower two-thirds of the rod bundle were in good agreement with the data. Above this, the

	condensation heat transfer, quench front	code adequately predicted the initial cladding temperature rise and peak temperature. However, calculated an earlier quench than was measured in the experiment. The earlier quench of the upper portion of the bundle was the result of a calculated top-down quenching of the upper regions of the bundle that was not observed in the experiment. For the most part, the code tended to over predict vapor temperatures in the latter part of the reflood transient. The difference between measured and predicted fluid temperatures may be due in part to thermocouples measuring a combination of superheated steam and saturated liquid as the quench front approaches the measurement location. The top-down and bottom-up calculated quench behavior resulted in increased steam generation in the bundle and a lower bundle mass inventory than was observed during the initial 90 s of the reflood experiment. However, when the entire core was calculated to quench at about 90 s, the calculated core mass inventory rapidly increased, filling the core with liquid by about 95 s into the transient.
Dukler air-water flooding	CCFL	The Dukler-Smith air-water flooding experiments assessed the Wallis CCFL model. Semi- and nearly-implicit predictions were in reasonable agreement with the Dukler-Smith experiment data for countercurrent air-water flow in a single tube over the range of air flows from 0.0126 to 0.126 kg/s. The assessment also shows that the Wallis correlation is implemented correctly.
UPTF downcomer countercurrent flow test 6, Run 131	Downcomer CCFL, lower plenum refill, condensation, noncondensables, two-phase level, thermal stratification, interphase drag, entrainment	UPTF downcomer CCFL Test 6, Run 131 was used to compare the relative performance of the annulus and pipe components for simulating the refill of the lower plenum during a loss-of-coolant accident. The two components are similar except that all the liquid is placed in the film, with no liquid allowed in drops, in the annulus component when in the annular-mist flow regime. Liquid is allowed in both the film and drops in the annular-mist flow regime in the pipe component. Both the semi- and nearly implicit calculations were judged to be in reasonable agreement with the measured liquid level data for UPTF Test 6, Run 131. The calculated refill was similar to that observed in the test but started about 6 s earlier. The RELAP5-3D calculation in which the downcomer was modeled with annulus components was in better agreement with the measured results than when pipe components were used. The flow regime model in the annulus component, which puts all the liquid in the film, resulted in a better prediction of the lower plenum refill for the UPTF test. The pipe component provided a conservative prediction of the amount of liquid in the lower plenum.
MIT pressurizer test ST4	Wall condensation, interfacial heat transfer, pressurizer level, thermal stratification	The MIT pressurizer test was used to assess the code capability to simulate pressurizer behavior under inflow and outflow conditions. The models tested during this simulation are steam condensation on the pressurizer wall and interfacial heat transfer between the stratified liquid and the vapor above the liquid. The MIT pressurizer test that is used for this assessment case is ST4. The code did a reasonable job of predicting the pressure response along with the axial temperature profile using both the semi-implicit and nearly-implicit advancement schemes, both with and without the mixture level tracking model, although use of that model improved the node boundary crossing response.
Neptunus test Y05	Pressurizer	Neptunus Test Y05 simulated pressurizer transient behavior involving surge line flows and pressurizer spray. The code-calculated pressurizer pressure and temperature results were judged to be in excellent agreement with the Neptunus pressurizer test data for test Y05 using either solution scheme. It was concluded that the pressurizer model appropriately calculated the expected two-phase, two-region, non-equilibrium behavior required to simulate pressurizer transient response.
MB2 test 1712	Steam generator behavior	MB2 Test 1712 was used to evaluate the performance of the code in modeling steady-state steam generator behavior. The overall results from the calculation were in reasonable agreement with the data. The primary temperature drops through the U-tube bundle showed reasonable agreement, indicating that the model correctly predicted the amount of energy transferred from the primary to the secondary system. Some other calculated steady-

		state conditions (e.g., narrow range level) lie outside the uncertainty range of the data, but not far enough to warrant concern.
LOFT experiment L3-1	Accumulator model	The LOFT L3-1 accumulator case assessed the performance of the accumulator model during a slow depressurization associated with a small break loss-of-coolant accident. The calculations with both solution schemes were judged to be in excellent agreement with the measured data.
Full scale reactor coolant pump	Two-phase pump behavior	The performance of the two-phase centrifugal pump model was evaluated using data from a full-scale reactor coolant pump. The calculations were judged to be in reasonable agreement with the measured data when pump head degradation multipliers that were specifically developed for that test condition were used. Improved results could be obtained by converting the head multipliers, which were developed from measured values in the pump suction, to estimated values in the pump. (Such an approach is needed to be consistent with the model incorporated in RELAP5-3D.) The calculations were judged to be in reasonable agreement with the measured data at intermediate pressures between two available head degradation multiplier curves. However, insufficient data were available to thoroughly test the validity of the interpolation. The semi- and nearly-implicit calculations produced identical results.
GE 1/6 scale jet pump	Jet pump	The performance of the jet pump model was evaluated using GE 1/6-scale jet pump data. Both the semi- and nearly-implicit calculations were judged to be in reasonable agreement with the measured data. For most of the measurement ranges, the code predictions were in excellent agreement with the data. However, for forward drive flows with flow ratios above 2.8, and for reverse drive flows with flow ratios below -2.5, the calculations were outside the measurement uncertainties, and the trends were away from the data. Most of the data taken were within the range that the code was performing well, suggesting that the regions where the code predictions were suspect may be encountered only in extreme cases. However, users should be aware of the potential problem.

Integral effect validation tests

Test case	Models validated	Description and evaluation
LOFT experiment L3-7	1in. small break LOCA	LOFT Experiment L3-7 simulated a small break LOCA in a PWR. For this test, the interest was in the overall system response, not that of the core, as there was no heat-up. Parameters of significance are the break flow rates, system pressure, emergency core coolant (ECC) system response, and system mass distribution. Most of the small break phenomena were simulated well by the code. The primary coolant system pressure response, pressurizer liquid level, and upper core cladding surface temperature calculations were reasonable. The coolant temperatures throughout the primary coolant system were evaluated as reasonable for most of the transient simulation. Simulations of the high-pressure injection system (HPIS) flow and the fluid velocity in the hot leg were judged to be in excellent agreement with the experiment data. The predictions of the break flow and the densities in the intact and broken loop cold legs were judged to be minimal, as were the lower core cladding surface temperatures and most of the coolant temperatures after 1200 s. The break flow was simulated well at the beginning of the transient, but was under predicted after about 600 s. Most of the problems in the calculation were attributed to the break flow. Improvement of the break flow rate prediction would likely result in acceptable calculations of the densities throughout the system. This in turn would likely improve the pressure and coolant temperature predictions after 1200 s. The assessment findings apply equally to both simulations; there were no significant differences between the calculations using the semi- or nearly-implicit solution schemes.
ROSA-IV test SB-CL-18	6in. small break LOCA	ROSA LSTF Test SB-CL-18 also simulated a small break LOCA in a PWR. Parameters of interest included the system pressure, break and injection flow rates, core liquid level, and peak core temperatures. The majority of the parameters calculated by the code were in reasonable agreement with the measured data, although there were several significant parameters that were not. The pressures in the primary and secondary coolant systems were in reasonable agreement with the data, as was the pressurizer liquid level. The mass flow rates in the hot and cold legs of both loops were judged to be reasonably predicted. The accumulator mass flow rates were in reasonable agreement with the experiment, although they were activated earlier in the calculation than in the experiment. The break mass flow rate was judged to be in minimal agreement with the data during the initial portion of the transient but was in reasonable agreement after the transition to primarily steam flow. The heater rod temperatures toward the bottom of the core were in good agreement with the data but became worse with increasing height in the core. The reactor vessel liquid level was under predicted for a portion of the calculation, which resulted in a longer uncovered core. The calculated densities in the intact loop hot and cold legs and in the broken loop hot leg were in minimal agreement with the measured data, although the broken loop cold leg density was in reasonable agreement. The primary deficiency in the calculation was that the code did not predict the clearing of the intact loop seal, which caused fluid to be retained in the intact loop. If this loop seal had cleared as it did in the experiment, the core liquid level would have been better predicted, and the core heat-up would have been shorter. Also, there would have been flow through the intact loop steam generator tubes, which would have improved the prediction of the steam generator pressure. Running the ROSA case with the H2ON and H2O95 water property files showed that the water property file had a negligible effect on the calculations.
Semi-scale NC tests 1, 2, 3, and 10	Loop natural circulation	Semi-scale natural circulation Tests S-NC-1, S-NC-2, S-NC-3, and S-NC-10 were used to assess the code capability for predicting the single- and two-phase natural circulation phenomena. Both solution scheme predictions were excellent for single-phase liquid natural circulation. For two-phase natural circulation, the code provided reasonable predictions of loop natural circulation behavior at high and intermediate core powers (100 kW and 60 kW), but a minimal prediction at low power (33.54 kW). A reasonable prediction of reflux condenser mode of natural circulation was provided when the core power was 60 kW and the primary system mass inventory was less than 67%. When the secondary system mass inventory was high enough to make the effective heat transfer area larger than around 50%, reasonable natural circulation was predicted. However, when it decreased to be smaller than 50%, the code predicted much higher mass flow rates in the primary system than were measured, which was in minimal agreement with experimental results.

LOBI test A1-04R	Large break LOCA	LOBI Test A1-04R was used to assess the code capability during the blowdown phase of a large break LOCA in a PWR. Both the semi- and nearly-explicit calculations were in excellent agreement for the system pressure and core differential pressure, and in reasonable agreement for the loop mass flow and fluid density and temperature at the accumulator injection location. The calculated results for the accumulator flow rate were significantly different from the LOBI data for both the nearly- and semi-implicit numerical schemes, although it is possible that the data are in error since the code does an excellent job in predicting the pressure response. The heater rod temperatures were judged to be in reasonable agreement for the lower core levels throughout the transient and for the upper levels through the initial heat-up and rewet, after which the upper core temperatures were in minimal agreement with the data.
LOFT experiment L2-5 (1-D)	Large break LOCA	LOFT Experiment L2-5 simulated a double-ended cold leg break in a PWR. The principal interest is in the core behavior: fuel rod heat-up and quench, peak cladding temperature. Other parameters of significance are the break flow rates, system pressure, and emergency core coolant (ECC) system response. For this case, the reactor vessel was modeled using one-dimensional components. Most of the significant parameters calculated by the code were in good agreement with the measured data. Four notable exceptions were the fluid densities in the intact and broken loop hot and cold legs. The pressures in both the primary and secondary coolant systems were found to be reasonably predicted, as was the pressurizer liquid level. The ECC behavior was well simulated, with the calculated accumulator level in excellent agreement with the data and the LPIS flow in reasonable agreement; no judgment on the HPIS flow was made since it was essentially a constant flow boundary condition. It can be inferred that the break flow was reasonably simulated, as the flow rates were in excellent or reasonable agreement with the data and the fluid densities were in reasonable agreement until after the core quenched. In the intact loop, the flow rates were reasonably simulated, but the calculated densities were in minimal agreement with the data, generally retaining more water in the loops than was measured; also, the pump speed was in minimal agreement for most of the transient, although it was in good agreement during the initial coast down. Fluid temperatures in the reactor vessel upper and lower plena were predicted reasonably, although the presence of accumulator nitrogen in the calculation reduced the saturation temperature in the lower plenum significantly. Both the fuel centerline and cladding surface temperatures were judged to be reasonably predicted. In the experiment, there was a partial top-down rewet of some of the fuel rods; this was not predicted by the code. The final quench of the core was both bottom-up and top-down in the experiment and in the calculation. The data showed some early heat-up in the center fuel assembly over the entire length of the core, but in the code calculation the top quarter of the core did not show any early cladding temperature excursions. The peak cladding temperature was predicted to be 72 to 88 K below the measured value, which is not unexpected as this is a comparison between a hot fuel rod and an assembly-average rod; the peak temperature also occurred earlier in the calculation than was measured (13 vs. 28 s). The assessment findings apply to both the semi- and nearly-implicit calculations, as there were no substantive differences between the two calculations.
LOFT experiment L2-5 (3-D)	Large break LOCA	LOFT Experiment L2-5 was also simulated using two multi-dimensional components in the reactor vessel, one for the downcomer and one for the region inside the core barrel. For this case, the nearly-implicit calculation failed part way through the transient and no assessment judgments were made. The results for the semi-implicit calculation were essentially the same as for the one-dimensional case, except for a better (excellent) prediction of the broken loop cold leg flow. The heat-up of the core extended higher than in the one-dimensional case, but the peak cladding temperature was nearly the same (11 K lower). The three-dimensional effects in the experiment were more pronounced than in the calculation. The essential behavior in the radial variations in the core temperatures was predicted by the code, but the azimuthal variations were not.

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