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RELAP-7 Software Verification and Validation Plan

Requirements Traceability Matrix (RTM) Part 2: Code Assessment Strategy, Procedure, and RTM Update

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DOE Office of Nuclear Energy

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ABSTRACT

This document addresses two subjects involved with the RELAP-7 Software Verification and Validation Plan (SVVP): (i) the principles and plan to assure the independence of RELAP-7 assessment through the code development process, and (ii) the work performed to establish the RELAP-7 assessment plan, *i.e.*, the assessment strategy, literature review, and identification of RELAP-7 requirements. Then, the Requirements Traceability Matrices (RTMs) proposed in previous document (INL-EXT-15-36684) are updated. These RTMs provide an efficient way to evaluate the RELAP-7 development status as well as the maturity of RELAP-7 assessment through the development process.

PREFACE

Document Version

This document is released as Revision 0.

It is the reader's responsibility to ensure he/she has the latest version of this document. Direct Questions may be directed to the owner of the document and project manager:

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CONTENTS

FIGURES

TABLES

ACRONYMS

ALWR	Advanced Light Water Reactor
B.C.	Boundary Condition
BCL	Battelle Columbus Laboratory
BWR	Boiling Water Reactor
CCTF	Cylindrical Core Test Facility
CFR	Code of Federal Regulation
CLB	Cold Leg Break
DOE	U. S. Department of Energy
ECCS	Emergency Core Cooling System
ESBWR	European Simplified BWR
FIST	Full Integral System Test
FP	Fission Product
HEM	Homogeneous Equilibrium two-phase flow Model
HPCS	High Pressure Core Spray
ICS	Integrated Control System
INL	Idaho National Laboratory
ISP	International Standard Problem
IV&V	Independent Verification and Validation
LOAF	Loss of All Feedwater
LOFW	Loss of Feedwater
LWR	Light Water Reactor
LWRS	Light Water Reactor Safety
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
NRC	Nuclear Regulatory Commission
NPP	Nuclear Power Plant
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Program
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System

RG	Regulatory Guide
RISMC	Risk Informed Safety Margin Characterization
RTM	Requirement Traceability Matrix
SCTF	Slab Core Test Facility
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SBO	Station Blackout
SGTR	Steam Generator Tube Rupture
SMR	Small Modular Reactor
SQA	Software Quality Assurance
SV&V	Software Verification and Validation
SVVP	Software Verification and Validation Plan
T/H	Thermal-Hydraulic
TLTA	Two Loop Test Apparatus
UPI	Upper Plenum Injection
UPTF	Upper Plenum Test Facility
U.S.	United States
UW	University of Wisconsin
V&V	Verification and Validation

RELATED DOCUMENTS

Item	Reference	Description

RELAP-7
Software Verification and Validation Plan
Requirements Traceability Matrix (RTM) Part 2: Code Assessment Strategy, Process, and RTM Update

1. INTRODUCTION AND OVERVIEW

RELAP-7 is a next-generation nuclear reactor system safety analysis code that has been developed by Idaho National Laboratory (INL) under the LWRS (Light Water Reactor Sustainability) program of DOE. In order to achieve the full potential of RELAP-7 beyond traditional analysis codes, the RELAP-7 has been developed by taking advantage of modern engineering and computational techniques while addressing the latest needs of nuclear industry [1]. As a result, the advances in knowledge/technology from various fields (*e.g.*, two-phase flow modeling, numerical methods, computing power, etc.) are incorporated into RELAP-7 [2], allowing us to expect an improved nuclear safety analysis using this code.

Since 2012, along with the effort for RELAP-7 development, INL has also launched a project RELAP-7 Software Verification and Validation Plan (SVVP). The primary goal is of this project to establish the RELAP-7 assessment plan that can be applied to evaluate the development and/or assessment activities through the RELAP-7 development process. According to INL's internal document PLN-4215 (2012), RELAP-7 SVVP aims to define all necessary actions required at each development phase so that people can determine if the specified requirements are completed properly through the RELAP-7 development process. This means that the various aspects of requirements for RELAP-7 (assessment) should be identified through this project.

Note that the term 'code assessment' is often used in literature as same meaning with 'code V&V' or just 'code validation'; or, the same term often indicates the different scope of work. In this document (*i.e.*, sections 2-5), the term 'code assessment' is consistently used to represent the general activity of evaluating the RELAP-7 code, and the scope can extend beyond the traditional V&V test problems. According to this definition, the code V&V activities are deemed subset of code assessment.

This document describes the INL's effort to establish the RELAP-7 assessment plan along with the discussion on the desired assessment process. Specifically, two different subjects are discussed:

(i) First, the general principles and plan to ensure the independence of RELAP-7 assessment activity from the other activities related to RELAP-7 are discussed. In particular, the desired relationship beween the groups working on the different duties (*e.g.*, code development vs. code assessment) are of interest when those activities are being made in parallel through the RELAP-7 development process.

(ii) Second, the specific work to establish the RELAP-7 assessment plan, from establishing the RELAP-7 assessment strategy to identifying the RELAP-7 requirements through literature review, is described. Also, the RELAP-7 assessment plan is proposed in the form of three RTMs. These RTMs provide an efficient way to evaluate the RELAP-7 development status as well as the maturity of RELAP-7 assessment through the development process.

The overall work addressed through this document is summarized with the section titles as follows:

Section No.	Section title
1.	Introduction and overview
2.	Approach for independent assessment of RELAP-7
3.	RELAP-7 code assessment strategy and procedure
4.	Knowledge base used for identifying RELAP-7 requirements
5.	Requirement Traceability Matrix (RTM) for RELAP-7
6.	Conclusions and future work
Appendix A.	RELAP-7 general RTM
Appendix B.	RELAP-7 specific RTM
Appendix C.	RELAP-7 code V&V RTM

In the following subsections, we reiterate the work scope and objectives of RELAP-7 SVVP described in previous documents [3, 4] while incorporating the recent updates due to the RELAP-7 activities since 2015. This document is a “living” document because it will be updated as new or revised information is achieved through the future activities of RELAP-7 development and assessment.

1.1 System Description

The RELAP-7 (Reactor Excursion and Leak Analysis Program) code is a nuclear reactor system safety analysis code being developed at Idaho National Laboratory (INL). The code is based on the INL’s modern scientific software development framework – MOOSE (Multi-Physics Object-Oriented Simulation Environment). 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 end result will be a reactor systems analysis capability that retains and improves upon RELAP5’s capability and extends the analysis capability for a variety of reactor system simulation scenarios.

1.2 Plan Objectives

The objective of this plan is to document the verification and validation activities for the software development process for RELAP-7. Additional information provided in this plan includes the Requirement Traceability Matrix which is the set of General Requirements, Specific Requirements and Code V&V Requirements.

For the INL, Software Quality Assurance (SQA) requirements are contract driven and interpreted from DOE Order 414.1D, “Quality Assurance”, 10 CFR 830 “Nuclear Safety Management”, Subpart A, “Quality Assurance Requirements”, and ASME NQA-1-2008 with the NQA-1a-2009 addenda, “Quality Assurance Requirements for Nuclear Facility Applications.” The INL internal document, PDD-13610 (Revision 13, 4/1/2015), "Software Quality Assurance Program" describes the SQA Program at the INL:

- PDD-13610 describes the Software Quality Assurance Program, which INL applies, to confirm that software used at INL is consistent with applicable requirements and is directed towards preventing software errors from occurring. The SQA Program includes a systematic set of standards, conventions, and methodologies implementing a standardized approach to the life cycle for all software at INL.

Per PDD-13610, INL SQA Program applies to all organizations, facilities, programs, projects, and subcontractors. The extent to which the program establishes required SQA activities is determined by the risk or quality level associated with failure of the software to meet established requirements. This graded approach consists of activities and sub-activities that can be implemented at varying levels of rigor based upon the potential impact on safety and the type of software. The more critical the software, the more formal and detailed the SQA activities must be performed and documented. Implementation of the SQA Program focuses on a life cycle management approach for Information Technology (IT) assets. This approach is described in LWP-13620, ″Managing Information Technology Assets.” The INL technology developed and/or used within RISMC pathway (*e.g.*, RELAP-7) also follows the LWP-13620.

PDD-13610 defines "Software" as *Computer programs and associated documentation and data pertaining to the operation of a computer system and includes application software and support software* [ASME NQA-1-2008 with the NQA-1a-2009 addenda edited]. Other softwares are defined as:

- Application software - A type of software designed to help users perform particular tasks or handle particular types of problems, as distinct from software tools (e.g., compilers) and system software (see def.) that controls the computer itself. Examples include timesheet, payroll, Electronic Document Management System (EDMS), Microsoft Excel spreadsheets, computer models, or process control applications. [ISO/IEC/IEEE Std. 24765-2010 edited]

- Support Software - Software that includes software tools (e.g., compilers) and system software. [ASME NQA-1-2008 with the NQA-1a-2009 addenda]

Note that within the INL SQA process, software that does not fall within the scope of the SQA Program includes any software covered by a contractual agreement, such as Work for Others, which includes references or requires a specific documented SQA process.

Applicable documents that apply to RELAP-7 development include:

- Software Quality Assurance Plan for RELAP-7, PLN-4212, 5/31/2012.
- Software Configuration Management Plant for the RELAP-7 Project, PLN-4214, 6/28/2012.
- Software Verification and Validation Plan for RELAP-7, PLN-4215, 6/28/2012.
- RELAP-7 Development Plan, INL/MIS-13-28183, 1/2013.

It is the responsibility of the **Software Owner** to make the determination as to whether a particular software can be classified as "Safety Software." Safety Software includes the following type of softwares:

- **Safety System Software**. Software for a nuclear facility that performs a safety function as part of a structure, system, or component *and* is cited in either (a) a DOE approved documented safety analysis or (b) an approved hazard analysis per DOE P 450.4, "Safety Management System Policy", dated 10-15-96, (or latest version) and 48 CFR 970-5223.1.

- **Safety Analysis and Design Software**. Software that is used to classify, design, or analyze nuclear facilities. This software is not part of a structure, system, or component (SSC) but helps to ensure that the proper accident or hazards analysis of nuclear facilities or an SSC that performs a safety function.

- **Safety Management and Administrative Controls Software**. Software that performs a hazard control function in support of nuclear facility or radiological safety management programs or technical safety requirements or other software that performs a control function necessary to provide adequate protection from nuclear facility or radiological hazards. This software supports eliminating, limiting or mitigating nuclear hazards to worker, the public, or the environment as addressed in 10 CFR Parts 830 and 835, the DEAR Integrated Safety Management System clause, and 48 CFR 970-5223.1. [DOE O 414.1D]

For all software that falls within the scope of the SQA Program, a **quality level** must be assigned by a qualified Quality Level Analyst with review and concurrence by a Quality Level Reviewer (i.e., a second Quality Level Analyst) per LWP-13014 (4/25/2013), "Determining Quality Levels." The Quality Level Analyst should then communicate to the Software Owner the determined quality level.

There is no consistent definition for the term Quality Level (QL). QLs only serve as a designator to identify the unmitigated risk or potential consequence level associated with the failure of an item or activity and to facilitate communication for a common understanding of the rigor to be applied through the appropriate implementation procedures:

Quality Level 1 High unmitigated risk or high potential consequence level of failure

Quality Level 2 Medium unmitigated risk or medium potential consequence level of failure

Quality Level 3 Low unmitigated risk

Quality Level 4 No risk item or service

The risk analysis used to designate QLs must be performed by personnel designated, trained, and qualified as QL Analysts. This initial training and every 3-year requalification of QL Analysts, also established by this procedure, is necessary to implement the graded approach effectively.

All documentation that furnishes evidence of the software quality is considered a QA record and should be handled as a quality record according to the organization, program, or project's “Records Management” as required by LWP-1202. QA records generated during the software development life cycle could include project plans, requirement specifications, configuration management plans, software quality assurance plans, security plans, and verification and validation documentation (e.g., test plans, test cases, and design review documents). Per LWP-1202, “Records Management,” the INL Records Schedule Matrix, and associated record types list(s) provide current information on the retention, quality assurance, and/or destruction moratorium requirements for these records. Contact a Records Coordinator for assistance if needed.

It is the responsibility of the contractor to ensure that these quality criteria are adequately addressed throughout the course of the research that is performed.

1.2.1 Software Quality Assurance

Software assurance is the planned and systematic set of activities that ensures that software processes and products conform to requirements, standards, and procedures. These processes are followed in order to enhance the robustness of the development process. Having formal documented development procedures and requirements helps to streamline the development cycle and focus on customer-driven needs.

In an attempt to improve the quality of the RELAP-7 tool set, effort has been made to establish criteria to which the development and control processes adhere. The recording of coding standards and the creation of the Requirements Traceability Matrix (RTM) will be added to improve code use and to establish traceability.

The roles and responsibilities of each team member are described below:

- Project Manager – Executes, maintains, and updates this plan. Monitors SV&V activities for the RELAP-7 Project. Coordinates formal user acceptance testing, when required. Performs as an alternate for technical team members.

- Software Developer – Performs design reviews, test case identification, design, construction, and functional unit testing during software development; reports anomalies and deviations to the Project Manager.

- Quality Assurance – Supports SV&V activities including RELAP-7 reviews. Is independent of the development and testing work

1.3 Supporting Activities

1.3.1 Development of MOOSE Application

RELAP-7 is a MOOSE (Multiphysics Object-Oriented Simulation Environment) based application which uses open source software packages, such as PETSC (a nonlinear solver developed at Argonne National Laboratory) and LibMesh (a Finite Element Analysis package developed at University of Texas). MOOSE provides numerical integration methods and mesh management for parallel computation. Therefore RELAP-7 code developers only need to focus upon the physics and user interface capability. By using the MOOSE development environment, RELAP-7 code is developed by following the same modern software design paradigms used for other MOOSE development efforts.

There are currently over 20 different MOOSE based applications ranging from 3-D transient neutron transport, detailed 3-D transient fuel performance analysis, to long-term material aging. Multi-physics and multiple dimensional analyses capabilities, such as radiation transport, can be obtained by coupling RELAP-7 and other MOOSE-based applications through MOOSE. This allows restricting the focus of RELAP-7 to systems analysis-type simulations.

The RISMC Toolkit is being built using the INL's MOOSE framework. MOOSE has been designed to solve multi-physics systems that involve multiple physical models or multiple simultaneous physical phenomena. Inside MOOSE, the Jacobian-Free Newton Krylov (JFNK) method is implemented as a parallel nonlinear solver that naturally supports effective coupling between physics equation systems (or Kernels). This capability allows for a tightly-coupled set of tools that work together, as shown in Figure 1.

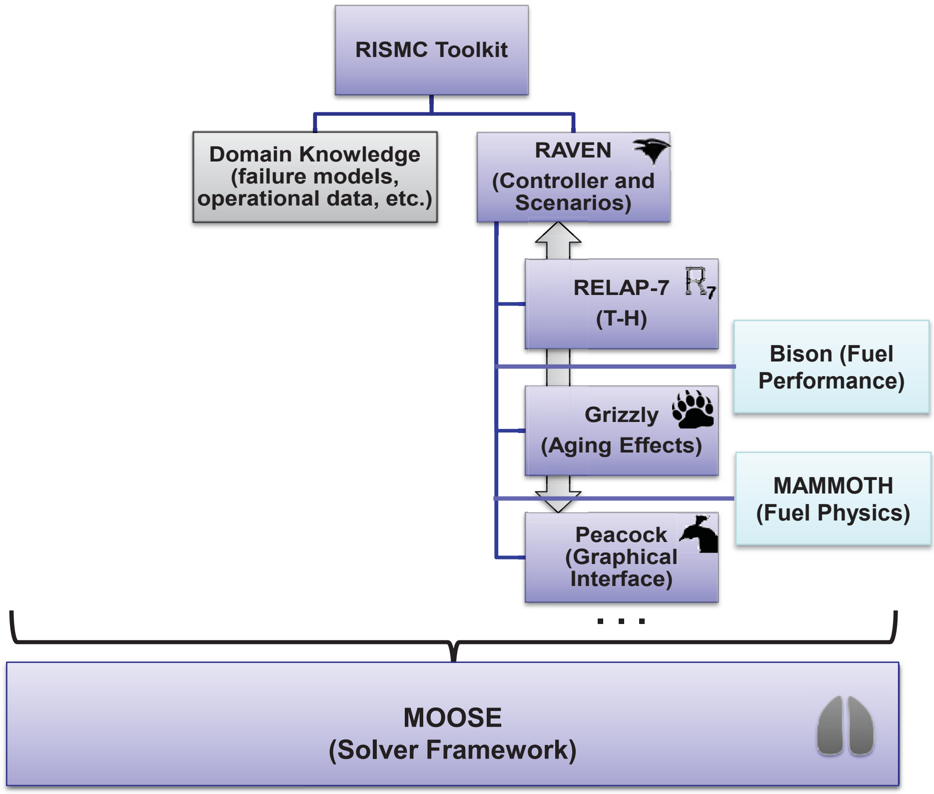

Figure 1. MOOSE-based applications

1.3.2 Technology Transfer

Development of RELAP-7 is to support US nuclear power industry and technical stewardship is envisaged. To realize this long-term vision, several items are considered.

The RELAP-7 quality assurance (QA) process includes the specific activities of verification, validation, assessment, and related documentation to facilitate reviews of these activities. To support these QA activities, a various results from facility operation, integral effects test, separate effect tests, and fundamental tests including experiments on individual components have been collected. The INL has started the QA process by implementing modern software management processes (including the use of tools such as source code version control) as a part of the RELAP-7 development, conducting NQA-1 audits, and creating a software verification and validation plan (SVVP).

The type of software license for RELAP-7 is still to be determined. RELAP-7 is subject to U.S. Export Control laws, including a complete embargo against any person from a T5 country (currently: Cuba, Iran, North Korea, Syria and Sudan). The software license for the supporting MOOSE framework is the open source license “Lesser GNU Public License (LGPL) version 2.1.”

1.4 RELAP-7 Features

In general RELAP-7 provides computational simulation of thermal-hydraulic behavior in a nuclear power plant and its components. Representative thermal hydraulic models are used to depict the major physical components and describe major physical processes. RELAP-7 has five main types of components/capabilities:

- Three-dimensional (3D) analysis coupled with other physics applications
- Two dimensional (2D)
- One-dimensional (1D) components (e.g., pipe)
- Zero-dimensional (0D) components for setting boundary conditions for the 1D components (e.g., Pressure boundary condition of pump)
- 0D components for connecting 1D components

RELAP-7 could be coupled to 3D core modeling MOOSE-based codes to enable detailed resolution.

The RELAP-7 code development started in 2012 based upon development input from the Electric Power Research Institute. During the first year of the code development, the software framework was created to establish the basic reactor system simulation capability with a number of components developed for single-phase thermal fluid flow. Later, two-phase flow modeling capability was implemented in the RELAP-7 code. These early capabilities have been demonstrated via application to a boiling water reactor simulation with representative components under extended Station Black Out (SBO) transient conditions.

The RELAP-7 α-0.1 was released in May 2012, and followed by α-0.2 version in August 2013 and α-0.6 version in September 2014. Since 2015, the code developers are using GitLab project which is the web base open community for code developers. Everytime the code has been updated the GitLab will automatically provide code version.

The RELAP-7 application is the next generation nuclear reactor system safety analysis code. The code is based upon the MOOSE (Multi-Physics Object-Oriented Simulation Environment). The goal of RELAP-7 development is to leverage of advancements in software design, numerical integration methods, and physical models.

The Homogeneous Equilibrium two-phase flow Model (HEM) has been archived.

Table 1 Component-related attributes for the RELAP-7 (as of 2016)

RELAP-7 Component	Dimensionality			Hydrodynamic Model		3D Linkage
	0D	1D	2D	Single Phase	Two Phase 7-Eq.	Application
Inlet	■	n/a	n/a	■	■	n/a
Outlet	■	n/a	n/a	■	■	n/a
SolidWall	■	n/a	n/a	■	■	n/a
Pipe	n/a	■	n/a	■	■	BISON
PipeWithHeatStructure	n/a	■	■	■	■	n/a
HeatStructure	n/a	n/a	■	n/a	n/a	n/a
HeatGeneration	n/a	n/a	■	n/a	n/a	n/a
CoreChannel	n/a	■	■	■	■	n/a
HeatExchanger	n/a	■	■	■	□	n/a
TimeDependentVolume	■	n/a	n/a	■	□	n/a
TimeDependentMassFlowRate	■	n/a	n/a	■	■	n/a
VolumeBranch	■	n/a	n/a	■	□	n/a
Branch	■	n/a	n/a	■	□	n/a
Valve	■	n/a	n/a	■	□	n/a
CompressibleValve	■	n/a	n/a	■	□	n/a
CheckValve	■	n/a	n/a	■	□	n/a
IdealPump	■	n/a	n/a	■	n/a	n/a
Pump	■	n/a	n/a	■	□	n/a
PointKinetics	■	n/a	n/a	n/a	n/a	n/a
SeparatorDryer	■	n/a	n/a	n/a	□	n/a
Downcomer	■	n/a	□	n/a	□	n/a
WetWell	■	□	n/a	■	□	n/a
Reactor	■	n/a	n/a	n/a	n/a	n/a
Turbine	■	n/a	n/a	■	□	n/a
Pressurizer	Δ	□	n/a	n/a	□	n/a
Accumulator	Δ	□	n/a	n/a	□	n/a
Steam Generator	n/a	□	n/a	n/a	□	n/a
Jet Pump	n/a	□	n/a	n/a	□	n/a

* ■: Available, Δ: Under review (developed, but not merged yet), □: Planned, n/a: Not applicable

In summary the RELAP-7 design is based upon:

- Modern Software Design:
    - Object-oriented C++ construction provided by the MOOSE framework
    - Designed to significantly reduce the expense and time of RELAP-7 development
    - Designed to be easily extended and maintain
    - Meets NQA-1 requirements
- Advanced Numerical Integration Methods:
    - Multi-scale time integration, PCICE (operator split), JFNK (implicit nonlinear Newton method), and a point implicit method (long duration transients)
    - New pipe network algorithm based upon Mortar FEM (Lagrange multipliers)
    - Ability to couple to multi-dimensional reactor simulators
- State-of-the-Art Physical Models:
    - All-speed, all-fluid (vapor-liquid, gas, liquid metal) flow
    - Well-posed 7-equation two-phase flow model
    - New reactor heat transfer model based upon fuels performance

Table 1 shows detailed features of RELAP-7.

1.4.1 Software Framework

The RELAP-7 (Reactor Excursion and Leak Analysis Program) code is based on INL developed framework software MOOSE (Multi-Physics Object Oriented Simulation Environment) which may model fully coupled nonlinear partial differential equations. The Graphical User Interface (GUI) of RELAP-7 can be provided by other MOOSE based softwares.

1.4.2 Governing Theory

Fundamentally, the RELAP-7 code is designed to simulate all-speed and all-fluid for both single and two-phase flow. However, current status RELAP-7 development focuses on simulation of the light water reactors (LWR), thus, two-phase flow model is described here.

The main governing theories of RELAP-7 are: 7-equation two-phase flow; reactor core heat transfer; and reactor kinetics models.

The 7-equation two-phase flow model consists of mass, momentum and energy (or pressure) equation for both liquid and vapor phases and a topological equation which explains the state of the two-phase mixture. This model may solve compressible fluid at all-speed multiphase flow which allows analyzing various transient phenomena and accident scenarios in LWR. In the RELAP-7, the 7-equation model is implemented in the MOOSE finite element framework.

Both convective and conduction heat transfer is simulated for fuel, fluid, and structures. The reactor core heat source is modeled by point kinetic method considering hydraulic reactivity feedback. The three-dimensional reactor kinetics may simulate through coupling with RattleSNake which is a reactor kinetics code with both diffusion and transport capabilities based on MOOSE framework.

1.4.3 Computational Approach

The RELAP-7 uses MOOSE-based applications with a multitude of mathematical and numerical libraries: LibMesh for the second-order accurate spatial discretization by employing linear basis, one-dimensional finite elements; Message Passing Interface (MPI) for distributed parallel processing; Intel Threading Building Blocks (Intel TBB) for parallel C++ programs to take full advantage of multi-core architecture found in most large-scale machines; and PETSc, Trilinos and Hypre for the mathematical libraries and nonlinear solver capabilities for Jacobian-Free Newton Krylov (JFNK).

To cover various time scale range of reactor transient and accident scenarios, the RELAP-7 pursues three-level time integration approaches: Pressure-Corrected Implicit Continuous-fluid Eulerian (PCICE) computational fluid dynamics (CFD) scheme for highly compressible and/or contain significant energy deposition, chemical reactions, or phase change problems; JFNK method for multi-physics problems during the transients; Point Implicit time Integration method for long duration and slow transient scenarios.

2. APPROACH FOR INDEPENDENT ASSESSMENT OF RELAP-7

One issue that must be addressed prior to any assessment activities (including V&V) for RELAP-7 is to identify the principles for operating the related INL organizations because the operation policy will substantially influence the whole process of RELAP-7 assessment. Specifically, the role, authority, and management policy of the RELAP-7 assessment organization should be clearly defined and declared before conducting any RELAP-7 assessment activities. In this context, several documents from industry, regulatory institute, and U.S. government are reviewed to obtain the useful guidance for managing/operating RELAP-7 code assessment organization and to support its functions properly. In particular, the main concern in this section is how to define and maintain “independence” between the code assessment team and the code development team throughout the RELAP-7 development process.

2.1 Industry Standards (IEEE Std. 1012-2004 [5])

IEEE Std. 1012-2004 [5] provides a V&V process standard that applies to all life cycle processes of software (*i.e.*, acquisition, supply, development, operation, maintenance). Particularly, to assure the independence of software V&V process this document proposes to establish *technical*, *managerial*, and *financial independence* as follows:

(1) For *technical independence*, the independent V&V (IV&V) effort should be made by personnel who are not involved in the software development. In principle, any problems should be understood, formulated, and solved separately through V&V effort, which will help detect subtle errors that can be overlooked by those too close to solutions such as code developers. *Technical independence* also requires that V&V effort be made with its own set of tools for test/analysis separate from the developer’s tools. Sharing tools is only limitedly allowed for computer support environments (e.g., compilers, assemblers, utilities) or for system simulations where an independent version would be too costly.

(2) For *managerial independence*, the V&V responsibility should belong to an organization separate from the code development and program management organizations. *Managerial independence* also requires that V&V effort independently select the segments of the software and system to test/analyze, V&V techniques, schedule of V&V activities, and the specific technical issues/problems to work on. The results of IV&V effort should be reported in a timely fashion to both the development and program management organizations without any restrictions or adverse pressures.

(3) For *financial independence*, the V&V budget should be vested in an organization separate from the software development organization. This is to prevent any situation where the V&V organization cannot complete its mission because of diverted funds or adverse financial pressures/influences.

Also, in IEEE Std. 1012-2004 [5] five forms of independence (*i.e.*, classical, modified, integrated, internal, and embedded) are defined for the three independence parameters mentioned above (see Table 2), which is to determine the degree of independence achieved through V&V process. For more details of these forms, readers are advised to refer to [5].

Table 2. Forms of IV&V described in IEEE Std. 1012-2004 [5]

IV&V Form	Technical	Management	Financial
Classical	I	I	I
Modified	I	i	I
Integrated	i	I	I
Internal	i	i	i
Embedded	e	e	e
Note: I=Rigorous independence; i=Conditional independence; e=Minimal independence			

It is noted that the V&V process standard described in IEEE Std. 1012-2004 [5] is industry consensus and is largely endorsed by the documents for the software V&V released by NRC and DOE.

2.2 NRC Regulatory Guides [6, 7]

The NRC regulatory guides (RG), 'NUREG/BR-0167 [6]' and 'RG 1.168 [7]', provide guidance for the software V&V for use in NRC or for regulation purposes. Specifically, NUREG/BR-0167 [6] offers a guidance to NRC organizations and NRC contractors involved in developing and maintaining software for use by the NRC staff, while RG 1.168 [7] provides a method of software V&V, reviews, and audits in compliance with NRC regulations. It is noted that these RGs largely accept various industry standards specified in IEEE Std. 1012-2004 [5] and IEEE Std. 1028-2008 [8].

As for the independence of software V&V (or QA) used in the safety analysis of nuclear power plants, the above documents describe the followings:

NUREG/BR-0167 [6] describes IV&V as an activity conducted by an organization that is both technically and managerially separate from the software development organization. Also, the sponsors and users of *Level 1 software*[1] should decide together if the fund for a project of IV&V is warranted.

Similarly, according to RG 1.168 [7] any organization with reviewers performing QA functions is required to use an independent organizational structure in terms of technical, financial, and managerial aspects, which follows the requirements specified in Criteria I and III in Appendix B to 10 CFR Part 50 as well as IEEE Std. 1012-2004 [5].

[1] NUREG/BR-0167 defines three levels of software used by the NRC. Level 1 software represents technical application software used in a safety decision by the NRC (e.g., RELAP5) and Level 2 software is technical or non-technical software not used in safety decision. The guidelines in NUREG/BR-0167 apply to Level 1 and Level 2 software only.

2.3 Federal Standards and DOE Guide

The 10 CFR 830 Subpart A describes the requirements of quality assurance program (QAP) for DOE nuclear facilities and activities, which is supplemented by DOE O 414.1C [9]. DOE G 414.1-4 [10] is also a software guide for use with both 10 CFR 830 Subpart A and DOE O 414.C, providing instructional guidance to be applied with the requirements specified in DOE O 414.C.

As for the independence of software V&V or QAP, 10 CFR 830 Subpart A and DOE G 414.1-4 include the following descriptions:

- 10 CFR 830 Subpart A, Criterion 10

  "Establish sufficient authority, and freedom from line management, for the group performing independent assessments."

- DOE G 414.1-4

  "SQA (software quality assurance) Evaluator - an independent reviewer of the computer software, who is not affiliated with the software developing organization."

  "Independence between the evaluator and the sponsor is critical for completion of a formal SQA evaluation, and should be maintained throughout the Central Registry submittal process."

NQA-1 [11], a guidance for implementing federal regulations associated with quality assurance (QA) described in 10 CFR 50 Appendix B, also share the fundamental principles for the role and responsibility of QA organization as follows:

"First and foremost, the QA team must be able to function independently from the organizations it is responsible for overseeing. This includes having the authority to stop work or bring an issue independently "up the chain" to the site manager or other top executive as defined by the facility's governance model."

2.4 Roles and Responsibilities of RELAP-7 Teams for Independent Assessment of RELAP-7

The general principles described in the documents above will be followed as closely as possible by INL to assure the independence of RELAP-7 assessment in relation to the other organizations like RELAP-7 development and project management teams. The roles and responsibility of each team will be specified such that the technical, managerial, and financial independence of RELAP-7 code assessment can be assured. To this end, the organizational structure supporting the independent activity of RELAP-7 assessment should also be determined (*e.g.*, see Annex F in IEEE Std. 1012-2004 [5]), which will subsequently be announced to the members in charge of any activities related to RELAP-7.

The general roles and responsibilities assigned to project manager, software developers, and quality assurance team (or code assessment team) were described in section 1.2.1, but this needs to be further specified from the views discussed through this section.

3. RELAP-7 CODE ASSESSMENT STRATEGY AND PROCEDURE

This section describes a basic strategy and approach taken for establishing RELAP-7 code assessment plan. In addition, the overall assessment procedure of RELAP-7 is discussed in order to clarify the work scope of this study in that context, *i.e.*, what we currently need.

The RELAP-7 code assessment plan has largely been established in a similar vein with that of RELAP-7 development, *i.e.*, taking full advantage of accumulated knowledge/experience in creating an advanced form of engineering tool/system. Figure 2 shows the basic principle/strategy that applies to the RELAP-7 code assessment plan. Similar to that of RELAP-7 code development, RELAP-7 assessment plan accounts for the improved knowledge in code qualification methods [12] and experience-based findings/demands from nuclear industry [1]. Also, there have been comprehensive efforts internationally to identify the general needs for next-generation nuclear system analysis code, necessary V&V efforts, and relevant safety issues [13-15]; these are also reviewed in this study to make the RELAP-7 assessment plan be in line with those efforts. In addition, the V&V activities made during the development phase of recently-developed reactor system analysis codes (*e.g.*, RELAP5-3D [16], TRACE v5.0 [17-20]) are investigated to identify the V&V test problems for RELAP-7. Lastly, the extensive validation data that are judged to be available and useful, but not used for validation purpose of existing codes, as well as the newly identified safety issues are also surveyed. Combining all these efforts, the requirement items for RELAP-7 and the outlines of RTMs are determined (sections 4 and 5), which will be eventually used to evaluate the maturity of RELAP-7 code development and assessment processes at a given point in time.

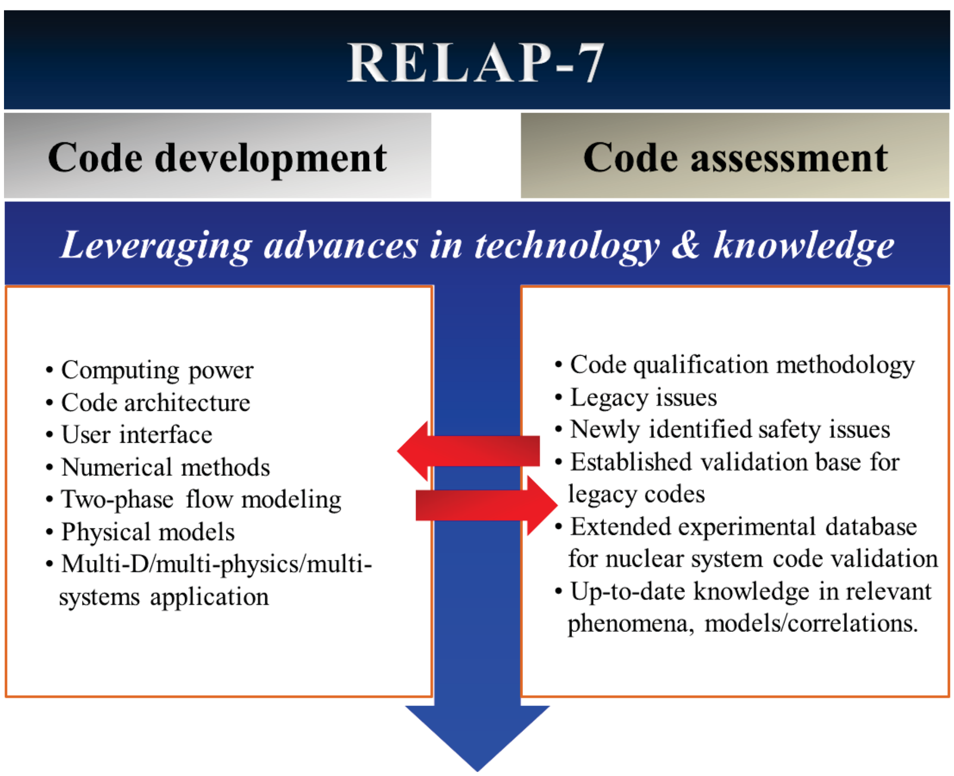

Figure 2. RELAP-7 code development and assessment strategy

Figure 3 shows the RELAP-7 code assessment procedure from its development to release. The RELAP-7 target capability was initially defined (*e.g.*, enhanced and integrated T/H analysis) based on the specific industry needs and requirements delivered by EPRI [1, 21]. In particular, the reports emphasize the necessity of an advanced nuclear system analysis code that is capable of addressing the design and regulatory issues over the “full (or extended)” plant time operation such as NPP life extension and power level uprates. Meanwhile, it should be noted that a successful code assessment program is key to achieving such code capability as well as demonstrating the overall quality of RELAP-7.

In general, the code assessment requires two stages as discussed by Petruzzi and D’auria [12]: (i) internal code assessment and (ii) external (or independent) code assessment. The internal code assessment is a process that should be conducted during a code development phase by code development team. The main activities in this stage include the (1) general SQA procedures, (2) code verification to check the correctness in models, interfaces, and numerical algorithms, etc., and (3) code validation to evaluate the code prediction accuracy as well as the consistency of the results by comparison with relevant experimental data. On the other hand, the external code assessment should be performed by independent code users after completing the internal code assessment, normally after the code beta version is released. In this stage, the transient simulation results of the code are further qualified against experimental data obtained from ITF (Integral Test Facility); and of course the databases should be independent from those used in the code development process. The nodalization strategy [22, 23], code application procedure [24], user qualification [23, 24], and evaluation of code prediction accuracy [24, 25] should be clearly addressed at this stage of code assessment. Also, the code capability must be demonstrated at the full scale of NPP through this stage [12, 26].

The code assessment procedures, both internal and external code assessment, should also be conducted for RELAP-7. At present, however, RELAP-7 is still under development and thus we focus on identifying the requirements needed for “internal” code assessment of RELAP-7 as illustrated in Figure 3.

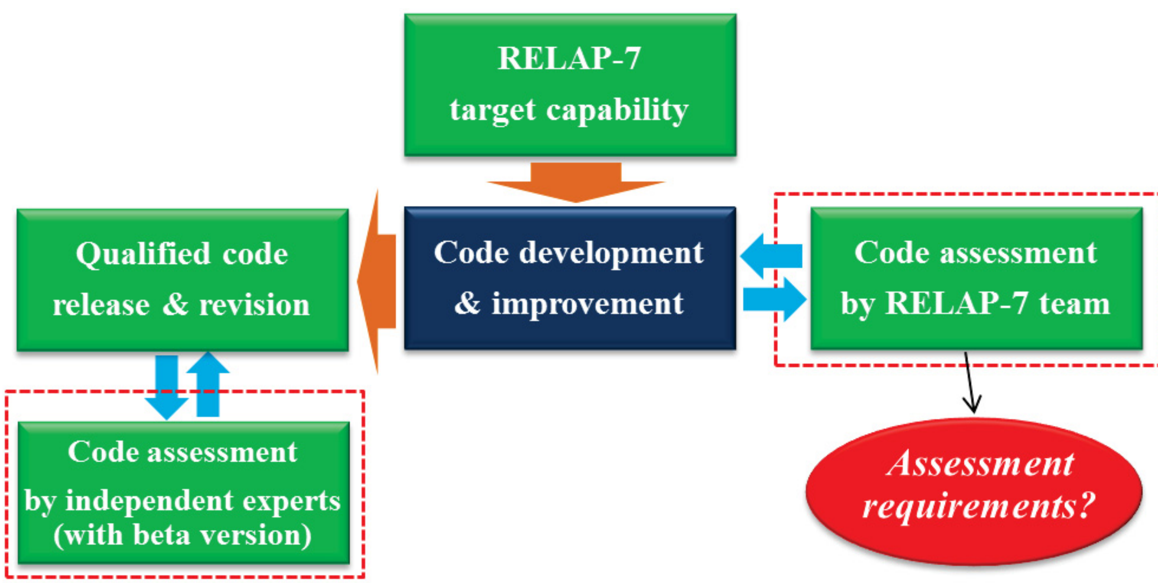

Figure 3. RELAP-7 code development and assessment procedure

4. KNOWLEDGE BASE USED FOR IDENTIFYING RELAP-7 REQUIREMENTS

The RELAP-7 code assessment strategy described in section 3 strongly implies that extensive literature review is required to identify the RELAP-7 requirements for the "internal" code assessment. To make the literature review process efficient, we have collected all the necessary information with categorizations as follows: (i) general needs for next-generation code, (ii) legacy issues of existing codes, (iii) V&V base of legacy codes, and (iv) validation database recently added or suggested by industry. In general, the information sources include international reports, research articles, legacy code assessment manuals (or validation matrices), U.S. regulatory rules/guidelines (*e.g.*, NUREG-0800, 10 CFR50, RG 1.70, etc.), PIRT reports, and recent reports on industry needs [1, 21]. In the following subsections, the knowledge bases mainly used to identify the requirements of RELAP-7, which are subsequently used as input to RELAP-7 RTMs (section 5), are discussed.

4.1 Desired Characteristics of Next-Generation Nuclear System Safety Analysis Code

This section discusses various code features that a next-generation nuclear system analysis code (*e.g.*, RELAP-7) is desired to have. Within nuclear system code community, the general demand for next-generation code, legacy issues of existing codes, improvement items, and industry needs, etc. have been discussed for decades based on the lessons learned from the experience of code developers and users. Also, such information has been documented in several reports and/or research articles [1, 13, 27-29]. In Table 3, the diverse aspects of code features discussed in those documents are summarized and categorized. Among them, some major demands of high-priority are described in detail as follows:

- One of the most highly-ranked demands from the system code user group is the code robustness. Consequently, fully implicit time integration scheme and high-order accurate spatial differencing schemes are recommended to enhance the code stability as well as the modeling accuracy. It is noted that, for two-phase problems, considering pressure difference between different phases instead of pressure equilibrium assumption (*i.e.*, $P_{liquid}=P_{vapor}$) can also be one way to improve the numerical stability of two-fluid model [30, 31].

- High level of modularity is desired for the new code and recommended to be achieved through object-oriented programming. This feature will allow the code maintenance and revisions more easily for code developers as well as provide more convenience for code users who want to add and/or test new models.

- Modeling through simplification that overtly sacrifices simulation fidelity, which were introduced previously due to the limitation of computing power and numerical methods, should be avoided. For instance, two-phase flow simulation with less than two-fluid six-equation models is barely recommended in system analysis with little exception.

- Code architecture is recommended to facilitate both multi-dimensional (multi-D) and multi-physics analyses for "realistic" reactor simulation, which will necessarily allow us to mitigate conservatism utilized in previous modeling approach. In addition, such code feature will enable the code to readily address the increasing safety concerns associated with plant life extension and/or power uprates (e.g., chemical effects like corrosion on fuel cladding and steam generator tubes, pellet-cladding-interaction (PCI), in-reactor anomalies caused by asymmetric flow behavior within core, fission gas release, etc.).

- Standard modules for code input with limited options are recommended to be provided in order to minimize the code user effects (e.g., to avoid arbitrary tuning by the code users). In the similar context, standard and recommended options for all aspects of using the code should be identified and documented in appropriate manner.

- Improvements in code execution time via both faster solution methods and computing power (e.g., parallel computing) are required.

- Needs for improvement in uncertainty evaluation method in system code are commonly pointed out in references [1, 13, 27]. In particular, the code capability of incorporating uncertainty quantification process into an integral part of simulation process has recently been paid increasing attention [29, 32].

Table 3. General demands for next-generation nuclear reactor system analysis code [1, 13, 27]

	Improvement items
Code architecture	- Object-oriented programming - Parallel computation (e.g., multi-threading) - Easy coupling with other codes including proprietary codes (for multi-scale/multi-physics simulation)
Mathematical formulation of governing equation	- Incorporation of interfacial pressure (i.e., pressure non-equilibrium)
Physical modeling	- Modeling of dynamic flow regime (e.g., interfacial area transport equation) - Multi-field model (e.g., droplet, film fields) - Modeling capability for sources and particle transport in vapor, gas, droplet, and liquid phases (e.g., boron concentration tracking) - Closure models for multi-D applications - Closure model improvement at low pressure/low flow conditions - Transport of non-condensable gases and their effect on heat transfer - Coordinate systems to represent the actual design of component or system - Turbulent diffusion models
Numerics	- Low diffusive schemes that can resolve sharp gradients - Availability of different numerical schemes that can be applied depending on the problem time-scales - Multi-D discretization - Fully implicit (for enhanced stability)
User needs	- Improved robustness - Documentation (e.g., theory, programming, user manual, validation bases, user guidelines) - GUI (for pre-/post-processing, online monitoring, input deck generation) - Notification function on the validity range of code models (e.g., warning signal if the validity range is exceeded for a given problem) - Standard modules to minimize user effect (e.g., standard modules with limited options as opposed to user-defined meshing) - Near-real-time code performance - Automatic evaluation of time step sensitivity - Platforms/compilers independence and easy installation - High level of modularity (e.g., easy replacement/addition of models) - Unified interface protocol to facilitate coupling - Standard or recommended options for all aspects of using the code - User-friendly steady-state initialization and restart capabilities - Clear and understandable diagnostic feature for debugging - Improved uncertainty quantification method (e.g., internal assessment of uncertainty)

4.2 Existing V&V Effort for Nuclear System Safety Analysis Codes

The V&V of best-estimate nuclear system analysis codes has always been among the most important subject in the field of nuclear system safety analysis, and thus there have been comprehensive efforts to support the activities. Of these, the most well-known are International Standard Problems (ISP) [33] and CSNI Code Validation Matrices (CCVM) [15, 34, 35], both of which have been led by OECD/NEA CSNI (Committee for Safety of Nuclear Installation). Both ISP and CCVM are the consequences of efforts to systematically collect the best sets of qualified experimental data for code validation as well as code assessment with respect to uncertainty quantification. Recently, similar effort has continued by D'auria's research group to consolidate qualified database (both experimental and code calculation results) through standardization, aiming to support the V&V activities of system codes and uncertainty methodologies [36]. In addition, new experimental data and thus new insight/safety issues have been continuously revealed [37, 38] all of which are obviously precious to demonstrate and improve the nuclear system code capability. Note that many of these efforts are mainly to support the "independent" code assessment after completing the code development phase (section 3).

As discussed in section 3, of our current interest is to identify the requirements and subsequently to establish a plan for the "internal" code assessment of RELAP-7. In this regard, we first review the validation bases used for the "internal" code assessment of recently-developed code in the U.S., *i.e.*, RELAP5-3D and TRACE. It is noted that the validation base of these codes can be utilized as a useful source for RELAP-7 assessment (*e.g.*, code-to-code comparisons) because the modeling process and the simulation results are easily accessible and well-documented [16, 18-20].

CSNI has identified a total of 67 two-phase flow phenomena relevant to LWR safety on the basis of study for the 187 identified facilities [39]. These cover the representative phenomena occurring during LOCA/non-LOCA sequences in nuclear reactor systems (LWR) and many of them are considered essential to validate the nuclear system analysis codes. Thus, comparing the list with the validation base used for RELAP5-3D and TRACE can also be one of the simple ways to gain an insight into how well the code capability was tested and/or demonstrated through the development process. Obviously, this will also help identify further needs to be supplemented for RELAP-7 code validation. The present study reveals that several basic two-phase flow phenomena and system-based phenomena are not explicitly addressed by the validation test problems described in the code manuals of RELAP5-3D and TRACE, the list of which is given in Table 4.

Table 4. Relevant two-phase flow phenomena to LWR safety, which are not explicitly covered by the validation base of either RELAP5-3D or TRACE

Classification	List of phenomena
Basic Phenomena	- Phase separation (vertical pipe) - Pressure wave propagation (*e.g.*, water hammer)
System-based Phenomena	- Phase separation at branches (*i.e.*, T-junction) and its effect on leak flow - Loop seal filling/clearance - Boron mixing and transport - Spray effects - Separator, steam dryer behavior - Condensation in stratified conditions in pressurizer/steam generator/horizontal pipes

Aside from the code V&V efforts discussed above, EDF and CEA also proposed benchmark tests that can be used to assess the numerical behavior of nuclear T/H system code under development [40]. The 27 benchmark tests proposed encompass various physical problems of two-phase flow relevant to nuclear safety. Also, the qualification process and V&V database of CATHARE code [41] is reviewed, revealing that the V&V database is quite different from that used by the U.S. domestic codes (*e.g.*, RELAP5-3D, TRACE) while accessibility to those data is uncertain.

4.3 EPRI-Recommended Code Assessment Base [21]

As a part of collaboration with INL on the U.S DOE LWRS program, the Electric Power Research Institute (EPRI) produced a report [21] assembling the code assessment database that can be used for the assessment of next generation safety analysis codes like RELAP-7. Specifically, this report provides relevant information on (i) experimental facilities and plant-scale tests/events that can provide relevant code assessment data, (ii) assessment activities of U.S. domestic safety analysis codes with EPRI's evaluation on them, and (iii) EPRI-recommended code assessment base.

The work scope is limited to the T/H transients and accidents in LWRs (see Chapter 15 of NUREG-800), especially for the reactor designs of current operating fleet of PWRs and BWRs in the U.S with some differences in design details (*e.g.*, steam generators, loop configurations, and auxiliary systems, etc.). Meanwhile, the code assessment activities or test data for demonstrating passive safety features of advanced reactor designs (*e.g.*, AP-1000, ESBWR, SMR) are not included to the work scope.

In order to attain the research objectives, mainly both compiling the assessment base of legacy codes and providing data sources available for validation of current or next generation codes, extensive literature review is performed by EPRI. Largely, the literature search covers legacy code assessment manuals (U.S. nuclear system safety analysis codes), industry summary reports (*e.g.*, NUREG-1230, CSNI Code Validation Matrix), PIRT reports for LWR designs (LOCA/non-LOCA events), and published code assessment reports (*e.g.*, NUREG/CR, NUREG/IA).

5. REQUREMENT TRACEABILITY MATRIX (RTM) FOR RELAP-7

Combining all the efforts described in sections 3 and 4, a broad spectrum of information, from the general requirements as a next-generation code to the specific test problems for the code V&V, is integrated with RELAP-7 RTMs. The RELAP-7 RTMs provide the wide range of tangible and traceable items (*i.e.*, requirements) that can be used for RELAP-7 assessment through the RELAP-7 development process. It is noted that the RELAP-7 RTMs presented in this section have been updated from the previous version [3] mainly through more thorough literature review and restructuring of the matrix for effective information tracking. The following sub-sections will describe in detail the current characterization of RELAP-7 requirements and the RELAP-7 RTMs established based on them.

5.1 Characterization of RELAP-7 Requirements

The various requirements for RELAP-7 identified through this study are categorized into three groups according to the characteristic of requirement items: (i) general requirements, (ii) specific requirements, and (iii) code V&V requirements (see Figure 4). Also, each requirement group is further divided into several sub-groups of different aspect as shown in Figure 4.

"General requirements" prescribe the software capabilities required for RELAP-7 as a next-generation nuclear system safety analysis code. The code capability of addressing various reactor and/or system designs (*e.g.*, PWR, BWR), multi-dimensional T/H analysis, T/H system analysis over a wide range of transients/accidents in LWRs, multi-scale/multi-physics analysis through the coupling with other codes, and an improved function of uncertainty analysis is subject to the assessment in this group.

"Specific requirements" identify the technical aspect of requirements for RELAP-7 especially by focusing on the legacy issues of existing nuclear system analysis codes. The various viewpoints of legacy issues, *i.e.*, computer science technology, software architecture, modeling accuracy/reliability, advanced modeling of physical phenomena, and the requirements for software quality assurance can be assessed in this group.

"Code V&V requirements" provide the detailed list of verification and validation test problems that needs to be performed through the RELAP-7 development process. The validation base of legacy codes [16-20, 40-42] and the extensive survey on the validation data conducted by EPRI [21] are utilized as major inputs to building up the requirement list in this group. Also, several test problems for the verification of RELAP-7 are incorporated into the current *code V&V RTM* (section 5.3) based on the interaction with RELAP-7 development team, but further update is still needed for this.

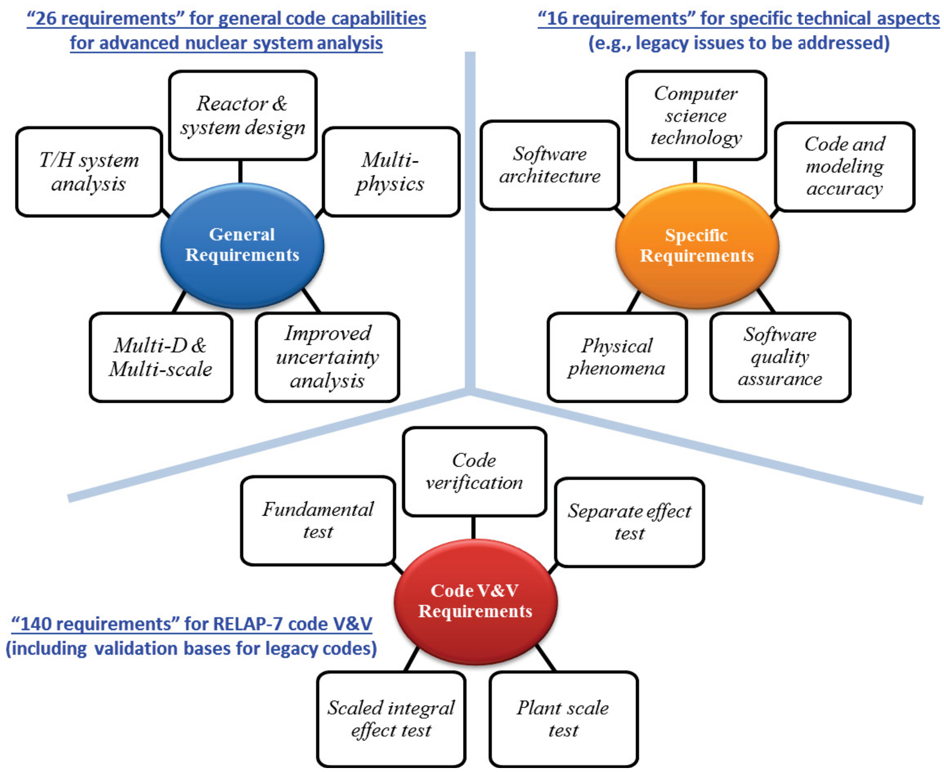

Figure 4. Current characterization of requirements for RELAP-7

5.2 RELAP-7 RTM Structure

Based on the characterization of RELAP-7 requirements described in section 5.1, three RTMs are suggested: (i) *general RTM*, (ii) *specific RTM*, and (iii) *code V&V RTM*. These RTMs will serve to evaluate the maturity of the RELAP-7 assessment process as well as the predictive capability at a given point in time; and the three RTMs allow such evaluation from various perspectives through the RELAP-7 development process. The current RTMs have been updated from the previous version [3] through more thorough literature review and restructuring of the matrix. In particular, the *code V&V RTM* (previously *technical RTM*) is reorganized to effectively track the V&V test problems according to the test type, data characteristic, and data availability, the detailed structure of which is discussed below in this section. Also, the current RTMs are configured such that each requirement defined in *general RTM* can be assessed in connection with the requirements included in *code V&V RTM*. That is, each requirement in general RTM should be sufficiently supported by the V&V test problems in *code V&V RTM* in order to achieve the adequacy for a given application, as shown in Figure 5 (see the column for 'Code V&V RTM No.'). The 'Modification Date' column shown in Figure 5 is to track the process of RTM update with time. The complete form of each RTM created through this study is provided in Appendices A-C.

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
⋮					

Figure 5. An example of *general RTM* structure

The *code V&V RTM* includes the extensive range of V&V test problems required for RELAP-7 (internal) assessment through the development process. The information on the validation data in the current *code V&V RTM* has been collected following the process shown in Figure 6: first, the transients/accidents in LWRs that should be taken care of from the design and licensing point of view are determined based on the literature review (*e.g.*, NUREG-0800 , RG-1.70, 10 CFR 50, industry report [21]). Then, the related experimental campaign and specific tests are reviewed to identify the data characteristics and data availability [14-16, 18-21]. Subsequently, the information is mapped into the *code V&V RTM* which provides the list of required V&V test problems, references, data availability, and RELAP-7 test status, etc. with several categorizations.

The specific categorizations and the overall structure applied to the current *code V&V RTM* is shown in Figure 7. First, the requirements (or test problems) included in *code V&V RTM* can be categorized according to the test type, *i.e.*, verification ('ver') or validation ('val'). The verification test problems falling into the 'ver' category are also segregated into two sub-categories, *i.e.*, code verification and solution verification (see Figure 7 and Table 5). According to Oberkampf and Trucano [43], the code verification indicates any activities to detect programming errors and/or coding mistakes without specific knowledge of numerical accuracy. Numerical solutions are usually compared with highly accurate solutions (*e.g.*, analytic solution) for the purpose of code verification. On the other hand, the solution verification focuses on estimating the numerical accuracy of a given (numerical) solution relative to a physics equation. This so-called high-level verification is conducted in general to identify the adequacy of a numerical solution for a given problem. In the current *code V&V RTM*, such different aspects of verification test problems are included as suggested by Oberkampf et al [43, 44], more details of which are given in Table 5. However, it is noted that the current *code V&V RTM* barely deals with the code verification associated with software configuration management (*i.e.*, software quality engineering). Also, the verification test problems of each aspect should be further detailed through future updates.

Table 5. Verification test aspects implemented into the *code V&V RTM* [43]

Verification Test Aspects Considered in RELAP-7	
Code Verification	Solution Verification
* Numerical Algorithm Verification - Spatial/temporal convergence rate tests - Iterative convergence tests - Solution independence tests to coordinate transformation - Conservation tests - Symmetry tests with various type of B.C.s * Software Qualify Engineering (SQE) - Consistency of simulation results on a specified computer hardware and in a specified software environment	- Discretization errors (to test the spatial/temporal discretization errors implemented into the code) - Iterative solution errors (to test the performance of non-linear solver implemented into the code)

The validation test problems ('val') in *code V&V RTM* can be categorized into 5 sub-groups (*i.e.*, FT, SET, IET, CT, PT) depending on the scale of experiment, test purpose, and phenomenological complexity observed during the experiments (see Appendix C). In the *code V&V RTM*, FT denotes the fundamental validation tests with a relatively simple geometry, performed to investigate the generic T/H phenomena relevant to the LWR safety. The relevant T/H phenomena are determined based on the study conducted by OECD/CSNI [14, 15, 45] and EPRI [21]. Also, the proper set of validation tests in this category (FT) are selected by considering both the availability of legacy code simulation results (for code-to-code comparison) and the data availability. The SET and IET represents the separate effect test and scaled integral effect test, respectively. CT denotes the validation test for the specific components used in LWRs (*e.g.*, jet pump, pressurizer, accumulator, etc.). PT indicates the validation data obtained from the actual full-scale nuclear power plants (NPPs). It is noted that, only publicly available data and code assessment activities are considered in the current *code V&V RTM* shown in Appendix C.

Another categorization applied to the *code V&V RTM* is based on the specific purpose of individual tests under each experimental program, *i.e.*, specific transients/accidents simulated to occur in LWRs (*e.g.*, LBLOCA, SBLOCA, natural circulation, etc.). In addition, the validation tests targeting a specific reactor design (*e.g.*, PWR or BWR) are also informed in the *code V&V RTM*.

The current structure of RELAP-7 RTM discussed above, especially the *code V&V RTM* provides us with several benefits as follows: (i) first, useful information required for the RELAP-7 assessment, from the verification test problems to the wide scope of validation sources (*e.g.*, available data, data characteristic, legacy code results, references, etc.), can be tracked efficiently, (ii) second, the maturity of RELAP-7 assessment process can be systematically evaluated and visualized, (iii) third, the current RTMs can help identify the data gap and/or technology gap that is worth challenging in the near future.

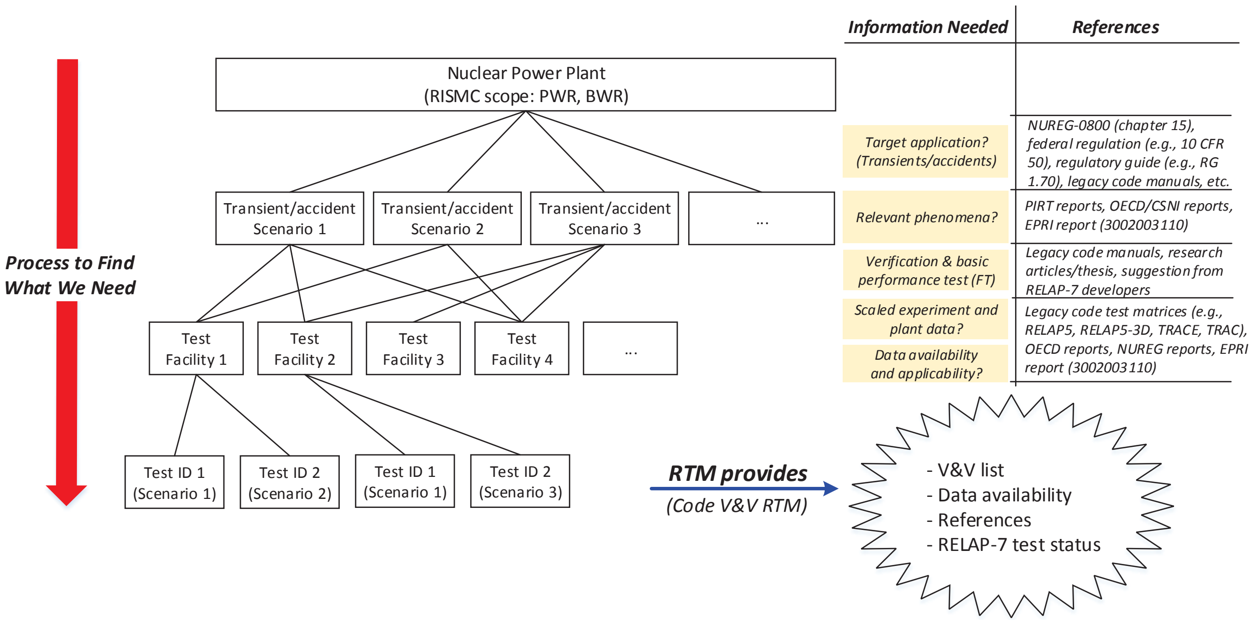

Figure 6. Information source tracking for *code V&V RTM*

Req #	Test Type		Requirement Specification (Target application, test feature)	Modification Date (mm/dd/yyyy)	NPP Design Targeted	Scale of Experiment	Reference (Experiment, Test ID, Reference)	Data Availability	RELAP-7 Test Status
VR-1	Code Ver	ver	Single-phase analytical test without flow (uniform pressure, zero flow velocity, variable cross-sectional area)	9/30/2016	-	-	Suggested by RELAP-7 development team	N/A	
VR-2		ver	Two-phase analytical test without flow (uniform pressure, zero flow velocity, spatially varying volume fraction)	9/30/2016	-	-	Suggested by RELAP-7 development team	N/A	
⋮									
VR-15	Sol Ver	ver	Numerical error estimation for a given physics equation depending on spatial and temporal discretization	9/30/2016			- Specific problems should be determined.	N/A	
VR-16		ver	Numerical error estimation for iterative schemes (e.g., JFNK)	9/30/2016			- Specific problems should be determined.	N/A	
⋮									
VR-26	Model Val	val	Two-phase shock problem (boiling front propagation)	9/30/2016	-	FT	J.R. Simoes-Moreira and J.E. Shepherd (J. Fluid Mech., "Evaporation waves in superheated dodecane," 1999)	A	
VR-27		val	Single-phase pressure drop at geometric discontinuities	9/30/2015	-	FT	- Ferrell-McGee pressure drop test (RETRAN-3D, TRACE) (abrupt area change test section data)	A	
VR-28		val	Two-phase pressure drop at geometric discontinuities	9/30/2015	-	FT	- Ferrell-McGee pressure drop test (TRACE) (abrupt area change test section data)	A	
⋮									

Figure 7. An example of *code V&V RTM* structure

6. SUMMARY AND CONCLUSIONS

This document addresses two subjects involved with the RELAP-7 assessment plan (RELAP-7 SVVP). The first discussion is about how the organization that is responsible RELAP-7 assessment should be operated in relation to the other organizations in INL (*e.g.*, RELAP-7 development team); in particular, the independence of the code assessment activities in view of technical, managerial, and financial aspects is emphasized. Next, as a second discussion, the specific work performed by the INL RELAP-7 assessment team is discussed. The work scope largely extends from establishing the fundamental strategy of RELAP-7 assessment to identifying the RELAP-7 requirements through the extensive literature review. Then, the RELAP-7 assessment plan is proposed in the form of three RTMs. These RTMs provide an efficient way to evaluate the RELAP-7 development status and the maturity of RELAP-7 assessment activities with time from various perspectives.

Note that the RTMs will be continuously updated through the future work, especially for the verification test problems and the validation data/tests.

REFERENCES

1. EPRI, *Desired characteristics for next generation integrated nuclear safety analysis method and software*. 2010: Palo Alto, CA. 1021085.
2. *RELAP-7 Theory Manual*. 2015, INL: Idaho National Laboratory, Idaho Falls, Idaho, USA.
3. Choi, Y.J., J. Yoo, and C.L. Smith, *RELAP-7 Software Verification and Validation Plan: Requirements Traceability Matrix (RTM) Part 1–Physics and numerical methods*. 2015, Idaho National Laboratory (INL), Idaho Falls, ID (United States).
4. Smith, C.L., Y.-J. Choi, and L. Zou, *RELAP-7 software verification and validation plan*. 2014, Tech. Rep. INL/EXT-14-33201, Idaho National Laboratory, USA.
5. Society, I.C., *IEEE Standard for Software Verification and Validation*. 2004: New York, NY10016-5997, USA.
6. NRC, U.S., *Softward Quality Assurance Program and Guidelines*. 1993: U.S. NRC, Washington, D.C., USA.
7. NRC, U.S., *Verification, Validation, Reviews, and Audits for Digital Computer Software used in Safety Systems of Nuclear Power Plants*. 2013: U.S. NRC, Washington, D.C., USA.
8. Society, I.C., *IEEE Standard for Software Reviews and Audits*. 2008: New York, NY10016-5997, USA.
9. DOE, *DOE O 414.1C (Quality Assurance)*. 2005: U.S. Department of Energy, Washington, D. C., USA.
10. DOE, *DOE G 414.1-4 (Safety Software Guide for Use with 10 CFR 830 Subpart A, Quality Assurance)*. 2010: U.S., Department of Energy, Washington, D.C., USA.
11. ASME, *NQA-1: An Overview for Federal Project Directors*. 2008: http://energy.gov/projectmanagement/downloads/nqa-1-overview-federal-project-directors.
12. Petruzzi, A. and F. D'Auria, *Thermal-Hydraulic System Codes in Nulcear Reactor Safety and Qualification Procedures.* Science and Technology of Nuclear Installations, 2008.
13. OECD/CSNI, *Proceedings of the OECD/CSNI workshop on transient thermal-hydraulic and neutronic codes requirements. NEA/CSNI/R(97)4*. 1996.
14. Aksan, N., D. Bessette, and e.a. I. Brittain, *Code validation matrix of thermo-hydraulic codes for LWR LOCA and transients*. 1987: Paris, France.
15. Aksan, N., et al., *Separate effects test matrix for thermal-hydraulic code validation: phenmena characterisation and selection of facilities and tests*. 1993.
16. *RELAP5-3D code manual - Volume III: development assessment*. 2015: Idaho National Laboratory, Idaho Falls, Idaho, USA.
17. *TRACE V5.0, Assessment Manual, Main Report*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
18. *TRAVE V5.0, Assessment Manual, Appendix A - Fundamental Validation Cases*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
19. *TRACE V5.0, Assessment Manual, Appendix B - Separate Effects Tests*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
20. *TRACE V5.0, Assessment Manual, Appendix C - Integral Effects Tests*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
21. EPRI, *Data Sources for Capability Assessments of Next Generation Safety Analysis Codes*. 2014, EPRI: Palo Alto, CA, USA.
22. Bonuccelli, M., et al. *A methodology for the qualification of thermalhydraulic codes nodalizations*. in *International Topical Meeting on Reactor Thermal Hydraulics (NURETH '93)*. 1993. Grenoble, France, October.
23. Ashley, R., et al., *Good practices for user effect reduction. NEA/CSNI/R(98)22*. 1998.
24. *IAEA Safety Report Series No. 23 (2002). Accident analysis for nuclear power plants*. IAEA, Vinenna.

25. D'Auria, F., M. Leonardi, and R. Pochard. *Methodology for the evaluation of thermalhydraulic codes accuracy*. in *International Conference on New Trends in Nuclear System Thermalhydraulics*. 1994. Pisa, Italy.
26. D'Auria, F. and G. Galassi, *Scaling in nuclear reactor system thermal-hydraulics.* Nuclear Engineering and Design, 2010. **240**(10): p. 3267-3293.
27. *IAEA Safety Report Series No. 52 (2008). Best estimate safety analysis for nuclear power plants: uncertainty evaluation*. IAEA, Vienna.
28. Aksan, S.N., F. Dauria, and H. Stadtke, *User Effects on the Thermal-Hydraulic Transient System Code Calculations.* Nuclear Engineering and Design, 1993. **145**(1-2): p. 159-174.
29. Petruzzi, A. and F. D'Auria, *Evaluation of uncertainties in system thermal-hydraulic calculations and key applications by the CIAU method.* International Journal of Nuclear Energy Science and Technology, 2008. **4**(2): p. 111-131.
30. Okawa, T. and Y. Kudo. *Effect of Interfacial Pressure Term on Numerical Stability of a Two-Fluid Model*. in *2014 22nd International Conference on Nuclear Engineering*. 2014. American Society of Mechanical Engineers.
31. Stuhmiller, J., *The influence of interfacial pressure forces on the character of two-phase flow model equations.* International Journal of Multiphase Flow, 1977. **3**(6): p. 551-560.
32. D'Auria, F. and W. Giannotti, *Development of a code with the capability of internal assessment of uncertainty.* Nuclear Technology, 2000. **131**(2): p. 159-196.
33. Karwat, I.H., *CSNI international standard problems (ISP)*. 2000.
34. Wolfert, K. and I. Brittain, *CSNI Validation Matrix for PWR and BWR Thermal-Hydraulic System Codes.* Nuclear Engineering and Design, 1988. **108**(1-2): p. 107-119.
35. Lewis, M., *State of the art report (SOAR) on Thermalhydraulics of Emergency Core Cooling in Light Water. Reactors*. 1989, OECD-NEA-CSNI report.
36. Petruzzi, A. and F. D'Auria, *Standardized Consolidated Calculated and Reference Experimental Database (SCCRED): A Supporting Tool for V&V and Uncertainty Evaluation of Best-Estimate System Codes for Licensing Applications.* Nuclear Science and Engineering, 2016. **182**(1).
37. Choi, K.-Y., et al., *A summary of 50th OECD/NEA/CSNI International Standard Problem exercise (ISP-50).* Nuclear Engineering and Technology, 2012. **44**(6): p. 561-586.
38. Yang, J.-H., et al., *Experimental study on two-dimensional film flow with local measurement methods.* Nuclear Engineering and Design, 2015. **294**: p. 137-151.
39. Aksan, N. *Overview on CSNI separate effects test facility matrices for validation of best estimate thermal-hydraulic computer codes*. 2004. OECD/NEA Seminar on Transfer of Competence, Knowledge and Experience Gained Through CSNI Activities in the Field of Thermal-Hydraulics, INSTN, Saclay, France.
40. Mimouni, S. and G. Serre, *List of benchmarks for simulation tools of steam-water two-phase flows.* 2001.
41. Barre, F. and M. Bernard, *The CATHARE code strategy and assessment.* Nuclear engineering and design, 1990. **124**(3): p. 257-284.
42. Boyack, B., M. Straka, and L. Ward, *TRAC-M Validation Test Matrix.* http://lib-www. lanl. gov/la-pubs/00818499. pdf, 2001: p. 4.
43. Oberkampf, W.L. and T.G. Trucano, *Verification and validation in computational fluid dynamics.* Progress in Aerospace Sciences, 2002. **38**(3): p. 209-272.
44. Oberkampf, W.L., M. Pilch, and T.G. Trucano, *Predictive capability maturity model for computational modeling and simulation*. 2007, Sandia National Laboratories.
45. Aksan, N., et al., *Evaluation of the Separate Effects Tests (SET) Validation Matrix.* OECDCSNI Report OECD/GD (97). **9**.

APPENDIX A. RELAP-7 GENERAL RTM

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 (archived 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-4		Capability of simulating various ECCS design influencing LOCA simulation (accumulators, safety injection systems such as UPI, cold-leg/hot-leg injection)	9/30/2015	VR-55, 56, 60-62, 99	Not tested
GR-5		Capability of modeling various plant components and systems for non-LOCA simulation	9/30/2015	All non-LOCA tests included in Code V&V RTM can be used to demonstrate this capability.	Not tested
GR-6		Capability of simulating advanced reactor designs with non-water coolants	9/30/2015	-	Not tested
GR-7	T/H System Safety Analysis (Design- and Licencing-Basis Transients/Accidents)	LBLOCA analysis capability	9/30/2015	SET: VR-51-53, 55-58, 61-63 IET/PT: VR-78, 83, 99	Not tested
GR-8		SBLOCA analysis capability	9/30/2015	SET: VR-49 IET/PT: VR-77, 84, 126	Not tested
GR-9		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
GR-12		Capability of simulating increase/decrease in reactor coolant inventory events (non-LOCA)	9/30/2015	-	Not tested
GR-13		Capability of simulating Station Blackout (SBO) and its consequence	9/30/2016	VR-98, 115	Not tested

GR-14		Capability of simulating BWR stability	9/30/2016	VR-102	Not tested
GR-15		Capability of simulating ATWS transients	9/30/2016	VR-82, 90	Not tested
GR-16	**Multi-physics Analysis (Reactor kinetics, fuel components behavior, chemical reactions, etc.)**	Generate physics parameters for reactor kinetics model in system code	9/30/2015	-	Sample Test 65-67
GR-17		Capability of coupled simulation with neutronics code (to reflect a reactivity feedback with 1D, multi-D core power calculation)	9/30/2016	-	Not tested
GR-18		Capability of coupled simulation for fluid/mechanical interaction analysis (e.g., water hammer, LOCA-load analysis)	9/30/2016	-	Not tested
GR-19		Capability of coupled simulation for T/H effect on structural integrity (e.g., PTS)	9/30/2016	-	Not tested
GR-20		Capability of coupled simulation with fuel performance code such as BISON (to reflect the feedback from mechanical/ thermal behavior of fuel pellet, gap, and cladding depending on reactor conditions)	9/30/2015	-	Not tested
GR-21		Capability of simulating chemical effect (e.g., localized corrosion, CRUD)	9/30/2015	-	Not tested
GR-22		Capability of simulating containment analysis (e.g., FP, aerosol behavior)	9/30/2015	-	Not tested
GR-23		Capability of simulating radiological consequence analysis	9/30/2015	-	Not tested
GR-24	**Multi-D & Multi-scale Analysis**	Capability of simulating multi-dimensional fluid flow (e.g., PWR reflood phenomena after LOCA)	9/30/2015	SET: VR-60-65 IET/PT: VR-99, 132	Not tested
GR-25		Capability of simulating multi-dimensional fluid flow at microscale level of detail (e.g., flashing, critical flow, boiling, etc.)	9/30/2015	-	Not tested
GR-26	**Integrated/Improved Uncertainty Analysis**	Incorporating uncertainty quantification process into an integral part of the simulation (e.g., coupling with RAVEN)	9/30/2016	-	Not tested

APPENDIX B. RELAP-7 SPECIFIC RTM

Req #	Category	Requirement Specification	Modification Date (mm/dd/yyyy)	Verification/Action Item	RELAP-7 Status
SR-1	**Computer Science & Software Architecture**	Use of the most advanced computer science technology (both computing power and numerical solvers) to optimize both accuracy and simulation speed	9/30/2015	Descritization scheme Time integration method Matrix solver Parallel computation capability	Written with C++. Capable of Multi-scale time integration, PCICE (operator split), JFNK (implicit nonlinear Newton method), and a point implicit method (long duration transients). New pipe network algorithm based upon Mortar FEM (Lagrange multipliers). Ability to couple to multi-dimensional reactor simulators
SR-2		Numerically robust and reliable (e.g., not subject to failure as a result of numerical methods)	9/30/2015	Numerical stability test Steady-state initialization test	Not tested
SR-3		Multi-scale/multi-physics simulation capability for the following scope through coupling: (i) fuel rod, (ii) fuel assembly, (iii) reactor, (iv) primary RCS, (v) secondary coolant system and BOP, (vi) I&C, (vii) containment, (viii) site radiological consequences, (ix) offsite radiological consequences, (x) fluid/structure interaction for dynamic loads	9/30/2015	Coupling test with other MOOSE-based applications	RELAP-7 can be coupled with MOOSE framework application to simulate multi-scale / multi-physics problems
SR-4		User-friendly steady-state initialization and restart capabilities	9/30/2015	-	Both steady-state and transient cases can be simulated by restart option
SR-5		Clear and easy diagnostics to assist with debugging and workaround	9/30/2015	-	Code will show highlighted error signal. For example, if wrong model type was give then shows in red: ***ERROR*** Unknown model type
SR-6		Comprehensive GUI for pre/post-processing and on-line monitoring	9/30/2015	-	in progress
SR-7		Coupling capability with other MOOSE-based codes (e.g., RAVEN, BISON) and CFD codes	9/30/2015	Verification test for coupling with other MOOSE-based applications	Can be coupled with MOOSE-based BISON code for 3D neutron transport model.
SR-7	**Code and Modeling Accuracy with Reliability**	Capable of achieving CFD-like resolution (3D) in selected parts (i.e., easily adjust the grid resolution as needed)	9/30/2015	Mesh management test	RELAP-7 does not have 3D mesh generation model.
SR-8		Coordinate system to represent the actual design of a component with high fidelity	9/30/2015	-	Provides x, y, z coordination system for components, functions, etc.
SR-9		Providing standard modules with limited options for various components or systems to lessen the user	9/30/2015	-	RELAP-7 supports standard component models

		effect			
SR-10		Providing standard or recommended options to lessen the user effect on the result	9/30/2015	-	
SR-11	**Physical Phenomena**	Capability of addressing legacy issues associated with two-phase flow (e.g., (i) phase separation, (ii) flow-regime transition, (iii) level tracking, (iv) water-packing, (v) flooding, (vi) entrainment, etc.)	9/30/2015	-	7 equation two-phase model can be simulated
SR-12		Modeling capability of a droplet field for BWR core spray, containment spray (PWR/BWR), and core uncovery events	9/30/2015	-	Not tested
SR-13		Modeling capability of sources and transport of particles in vapor, gas, droplet and liquid	9/30/2015	Model V&V in RELAP 7 framework and/or code-to-code comparison	Not tested
SR-14		Modeling capability of non-condensable gas transport and its heat transfer effect	9/30/2015	Model V&V in RELAP 7 framework and/or code-to-code comparison	The compressible valve component can handle non-condensible gas model
SR-15	**Software Quality Assurance**	Writing the source code under a consistent programming standard for simplified maintenance and revision	9/30/2015	-	
SR-16		Providing detailed documentation of theory, programming, user manual, validation basis and user guidelines	9/30/2015	-	RELAP-7 provides revised theory manual. Other documents are in progress

APPENDIX C. RELAP-7 CODE V&V RTM

(A: Available, N/A: Not applicable or not available, P/A: Partially available, ver: verification, val: validation, sol: solution)

Req #	Test Type		Requirement Specification (Target application, test feature)	Modification Date (mm/dd/yyyy)	NPP Design Targeted	Scale of Experiment	Reference (Experiment, Test ID, Reference)	Data Availability	RELAP-7 Test Status
VR-1	**Code Ver**	ver	Single-phase analytical test without flow (uniform pressure, zero flow velocity, variable cross-sectional area)	9/30/2016	-	-	Suggested by RELAP-7 development team	N/A	
VR-2		ver	Two-phase analytical test without flow (uniform pressure, zero flow velocity, spatially varying volume fraction)	9/30/2016	-	-	Suggested by RELAP-7 development team	N/A	
VR-3		ver	Two-phase analytical test for volume fraction advection with uniform pressure and velocity field	9/30/2016	-	-	Suggested by RELAP-7 development team	N/A	
VR-4		ver	Grid convergence study for single-/two-phase fluid flow problems (comparing with highly accurate solutions)	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-5		ver	Grid convergence study for heat conduction problems (comparing with highly accurate solutions)	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-6		ver	Grid convergence study with available stabilization schemes (e.g., SUPG, Lapidus, Entropy based viscosity scheme)	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-7		ver	Time step convergence study with available options (comparing with highly accurate solutions) (e.g., Backward Euler, Crank-Nicolson, BDF2)	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-8		ver	Iterative scheme convergence tests (e.g., JFNK)	9/30/2016	-	-	- Specific problems should be determined.	N/A	

VR-9		ver	Symmetry solution tests with various boundary conditions (e.g., fully developed channel flow problem)	9/30/2016	-	-	'- Specific problems should be determined.	N/A	
VR-10		ver	Solution independence tests to coordinate transformation (e.g., rotation, translation of physical domain)	9/30/2016	-	-	- Specific problems should be determined.	N/A	
VR-11		ver	Conservation tests in 0-dimensional components (e.g., Branches/Junctions, LWR components like steam generator or pressurizer)	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-12		ver	Conservation tests in system level of loop configuration	9/30/2015	-	-	- INL/EXT-14-33201 - Specific problems should be determined.	N/A	
VR-13		ver	Consistency test of simulation results depending on computer hardware and/or software environment (e.g., compiler, libraries, etc.)	9/30/2016	-	-	- Specific problems should be determined.	N/A	
VR-14		ver	Propagation of a passive scalar property (related to the capability of particle transport simulation)	9/30/2016	-	-	- Specific problems should be determined.	N/A	
VR-15		ver	Regression test and code coverage test after any updates in the source code	9/30/2015	-	-	-	N/A	
VR-16	**Sol Ver**	ver	Numerical error estimation for a given physics equation depending on spatial and temporal discretization	9/30/2016	-	-	- Specific problems/applications should be determined.	N/A	
VR-17		ver	Numerical error estimation for all non-linear solver settings (e.g., JFNK)	9/30/2016	-	-	- Specific problems/applications should be determined.	N/A	
VR-18		ver	Laminar single-phase flow in a heated tube	9/30/2016	-	-	- Hagen-Poiseuille type pipe flow with heated wall	N/A	
VR-19		ver	Gravitational head effect	9/30/2016	-	-	- Water faucet problem (RELAP5-3D) - Water over steam problem (RELAP5-3D)	N/A	
VR-20		ver	Heat conduction (1D/Multi-D)	9/30/2016	-	-	Heat conduction enclosure (RELAP5-3D, TRACE)	N/A	

VR-21		ver	Decay heat model test with variоius decay options	9/30/2015	-	-	Decay heat model test (RELAP5-3D)	N/A	
VR-22		ver	Reactor kinetics model	9/30/2015	-	-	Reactor kinetics model test (RELAP5-3D)	N/A	
VR-23		ver	Metal-water reaction model (e.g., Zr-cladding oxiation)	9/30/2015	-	-	Metal-water reaction model test (RELAP5-3D)	N/A	
VR-24		ver	Wall-to-fluid friction (single phase)	9/30/2015	-	-	- Darcy pressure drop equation (horizontal pipe) (TRACE) - Wang's falling film data (TRACE)	N/A, A	
VR-25		ver	Single-phase shock problem	9/30/2016	-	FT	- D. L. Youngs, "Shock Tube", Multiphase Science and Technology, Vol. 6 p653-662 - S. Mimouni and G. Serre, "List of benchmarks for simulation tools of steam-water two-phase flows", 2000	N/A	
VR-26	**Model Val**	val	Wall-to-fluid friction (two phase)	9/30/2015	-	FT	- Ferrell-Bylund uniform test section data (TRACE)	A	
VR-27		val	Two-phase shock problem (boiling front propagation)	9/30/2016	-	FT	J.R. Simoes-Moreira and J.E. Shepherd (J. Fluid Mech., "Evaporation waves in superheated dodecane," 1999)	A	
VR-28		val	Single-phase pressure drop at geometric discontinuities	9/30/2015	-	FT	- Ferrell-McGee pressure drop test (RETRAN-3D, TRACE) (abrupt area change test section data)	A	
VR-29		val	Two-phase pressure drop at geometric discontinuities	9/30/2015	-	FT	- Ferrell-McGee pressure drop test (TRACE) (abrupt area change test section data)	A	
VR-30		val	Water hammer or pressure wave propagation (single-phase)	9/30/2015	-	FT	- EPRI NP-6766, Vol.4, Part1 (1992) - NUREG/IA-0206 (2007) - Simpsons water hammer test (A.R. Simption's PhD Thesis, 1986; Serre and Bestion, "Two-Phase Water Hammer Simulation with CATHARE Code")	A	
VR-31		val	Water hammer or pressure wave propagation (two-phase)	9/30/2015	-	FT	- Tiselj and Cerne (Nucl. Sci. Eng., Vol. 134, 2000) - Cerne et al. (Trans ANS, Vol. 75, 1996) - Serre and Bestion ("Two-Phase Water Hammer Simulation with CATHARE Code")	A	

VR-32		val	Flow split (T-junction)	9/30/2016	-	FT	-	N/A	
VR-33		ver, val	Convective heat transfer (single-phase)	9/30/2015	-	FT	- Turbulent forced convection: Dittus-Boelter, Petukhov, Inayatov (for vertical bundles), etc. - Laminar forced convection: Nu=7.63 (ORNL/ANS/INT-5/V19, RELAP5-3D), Elenbaas, etc. - Natural convection: McAdams, Churchill-chu, etc. - No models for forced laminar or natural convection for vertical bundles	P/A	
VR-34		val	Interphase friction in vertical flow	9/30/2015	-	FT	- CISE Adiabatic Tube (TRACE)	A	
VR-35		val	Phase separation/distribution in vertical flow	9/30/2015	-	FT	- Wilson Bubble Rise test data (TRACE) - GE Vessel Blowdown Level Swell data (1 ft small diameter vessel; 4 ft large diameter vessel) (TRACE, RELAP5-3D, RETRAN-3D) - Sedimentation test problem (RELAP-7 HPC repository)	A	
VR-36		val	Phase separation/distribution in horizontal flow	9/30/2015	-	FT	- Edward's Pipe Blowdown data (RELAP5,RETRAN-3D): ISP-01 - TPTF Horizontal Flow (TRACE)	A	
VR-37		val	Phase separation/distribution at branch	9/30/2016	-	FT	-	N/A	
VR-38		val	Level tracking during flow oscillation (single-phase)	9/30/2015	-	FT	- Fill-drain assessment problem - Manometer problem (RELAP5-3D, TRACE) - Gravity wave tests (1D, 3D) (RELAP5-3D)	A	
VR-39		val	Two-phase mixture level swell	9/30/2015	-	FT	- Single tube flooding test (TRACE) - Bubbling steam through liquid (RELAP5-3D)	-	
VR-40		val	Entrainment/de-entrainment in vertical flow	9/30/2015	-	FT	- GE Vessel Blowdown Level Swell data (1 ft small diameter vessel; 4 ft large diameter vessel) (TRACE, RELAP5-3D, RETRAN-3D)	A	
VR-41		val	Flashing in vertical flow	9/30/2016	-	FT	- GE Vessel Blowdown Level Swell data (1 ft small diameter vessel; 4 ft large diameter vessel) (TRACE, RELAP5-3D, RETRAN-3D)	A	

VR-42		val	Flashing in horizontal flow	9/30/2016	-	FT	- Edward's Pipe Blowdown data (RELAP5,RETRAN-3D): ISP-01 - TPTF Horizontal Flow (TRACE) - Saruel et al. (2008), "Modelling phase transition in metastable liquids: application to cavitating and flashing flows," J. Fluid Mech.	A	
VR-43		val	Counter-current flow	9/30/2015	-	FT	- Single tube flooding test (TRACE)	A	
VR-44		val	Counter-current flow limitation (CCFL)	9/30/2015	-	FT	- Single tube flooding test (TRACE) - Bankoff CCFL test (TRACE) - Dukler-Smith Air-Water Flooding test (RELAP5-3D)	A	
VR-45		val	Boiling heat transfer	9/30/2015	-	FT	- Christensen Subcooled Boiling (RELAP5, RELAP5-3D) - Bennett Heated Tube (RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-46		val	Critical Heat Flux (CHF)/dryout	9/30/2015	-	FT	- Bennett Heated Tube (RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-47		val	Re-wetting heat transfer	9/30/2015	-	FT	- ORNL THTF Transient Blowdown test (TRACE) - GOTA BWR Reflood test (TRACE)	A	
VR-48		val	Flim Boiling (FB)/superheating heat transfer	9/30/2015	-	FT	- Bennett Heated Tube (RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-49		val	Superheating due to compression	9/30/2016	-	FT	- MIT pressurizer (TRACE, RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-50		val	Radiation heat transfer	9/30/2016	-	FT	- GOTA BWR Radiation (Run 27, TRACE)	A	
VR-51		val	Interphase heat transfer	9/30/2016	-	FT	- UCB-Kuhn Condensation (TRACE) - MIT pressurizer (TRACE, RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-52		val	Condensation heat transfer	9/30/2015	-	FT	- Dehbi-MIT Condensation With NCG (TRACE) - University of Wisconsin Condensation (TRACE)	A	
VR-53		val	Critical flow and blowdown	9/30/2015	-	FT	- Marviken test data (NUREG/IA-0007) (TRACE, RELAP5, RELAP5-3D, RETRAN-3D) - Moby Dick nozzle tests (RELAP5-3D, TRACE) - Super Moby Dick - Edwards-O'Brien blowdown test (RELAP5-3D), ISP-01	A	

VR-54		val	Non-condensable gas effects	9/30/2016	-	FT	- UCB-Kuhn Condensation Tests (TRACE) - Dehbi-MIT Condensation Tests (TRACE) - University of Wisconsin Condensation Tests (TRACE)	A	
VR-55		val	Single-phase natural circulation	9/30/2015	-	FT	NUREG/IA-0151 (1999)	A	
VR-56		val	Boron mixing and transport	9/30/2015	-	FT	-	N/A	

- Code V&V RTM for SET and CT (continued)

Req #	Test Type		Requirement Specification (Target application, test feature)	Modification Date (mm/dd/yyyy)	NPP Design Targeted	Scale of Experiment	Reference (Experiment, Test ID, Reference)	Data Availability	RELAP-7 Test Status
VR-57	**Model Val**	val	SBLOCA (Boil-off, Void Distribution)	09/31/2016	PWR	SET	ORNL THTF SBLOCA test series data: - Tests 3.09.10I to 10N (core uncovered) - Tests 3.09.10AA to 10FF (core covered)* - NUREG/CR-2456, NUREG/CR-2640 - (TRACE, RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-58		val	Film Boiling Heat Transfer, CHF	09/31/2016	PWR	SET	ORNL THTF Film Boiling Bundle Uncovery CHF data: - Tests 3.07.9B, H, N, W - NUREG/CR-2640 - (TRACE, RELAP5, RELAP5-3D)	A	
VR-59		val	Blowdown, Film Boiling Heat Transfer	09/31/2016	PWR	SET	ORNL THTF Transient Blowdown data: - Tests 3.03.6AR, 3.06.6B, 3.08.6C - NUREG/CR-2640 - (TRACE)	A	
VR-60		val	Blowdown (Pressure/Void fraction Variation)	09/31/2016	-	SET	- GE Level Swell Test, 1 ft. diameter: Test 1004-3 - GE Level Swell, 4 ft. diameter: Tests 5801-15, 5702-16 (RELAP5-3D, TRACE)	A	
VR-61		val	ECC Bypass (downcomer), CCFL	09/31/2016	PWR	SET	UPTF Downcomer CCFL Test - Test 6, Run 131 (RELAP5-3D) - Test 5, 6, 7, 21 (TRACE)	A	

VR-62		val	Phase Distribution in a BWR core	09/31/2016	BWR	SET	FRIGG experiments void data - FRIGG-2 Tests 313001 to 20, 24, 27, 30, 34, 37, 40, 43, 56, 60 - FRIGG-4 Tests 613001, 10, 13, 14, 19 (TRACE, RETRAN-3D)	A	
VR-63		val	LBLOCA, Reflood	09/31/2016	BWR	SET	GOTA BWR Reflood test data: - Run 42 (reflood experiment) (TRACE)	A	
VR-64		val	LBLOCA, Reflood	09/31/2016	PWR	SET	1. FLECHT-SEASET Reflood Heat Transfer data: - Tests 31108, 31504, 31701, 31203, 31805, 32114, 32013, 31302 - (TRACE, RELAP5, RELAP5-3D, RETRAN-3D) 2. RBHT Reflood Heat Transfer test data: - Tests 1096, 1108, 1170, 1196, 1285, 1383 - (TRACE)	A	
VR-65		val	LBLOCA, Steam-Cooling	09/31/2016	PWR	SET	RBHT Reflood Heat Transfer test data: - Tests 3173A, 3216D, 3205A, 3216A, 3216G, 3205G, and 3214A - NUREG/CR-7152 (TRACE)	A	
VR-66		val	LBLOCA, Core Uncovery	09/31/2016	PWR	SET	RBHT Reflood Heat Transfer test data: - Tests 1560, 1566, 1570, 1572, 1582, 1637, 1648, 1651, 1659 (steady-state test) - Test 1690 (transient test) (TRACE)	A	

VR-67		val	Boil-off	09/31/2016	PWR	SET	FLECHT-SEASET Boil Off test data: - Test 35658 (RELAP5)	A	
VR-68		val	LBLOCA, Reflood (Multi-D effects)	09/31/2016	PWR	SET	1. UPTF - Tests 5A, 6 (TRACE) - NUREG/IA-0127, GRS-100 (ISBN: 3-923875-50-9) 2. BCL - Test 29402 (Transient, CSNI Report (87)132) - NUREG-1230 3. CREARE - NUREG-1230, CSNI report (87)132	P/A	
VR-69		val	LBLOCA, Reflood (Multi-D effects)	09/31/2016	PWR (B&W)	SET	1. UPTF - NUREG/IA-0127 2. CCTF - NUREG/IA-0127	P/A	
VR-70		val	LBLOCA, Reflood (Multi-D effects)	09/31/2016	BWR	SET	SSTF - Test EA 3.1: LBLOCA recirculation line rupture (TRACE) - Test EA 3.3-1 LBLOCA 73% recirculation line rupture (TRACE)	A	
VR-71		val	BWR Core Spray Distribution (Multi-D TH effects)	09/31/2016	BWR	SET	SSTF - Tests CS-1.3, CS-1.3A (NUREG-1230)	P/A	
VR-72		val	Cold Leg and Downcomer ECCS Mixing (Multi-D TH effects)	09/31/2016	PWR	SET	* CREARE 1/2 Scale Facility - NUREG-1809 (App. B.3) * UPTF - Test 1 - NUREG-1809, App. B.7; NUREG/IA-0127)	P/A	

VR-73		val	UPI ECCS during LOCA (Multi-D effects)	09/31/2016	PWR	SET	1. UPTF - NUREG/IA-0127 2. CCTF - NUREG/IA-0127, NUREG-1230 (Rev. 4) 3. SCTF - NUREG/IA-0127, NUREG-1230 (Rev. 4)	P/A	
VR-74		val	Jet Pump	09/31/2016	BWR	CT	1. INEL 1/6 Scale Jet Pump Test - RELAP5-3D - H. S. Crapo, Idaho National Engineering Report (EGG-LOFT-6063), Nov. 1979 2. Small Scale Jet Pumps for the FIST facility - NUREG/CR-2576 3. Full Scale Jet Pumps - For BWR4, Boiling Water Reactor Turbine Trip (TT) Benchmark, Vol. I: Final Specifications, NEA/NSC/2001-1. - For BWR6 (N/A)	A	
VR-75		val	Recirculation pump	09/31/2016	BWR	CT	1. Small Scale Recirculation Pumps in FIST facility -NUREG/CR-2576 2. Full Scale Recirculation Pumps - For BWR4, Boiling Water Reactor Turbine Trip (TT) Benchmark, Vol. I: Final Specifications, NEA/NSC/2001-1.	A	
VR-76		val	Separator	09/31/2016	BWR	CT	- Boiling Water Reactor Turbine Trip (TT) Benchmark, Vol. I: Final Specifications, NEA/NSC/2001-1.	A	
VR-77		val	Reactor coolant pump steady-state, startup, and coastdown	09/31/2016	PWR	CT		A	

VR-78		val	Reactor coolant pump two-phase operation	09/31/2016	PWR	CT	1. Scaled RCP - LOFT tests - ROSA-IV tests 2. Scaled RCP two-phase - EPRI/CE 1/5 scale (EPRI NP-1556) - LOFT Tests L3-5 and L3-6	A	
VR-79		val	Pressurizer	09/31/2016	PWR	CT	1. Full scale pressurizer - Doel 4 startup test (NUREG/IA-0020) 2. Scaled pressurizer - MIT pressurizer (TRACE, RELAP5, RELAP5-3D, RETRAN-3D) - NEPTUNUS test (RELAP5-3D, NUREG/IA-0040) - ISP-38	A	
VR-80		val	Accumulator	09/31/2016	PWR	CT	Scaled accumulator data - LOFT accumulator blowdown test (L3-1) (RELAP5, RELAP5-3D, RETRAN-3D)	A	
VR-81		val	U-tube steam generator	09/31/2016	PWR	CT	1. Full scale U-tube steam generator - NUREG/IA-0113 - NUREG/IA-0106 2. Scaled U-tube steam generator - Westinghouse Model Boiler-2 (NUREG/IA-224) - Kalra, S., Yao, L. S., and Davis, W. E. R. "Flow Behavior in a Static Vane Centrifugal Separator-Simulation Experiments and Analysis," Second International Topical Meeting on Nuclear Reactor Thermal Hydraulics, January, 1983. (RELAP5-3D, TRACE)	A	
VR-82		val	Once-through steam generator (OTSG)	09/31/2016	PWR (B&W)	CT	1. Scaled OTSG - NUREG/CR-5395, NUREG/CR-4567 - "Simulation of a 30-Tube Once-Through Steam Generator with RELAP5/MOD3 and RELAP5/MOD2 Computer Codes," Hassan, Y. A., Salim, P., ANS Winter Meeting, November, 1990 (OSTI ID: 6780203)	A	

VR-83		val	Hot leg (two-phase)	09/31/2016	PWR (B&W)	CT	1. Scaled hot leg - NUREG/CR-5395 - NUREG/CR-4567 2. Full scale - No data exists	A	
VR-84		val	Reactor vessel internals vent valves (RVVV)	09/31/2016	PWR (B&W)	CT	1. Full Scale RVVV - NUREG-1230, Section 6.4.3.11 2. Scaled RVVV - UPTF test data (NUREG/IA-0127) - "Summary of Downcomer Injection Phenomena for UPTF and TRAC Post-Test Analysis," LACP-92-188, May 1992 - CCTF test data (NUREG/IA-0127): Test C2-AS2, Test C2-10	A	

- Code V&V RTM for IET and PT (continued)

Req #	Test Type		Requirement Specification (Target application, test feature)	Modification Date (dd/mm/yyyy)	NPP Design Targeted	Scale of Experiment	Reference (Experiment, Test ID, Reference)	Data Availability	RELAP-7 Test Status
VR-85	**Model Val**	val	SBLOCA	09/31/2016	BWR	IET	FIST (Full Integral System Test) facility - SBLOCA test 6SB2C - NUREG/CR-2576 - (TRACE)	A	
VR-86		val	LBLOCA	09/31/2016	BWR	IET	1. FIST facility - LBLOCA test 6DBA1B - NUREG/CR-2576, NUREG/CR-3711 2. FIST facility - LBLOCA test 4DBA1 - NUREG/CR-2576, NUREG/CR-4128 3. TLTA (Two Loop Test Apparatus) facility - LBLOCA Conservative test 6423 - NUREG/CR-2229, GEAP-23592 (TLTA facility description), GEAP-NUREG-23977 4. TLTA (Two Loop Test Apparatus) facility - LBLOCA test 6425 - NUREG/CR-2229, GEAP-23592 (TLTA facility description), GEAP-NUREG-23977 - (TRACE) 5. TLTA (Two Loop Test Apparatus) facility - LBLOCA tests 6425, 6424 - NUREG/CR-2229, GEAP-NUREG-23977 - (TRACE)	A	

VR-87		val	Natural Circulation	09/31/2016	BWR	IET	FIST program - Natural circulation test 6PNC2 - AURORA-B, ANP-10300 (2009), ML100040158 - NUREG/CR-2576, NUREG/CR-4128	A	
VR-88		val	Turbine Trip	09/31/2016	BWR	IET	FIST program - Turbine Trip test 4PTT1 - AURORA-B, ANP-10300 (2009), ML100040158 - NUREG/CR-2576, NUREG/CR-4128	A	
VR-89		val	SLB	09/31/2016	BWR	IET	FIST program - Steam Line Break test 6MSB1 - AURORA-B, ANP-10300 (2009), ML100040158 - NUREG/CR-2576, NUREG/CR-4128	A	
VR-90		val	ATWS (MSIV closure w/o HPCS)	09/31/2016	BWR	IET	FIST program - ATWS test 6PMC2 - NUREG-1230 (Rev. 4) - NUREG/CR-2576, NUREG/CR-3711	A	
VR-91		val	LBLOCA	09/31/2016	PWR	IET	1. LOFT (Loss of Fluid Test) - Test L2-5 (ISP-13) (TRACE, RELAP5, RELAP5-3D) - Test LB-1 (TRACE) - NUREG/IA-28 2. PKL facility - Test K9 (ISP-10) - CSNI Report No. 64 (1981) 3. Achilles - CSNI Report No. 11 (ISP-25)	A	

VR-92		val	SBLOCA	09/31/2016	PWR	IET	1. LOFT (Loss of Fluid Test) - Test L3-1 (ISP-09) (TRACE) - Test L3-5 (RCP running) (NUREG/IA-0024) - Test L3-6 (RCP tripped) (ISP-11) 2. ROSA-IV - Test SB-CL-01 2.5% CLB with delayed ECCS (TRACE) - Test SB-CL-05 5% CLB with ECCS and AFW (TRACE) - Test SB-CL-14 10% CLB with ECCS with LPI only (TRACE) - Test SB-CL-15 0.5% CLB with no ECCS and no AFW (TRACE) - Test SB-CL-18 5% CLB with LPI only (TRACE, RELAP5, REAL5-3D) (NUREG/IA-0095, ISP-26, CSNI (91)13) - Test IB-CL-02 17% CLB	A	
VR-93		val	Loss of load	09/31/2016	PWR	IET	LOFT (Loss of Fluid Test) - No publicly available data or reference	A	
VR-94		val	RCP trip	09/31/2016	PWR	IET	LOFT - No publicly available data or reference	A	
VR-95		val	Excessive load increase	09/31/2016	PWR	IET	LOFT - No publicly available data or reference	A	
VR-96		val	Overcooling (Increase in secondary heat removal)	09/31/2016	PWR	IET	LOFT - Tests L6-7, L9-2 - NUREG/CR-2277	A	
VR-97		val	LOAF with subsequent feed-and-bleed operation	09/31/2016	PWR	IET	1. LOFT - Tests L9-1, L3-3 - NUREG/IA-0114 (RELAP5/MOD3) - NUREG/IA-0228 (RELAP5/MOD3.3) 2. ROSA-IV program - No publicly available data or reference	A	

VR-98		val	ATWS (LOFW)	09/31/2016	PWR	IET	1. LOFT - Test L9-3 - NUREG/IA-0192 (RELAP5/Mod3.2.2) 2. ROSA-IV - Test 3-2 - NUREG/IA-0410 (RELAP5)	A	
VR-99		val	Turbine Trip	09/31/2016	PWR	IET	ROSA-IV program - No publicly available data or reference	N/A	
VR-100		val	LOFW	09/31/2016	PWR	IET	ROSA-IV program - No publicly available data or reference	N/A	
VR-101		val	Natural circulation (single-phase)	09/31/2016	PWR	IET	1. ROSA-IV program (single-phase test) - Test 1.1 (NUREG/IA-0419, TRACE) 2. SEMISCALE experiment - S-NC-1, S-NC-10 (RELAP5-3D) - S-NC-2 (RELAP5-3D, TRACE) 3. PACTEL natural circulation experiment - ISP-33 4. PANDA natural circulation tests - ISP-42 (PCCS for ALWR is of main interest.)	A	

VR-102		val	Natural circulation (two-phase)	09/31/2016	PWR	IET	1. SEMISCALE natural circulation tests (Mod-2A) - S-NC-2, S-NC-3 (RELAP5-3D, TRACE) - S-NC-10 (RELAP5-3D) 2. PACTEL natural circulation experiment - ISP-33 3. PANDA natural circulation tests - ISP-42 (PCCS for ALWR is of main interest.) 4. PKL facility - Tests PKL-B4.2, B4.3 (NUREG/IA-0170) - Tests PKL-1D1-9, 1D1-15 (NUREG/CR-3280) - CSNI Report No. 10 (1981) (ISP-10)	N/A	
VR-103		val	SGTR	09/31/2016	PWR	IET	1. ROSA-IV - Test SB-SG-06 - NUREG/IA-0130 2. BETHSY tests 4.3b, 3.4b (CATHARE-2) 3. LOFT tests L6-8C-1, L6-8C-C2 (CATHARE-2)	P/A	
VR-104		val	Multiple SGTR	09/31/2016	PWR	IET	ROSA-IV program - No publicly available data or reference	N/A	
VR-105		val	SLB	09/31/2016	PWR	IET	1. ROSA-IV - Test SB-SL-01 (10 % MSLB) - NUREG/IA-0148 2. Westinghouse Model Boiler-2 test facility - SLB tests 100%, 50%, 8% (with SGTR) - NUREG/IA-0106 3. LOBI facility - Test BT12 (NUREG/IA-0079)	P/A	
VR-106		val	SBO	09/31/2016	PWR	IET	ROSA-IV - No publicly available data or reference	N/A	

VR-107		val	Reflood during LOCA (Multi-D effects)	09/31/2016	PWR	IET	1. CCTF - Tests C2-4, C2-5, C2-8, C-12 (TRACE) - NUREG-1230, NUREG/IA-0127, GRS-100 (ISBN: 3-923875-50-9) 2. SCTF - Tests S2-01 (Run 606), S2-02 (Run 607), S2-06 (Run 611), S2-16 (Run 621), S2-17(Run 622) (TRACE) - Tests S2-SH1 (Run 604), S2-SH2 (Run 605) (TRACE) - NUREG-1230	P/A	
VR-108		val	Loss of decay heat removal	09/31/2016	PWR	IET	1. ROSA-IV - Four configurations were tested for Loss of Decay Heat Removel scenarios, each with three different opening areas (NUREG/IA-0143) (i. Loop intact, ii. Cold leg opening (RCP maintenance), iii. SG manway open, iv. Pressurizer manway open) 2. BETHSY - Tests 6.9a, 6.9c (ISP-38) - CSNI Report 2000(5), NUREG/IA-0188, NUREG/IA-0187 3. PKL - Tests E3.1, F2.2, F2.1 - NUREG/IA-256, NUREG/IA-257	A	
VR-109		val	Turbine Trip	09/31/2016	BWR	PT	1. Peach Bottom Unit 2 Turbine Trip Tests - NEA/NSC/2001-1, NEA/NSC/2004-21, NEA/NSC/2006-23, NEA/NSC/2010-11 - RETRAN-3D (code manual Vol. 4) 2. Cofrentes NPP turbine trip transient - NUREG/IA-0120, RETRAN-3D 3. Santa Maria De Garoña NPP turbine trip transient - NUREG/IA-0226	A	
VR-110		val	BWR core stability	09/31/2016	BWR	PT	Peach Bottom Unit 2 Stability Tests - RETRAN 3D (code manual Vol 4)	P/A	

VR-111		val	BWR start-up tests	09/31/2016	BWR	PT	Grand Gulf Startup Transient Tests - No publicly available data or reference	N/A	
VR-112		val	Recirculation pump trip	09/31/2016	BWR	PT	1. Santa Maria De Garoña NPP - Single Recirculation Pump Trip Transient (NUREG/IA-0193) 2. BWR-5 - One recirculation pump trip (RETRAN-3D)	A	
VR-113		val	Feedwater Pump Trip	09/31/2016	BWR	PT	Cofrentes NPP - One Feedwater Pump Trip Transient (NUREG/IA-0068) - Feedwater pump trip (RETRAN-3D)	A	
VR-114		val	MSIV closure	09/31/2016	BWR	PT	1. Laguna Verde NPP (RETRAN-3D) 2. BWR-5 (RETRAN-3D) - single MSIV closure - closure of all MSIVs 3. Santa Maria De Garoña NPP - MSIV Full Closure (NUREG/IA-0122)	A	
VR-115		val	Other BWR plant tests	09/31/2016	BWR	PT	RETRAN-3D (code manual Vol. 4) assessed its capability based on the following test data from various plants [EPRI 3002003110 (2014)].	P/A	
VR-116		val	Loss of load	09/31/2016	PWR	PT	DOEL-4 NPP - Manual Loss of Load Test of November 23, 1985 - (NUREG/IA-0043)	A	
VR-117		val	PWR startup tests	09/31/2016	PWR	PT	1. Arkansas Nuclear One – Unit 2 (4 transient tests) - No publicly available data or reference 2. DOEL 2 NPP - Pressurizer spray tests (NUREG/IA-0020)	P/A	
VR-118		val	Turbine Trip	09/31/2016	PWR	PT	Arkansas Nuclear One – Unit 2 - No publicly available data or reference Vandellos II NPP (NUREG/IA-0108)	P/A	

VR-119		val	Loss of Off-Site Power	09/31/2016	PWR	PT	1. Arkansas Nuclear One – Unit 2 - No publicly available data or reference 2. McGuire 1 nuclear station - No publicly available data or reference 3. Kori 1 NPP (RETRAN-3D, NUREG/IA-0030)	P/A	
VR-120		val	SGTR	09/31/2016	PWR	PT	1. Prairie Island Unit 1 - No publicly available data or reference 2. R. E. Ginna NPP - Ginna 1/25/1982 steam generator tube rupture accident (NUREG-0909) 3. DOEL 2 NPP - NUREG/IA-0008, ISP-20 (CSNI report No. 154)	P/A	
VR-121		val	Load rejection	09/31/2016	PWR	PT	Comanche Peak Unit 1 (RETRAN-3D) Cofrentes NPP (RETRAN-3D) Laguna Verde NPP (RETRAN-3D) BWR-5 (load rejection with bypass) (RETRAN-3D) Kori 4 (RETRAN-3D) Vandellos II NPP (NUREG/IA-0107, NUREG/IA-0109)	A	
VR-122		val	Multiple failures	09/31/2016	PWR	PT	Kori 2 NPP (RETRAN-3D)	A	
VR-123		val	SBO	09/31/2016	PWR	PT	Asco NPP Blackout Transients (NUREG/IA-0119)	A	
VR-124		val	Feedwater line isolation	09/31/2016	PWR	PT	Ringhals 4 NPP (NUREG/IA-0038)	A	
VR-125		val	Steam Line Isolation Valve Closure	09/31/2016	PWR	PT	Ringhals 2 NPP (NUREG/IA-0041)	A	
VR-126		val	RCP trip	09/31/2016	PWR	PT	Almaraz I NPP (NUREG/IA-0233) Vandellos II NPP (NUREG/IA-0243)	A	
VR-127		val	Reactor trip	09/31/2016	PWR	PT	Tihange-2 NPP (NUREG/IA-0044) DOEL 4 NPP (NUREG/IA-0051)	A	

VR-128		val	Natural circulation	09/31/2016	PWR	PT	- Borssele NPP (NUREG/IA-0091) - Yong-Guang Unit 2 NPP (NUREG/IA-0125) - KNU-1 loss of offsite power (NUREG/IA-0030)	A	
VR-129		val	Load trip	09/31/2016	PWR	PT	Yong-Gwang Unit 2 - Net Load Trip Test Data (NUREG/IA-0092)	A	
VR-130		val	Other PWR plant tests	09/31/2016	PWR	PT	1. Kori Unit 3 - Inadvertent Safety Injection Incident (NUREG/IA-0105) 2. Vandellos II - Main Feedwater Turbopump Trip (NUREG/IA-0110) 3. Asco NPP - Pressurizer Spray Valve Faulty Opening Transient (NUREG/IA-0121) 4. Jose Cabrera Nuclear Station - Pressurizer Spray Valve Inadverted Fully Opening Transient and Recovery by Natural Circulation (NUREG/IA-0124) 5. Maanshan PWR NPP Transient Data (NUREG/IA-0241)	A	
VR-131		val	MSIV closure	09/31/2016	PWR	PT	Vandellos-II NPP (NUREG/IA-0197)	A	
VR-132		val	Loss of decay heat removal	09/31/2016	PWR	PT	1. Vogtle Unit 1 (NUREG-1410) 2. Diablo Canyon Unit 2 (NUREG-1269)	A	
VR-133		val	Turbine Trip	09/31/2016	PWR (B&W)	PT	Oconee Unit 3 - Oconee Unit 3 turbine trip with feedwater overfeed transient of 3/14/1980 (No publicly available data or reference)	N/A	
VR-134		val	SBLOCA	09/31/2016	PWR (B&W)	PT	TMI-2 Accident - No publicly available data or reference	N/A	

VR-135		val	Stuck-open PORV transient	09/31/2016	PWR (B&W)	PT	Crystal River-3 NPP - Crystal River Unit 3 stuck-open PORV transient of 2/26/1980 - No publicly available data or reference	N/A	
VR-136		val	Loss of Off-Site Power	09/31/2016	PWR (B&W)	PT	Arkansas Nuclear One – Unit 1 - No publicly available data or reference	N/A	
VR-137		val	LOAF	09/31/2016	PWR (B&W)	PT	Davis-Besse NPP - Davis-Besse loss of all feedwater event of 6/9/1985 - No publicly available data or reference	N/A	
VR-138		val	Loss of ICS	09/31/2016	PWR (B&W)	PT	Rancho-Seco NPP - Rancho-Seco loss of integrated control system (ICS) power event of 12/26/1985 - No publicly available data or reference	N/A	
VR-139		val	RCP trip	09/31/2016	PWR (B&W)	PT	Oconee Unit 1 and Crystal River Unit 3 - Trip of all RCP tests - No publicly available data or reference	N/A	
VR-140		val	Vessel Mixing (Multi-D TH effects)	09/31/2016	PWR (B&W)	PT	Oconee B&W PWR - Testing of thermal mixing in the lower plenum and core at Oconee Unit 1 - No publicly available data or reference	N/A	