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RELAP-7 Code Assessment Plan and Requirement Traceability Matrix

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Junsoo Yoo, Yong-joon Choi, Curtis L. Smith

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Junsoo Yoo[*], Yong-joon Choi, Curtis L. Smith
Idaho National Laboratory
2525 North Fremont Ave., P. O. Box 3860, Idaho Falls, ID 83415-3870, USA
junsoo.yoo@inl.gov

ABSTRACT

The RELAP-7, a safety analysis code for nuclear reactor system, is under development at Idaho National Laboratory (INL). Overall, the code development is directed towards leveraging the advancements in computer science technology, numerical solution methods and physical models over the last decades. Recently, INL has also been putting an effort to establish the code assessment plan, which aims to ensure an improved final product quality through the RELAP-7 development process. The ultimate goal of this plan is to propose a suitable way to systematically assess the wide range of software requirements for RELAP-7, including the software design, user interface, and technical requirements, etc. To this end, we first survey the literature (i.e., international/domestic reports, research articles) addressing the desirable features generally required for advanced nuclear system safety analysis codes. In addition, the V&V (verification and validation) efforts as well as the legacy issues of several recently-developed codes (e.g., RELAP5-3D, TRACE V5.0) are investigated. Lastly, this paper outlines the Requirement Traceability Matrix (RTM) for RELAP-7 which can be used to systematically evaluate and identify the code development process and its present capability.

KEYWORDS

RELAP-7, Code Assessment Plan, Requirement Traceability Matrix (RTM)

Acronyms

BDF2	Second-order backward difference scheme
BWR	Boiling Water Reactor
CS	Coolant System
DOE	U.S. Department of Energy
EPRI	Electric Power Research Institute
FSI	Fluid-Structure Interaction
I&C	Instrumentation and Control
INL	Idaho National Laboratory
JFNK	Jacobian-free Newton-Krylov
LWRS	Light Water Reactor Sustainability
MOOSE	Multi-Physics Objective-Oriented Simulation Environment
NPP	Nuclear Power Plant
PCICE	Pressure-Corrected Implicit Continuous-fluid Eulerian algorithm
PWR	Pressurized Water Reactor
RTM	Requirement Traceability Matrix
SETS	Stability Enhancing Two-Step method
SQA	Software Quality Assurance
SVVP	Software Verification and Validation Plan
TH	Thermal-Hydraulic
UA	Uncertainty Analysis
UI	User Interface

V&V Verification and Validation

1. INTRODUCTION

RELAP-7 is a next-generation nuclear reactor system safety analysis code that has been developed by Idaho National Laboratory (INL) under LWRS (Light Water Reactor Sustainability) program of DOE. In order to achieve the full potential of RELAP-7 beyond traditional thermal-hydraulic system analysis codes, the RELAP-7 development has been directed toward taking full advantage of modern engineering and computational techniques while incorporating the latest needs of nuclear industry [1]. As a result, the advances in knowledge/experience concerning the nuclear reactor system safety analysis (e.g., two-phase flow modeling, numerical methods, closure models), computing power (e.g., parallelizing capability), mesh management techniques, etc. are incorporated into RELAP-7 framework [2], which subsequently allows us to expect an improved prediction capability and efficiency of the code for a wide range of applications for nuclear reactor safety analysis.

Meanwhile, besides the code development work, INL has been putting an effort to establish the code assessment plan for RELAP-7 which is to ensure an improved work product quality through the code development process. Specifically, a research project “Software Verification and Validation Plan (SVVP)” has been launched and conducted in INL since 2012 as a part of software quality assurance program. According to INL’s internal document PLN-4215 (2012), the SVVP aims to define all necessary actions to determine whether the requirements of the software are completed properly at each development phase and the final product complies with the specified requirements. This implies that the first step to attain the goal of SVVP for RELAP-7 is to understand and identify the various aspects of requirements for the code (RELAP-7) which is a main subject of this paper.

The basic principle that applies to the code assessment for RELAP-7 is schematically described in Figure 1. Similar to that of RELAP-7 development, the code assessment plan is established by taking into account the knowledge accumulated over the last decades especially for both code qualification methods [3] and experience-based findings/demands from nuclear industry [1]. It is noted that there have been comprehensive international efforts to identify the general needs for the next-generation system analysis code, necessary V&V efforts, and relevant safety issues to be addressed [4-6], and they are reviewed in this paper to make the RELAP-7 assessment plan be in line with those efforts. Also, the code V&V works performed during the development phase of recently-developed reactor system analysis codes (i.e., RELAP5-3D [7], TRACE v5.0 [8-11]) are investigated which will also help define the action items for the RELAP-7 code assessment. Combining all these efforts, this paper outlines the requirement traceability matrix (RTM) which can be used to systematically evaluate and identify the RELAP-7 development process and its present capability.

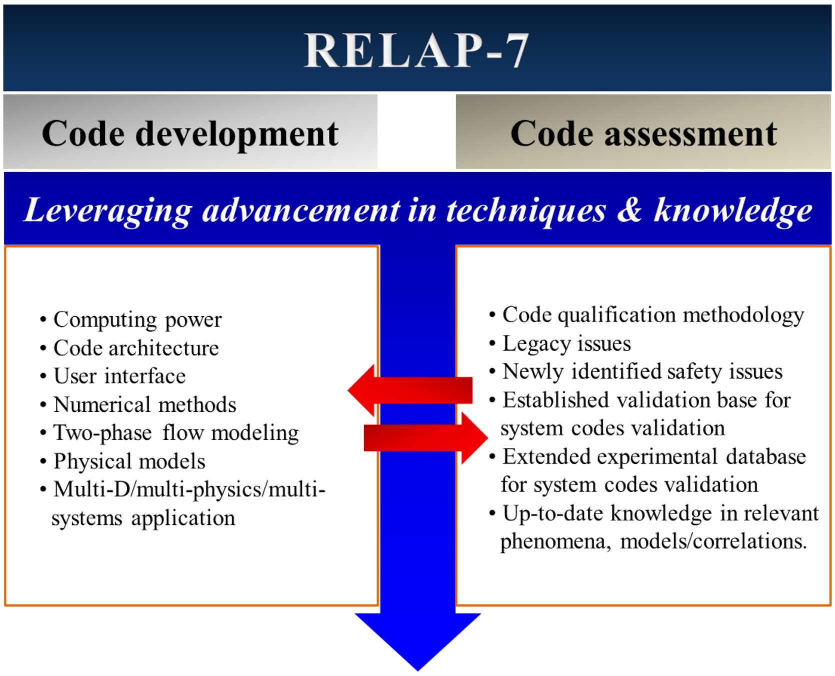

Figure 1. RELAP-7 code development and assessment strategy

2. RELAP-7 CODE ASSESSMENT PROCEDURE

Figure 2 shows the overall assessment (or qualification) procedure of RELAP-7 from its development to release. The target capability of RELAP-7 was initially defined based on the specific industry requirements delivered by EPRI [1] which reports the perspectives of the largest system code user community like NPP owners and operators. In their report, a special emphasis is put on the necessity of an advanced system analysis code that can address the design and regulatory issues over the "full (or extended)" plant time operation such as NPP life extension and power level uprates.

For the code assessment, the following two stages are generally required as discussed by Petruzzi and D'auria [3]: (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) for which the databases should be independent from those used in the code development process. In particular, the nodalization strategy [12, 13], code application procedure [14], user qualification [13, 14], and evaluation of code prediction accuracy [14, 15] 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 [3, 16].

The aforementioned code assessment procedure will be followed in general for RELAP-7 as well. Of these, this paper focuses particularly on identifying the specific requirements needed for the "internal" code assessment of RELAP-7 as illustrated in Figure 2. As a first step to this end, both domestic (U.S.) and international efforts for decades to identify the desired characteristics of next-generation nuclear system analysis code are discussed in the following section.

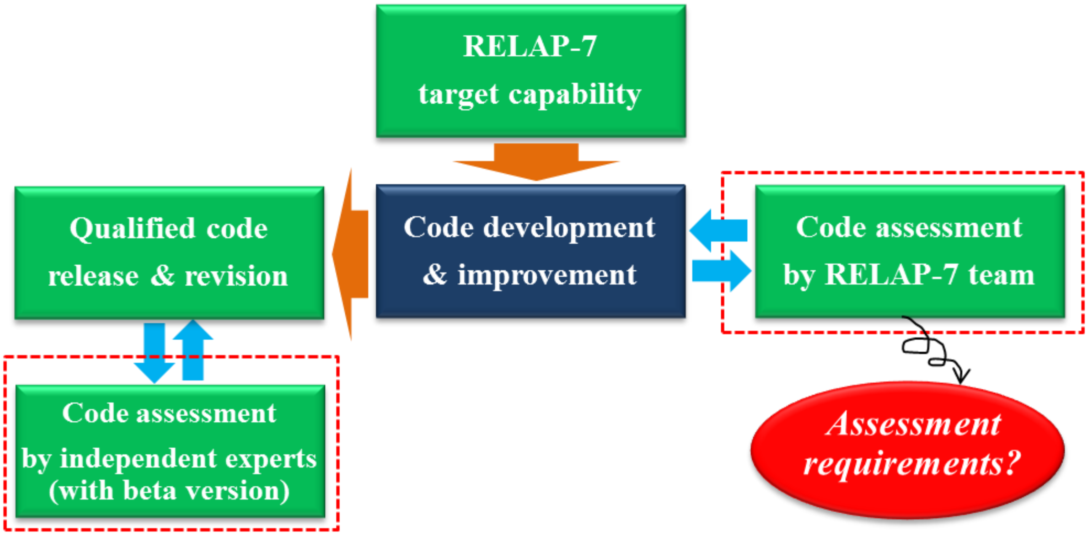

Figure 2. RELAP-7 code development and assessment procedure

3. DESIRED CHARACTERISTICS OF BEST-ESTIMATE SYSTEM ANALYSIS CODE AND RELAP-7 FEATURES

3.1. Desired Characteristics of Next-Generation Best-Estimate System Analysis Code

The desired features of next-generation best-estimate system analysis code, legacy issues which remains unresolved in the existing codes, and specific improvement items for the future code have been discussed in several reports and research articles [1, 4, 17-19]. Table 1 summarizes the various aspects of such demands for the next-generation system analysis code which are primarily derived based on the lessons learned from previous experience of system code developers and users. 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 [20, 21].

- 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, 4, 17]. In particular, a code with capability of incorporating uncertainty quantification process into an integral part of simulation process has recently been paid increasing attention [19, 22].

Table 1. Specific demands for next-generation nuclear reactor system analysis code [1, 4, 17]

	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)

3.2. RELAP-7 Features

RELAP-7 code is designed to eventually retain a variety of desired features needed for the next-generation system analysis code discussed in section 3.1. The code has been developed based on the INL's modern scientific software development framework, MOOSE (Multi-Physics Object Oriented Simulation Environment) [23]. This provides an advantage for RELAP-7 in the sense that the code developers can focus more on the physics and user interface capability because numerical solvers, mesh management for parallel computation, and other open source software packages can be simply imported within the MOOSE framework [2]. Also, MOOSE provides environment enabling RELAP-7 to easily couple with other MOOSE-based softwares (see Figure 3) addressing various physical processes within reactor system (*e.g.*, 3-D transient neutron transport, 3-D transient fuel performance, component aging). As a result, as shown in Figure 3, multi-physics, multi-scale, and multi-D simulation can be achieved more easily and efficiently with parallel computing capability which is one of the key features of RELAP-7 upon traditional system analysis codes such as RELAP5.

In Table 2, compared are the specific features of RELAP-7, including governing theory, numerical methods, non-linear solvers, and physical assumptions, with those of existing system analysis codes. RELAP-7 is originally designed to simulate both single- and two-phase flow for all-speed and all-fluids in conjunction with heat transfer and reactor kinetics models in it. At present, however, the RELAP-7 development focuses on demonstrating the simulation capability of light water reactors (LWR), and thus the two-phase flow model in RELAP-7 is naturally of particular interest at this stage.

RELAP-7 employs a 7-equation two-phase flow model in which each phase (*i.e.*, liquid, vapor) is treated as being compressible. Compared to 6-equation model that most of the existing TH system analysis codes rely on, a volume fraction evolution equation is additionally introduced to represent the dynamic flow regime transition. Also, the 7-equation model in RELAP-7 does not take the pressure equilibrium assumption (*i.e.*, $p_{liquid} \neq p_{vapor}$) as opposed to the classical 6-equation model (see Table 2), which allows the governing equations to mathematically maintain the well-posed nature and exhibit hyperbolicity. It is noted such well-posedness also enables us to strictly verify the governing equations of RELAP-7 through mesh sensitivity tests like any modern CFD models.

To achieve advanced numerical solutions using RELAP-7, modern solving methods and numerical schemes are introduced. Specifically, higher-order numerical schemes (*i.e.*, second-order accurate schemes in both space and time) are employed in conjunction with implicit time integration. And the code is designed to solve multi-physics involving the fluids flow, heat conduction, conjugate heat transfer, and reactor kinetics in a fully coupled fashion which is opposed to the operator-splitting method (also known as the semi-implicit method) used in the existing system analysis codes. Besides, to efficiently solve the highly non-linear system, JFNK is used as a parallel nonlinear solver through MOOSE framework which supports the effective coupling between physics equation systems (or Kernels) [24]. In addition, RELAP-7 is currently pursuing a three-level time integration approach to achieve an all-time scale capability and thus to handle any systems-level transient (see table 2) [2, 24].

Overall, RELAP-7 has been developed toward achieving a variety of desired features of next-generation system analysis code discussed in section 3.1. Especially, (i) the code architecture that allows efficient coupling with other physics equation systems and high modularity, (ii) the advanced numerical methods/solvers (*e.g.*, higher-order accuracy, implicit time integration), and (iii) the 7-equation formulation for two-phase flow are expected to contribute significantly to overcoming many legacy issues of system analysis codes (*e.g.*, physical issues, numerical issues, modeling issues, etc.) revealed by system code user community [1, 4].

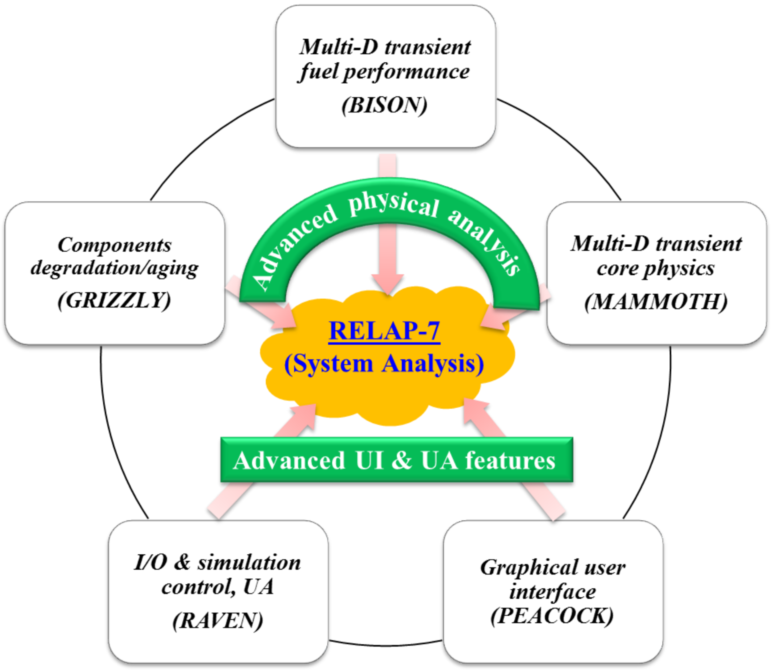

Figure 3. MOOSE-based softwares supporting "advanced" reactor system analysis using RELAP-7

Table 2. Main features of RELAP-7, RELAP5-3D, and TRACE

	RELAP-7	RELAP5-3D	TRACE
Source code program language	C++	FORTRAN	FORTRAN
Two-phase modeling	Two-fluid model, 7 Eq.	Two-fluid model, 6 Eq.	Two-fluid model, 6 Eq.
Mathematical formulation	Compressible, all-speed/all-fluid flow (well-posed, hyperbolic)	Incompressible flow (ill-posed, non-hyperbolic)	Incompressible flow (ill-posed, non-hyperbolic)
Phasic pressures	Pressure non-equilibrium between phases	Pressure equilibrium between phases	Pressure equilibrium between phases
Interfacial pressure		Equal to phasic pressures (except for stratified flow)	-
Reynolds stresses	Neglected[(1)]	Neglected[(1)]	Neglected[(1)]
Time integration scheme	Fully implicit[(2)] (BDF2, backward Euler)	Semi-implicit, Nearly-implicit	Semi-implicit, SETS
Non-linear solver	JFNK	Newton's method	Newton's method
Multi-D analysis capability	Yes[(3)]	Yes	Yes

[(1)] *Turbulence model is not explicitly coupled to the field equations.*
[(2)] *PCICE (semi-implicit) and point implicit method is also available in RELAP-7 depending on the time scale of problems simulated (three-level time integration approach).*
[(3)] *RELAP-7 code itself mainly consists of 0-D and 1-D components, but the multi-D capabilities can be enhanced through the coupling of other MOOSE-based codes.*

4. V&V EFFORTS TO VALIDATE THE BEST-ESTIMATE TH SYSTEM ANALYSIS CODES

The verification and validation (V&V) of best-estimate TH 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 for decades to support the activities. Of these, the most well-known are International Standard Problems (ISP) [25] and CSNI Code Validation Matrices (CCVM) [6, 26, 27], both of which have been led by OECD/NEA CSNI (Committee for Safety of Nuclear Installation). The

ISP and CCVM both are the consequences of efforts to systematically collect the best sets of qualified experimental data for code validation as well as for code assessment with respect to uncertainty quantification. Recently, efforts have continued in the similar context 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 [28]. In addition, new experimental data and thus new insight/safety issues have been continuously revealed [29, 30] all of which are obviously precious to demonstrate and improve the system code capability. It is noted, however, many of these efforts are mainly to support the "independent" code assessment after completing the code development phase.

As discussed in section 2, of our current interest is to develop requirements and subsequently to establish plan for the "internal" code assessment of RELAP-7. In this regards, we first review the validation bases adopted for the code V&V of recently-developed system codes, *i.e.*, RELAP5-3D and TRACE during their development phase (*i.e.*, validation bases for "internal" code assessment), and the results are summarized in Table 3 (fundamental tests), Table 4 (separate effect tests), and Table 5 (integral effect tests). These will provide guidance to establishing the RELAP-7 code assessment plan especially in terms of defining the code V&V needs that will be discussed in detail in Section 5. Also, it is noted that the test problems shown in Tables 3-5 can become a useful basis for code-to-code comparisons during the RELAP-7 development phase because the modeling process for each test problem and the results are relatively well-documented [7, 9-11].

The fundamental tests shown in Table 3 are to confirm the basic performance of codes with relatively simple problems, some of which are also used for code verification by comparing the results with analytic solutions when experimental data are unavailable. Table 4 shows the separate effect tests for the codes validation which are to ascertain that the specific phenomenon relevant to LWR safety is simulated appropriately by the code. In Table 5, presented are the code validation tests against the data obtained from the scaled integral test loops; these are to demonstrate the code capability simulating the overall behavior of reactor system during the course of LOCAs/non-LOCAs.

It is noted that 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 [31]. These cover the representative phenomena occurring during the LOCA/non-LOCA sequences in nuclear reactor systems (LWR) and many of them are considered essential to validate the system analysis codes. Thus, comparing the list with the validation tasks listed in Tables 3-5 can be one of the simple ways to see how well the codes capability was tested and/or demonstrated through the development process, which will also help us to identify further needs to be supplemented for RELAP-7 code validation. From this, it is revealed that some of basic two-phase flow phenomena and system-based phenomena are not explicitly covered by the validation tests shown in Tables 3-5, the list of which is given in Table 6.

Table 3. Fundamental tests for internal code assessment of RELAP5-3D and TRACE v5.0 [7, 9]

Test physics	Test title	Code model/predicted parameters involved	RELAP5-3D	TRACE v5.0
gravity head effect	Water faucet	Hydro numerics, gravity	O	
	Water over steam (1D/3D)	Gravity, liquid level	O	
Level tracking	Fill-drain	Mixture level tracking, gravity head	O	O
	Bubbling steam through liquid	Entrainment, mixture level tracking	O	O
	Single tube flooding (vertical)	Flooding, counter-current flow (CCFL)		O
Flow oscillation	Manometer	Inertia and gravity effects, mixture level tracking, non-condensables	O	O
	Gravity wave (1D, 3/D)	Conter-current flow, wave propagation	O	
Water packing	Pryor pressure comparison	Model for mitigating water packing	O	
Two-phase convective heat transfer	Christensen test	Void fraction, interfacial mass/heat transfer, wall boiling model	O	
Two-phase flow phase distribution (horizontal)	TPTF horizontal flow tests (steam-water co-current flow behavior and its stratification in horizontal channels)	Void fraction		O
Two-phase flow phase distribution (vertical)	Vertical two-phase flow tests (adiabatic air-water upflow in a simple vertical pipe)	void fraction		O
Wall friction, pressure drop	Single-phase wall friction	Wall friction model		O
	Two-phase wall friction	Wall friction model		O
Interfacial shear	CISE adiabatic tube	Interfacial shear model		O
Heat conduction	Conduction enclosure (steady-state & transient)	Heat conduction eq. (1D/2D)	O	O
Multi-D hydrodynamics	Pure radial symmetric flow (3-D)	Radial momentum flux terms in MULTID component	O	
	Rigid body rotation (3-D)	Azimuthal momentum flux terms in MULTID component	O	
	R-theta symmetric flow (3-D)	Radial and azimuthal terms in momentum eq.	O	
Neutronics, core power	Core power	Fission product and actinide decay heat model	O	
	Point kinectics ramp	Point reactor kinetics model (prompt/delayed neutron fission power model)	O	
Metal-water reaction	Cladding oxitation	Metal-water reaction model	O	

Table 4. Separate effect tests for internal code assessment of RELAP5-3D and TRACE v5.0 [7, 10]

Major phenomena involved	Test title	Test geometry	Test type	Code model involved	RELAP5-3D (test No.)	TRACE v5.0 (test No.)
Critical flow	Edwards-O'Brien Blowdown Test	horizontal straight pipe	transient	choked flow model (subcooled liquid, saturated two-phase mixture)	○	
	SUPER MOBY DICK	vertical divergent nozzle, abrupt expansion	steady state			○ (10 tests)
	MOBY DICK	vertical divergent nozzle	steady state		○ (3141)	○ (10 tests - 5 for CFT, 5 for non-CFT)
	MARVIKEN	vertical pipes (full scale BWR)	transient		○ (21, 22, 24, JIT-11)	○ (6 tests)
Single-/two-phase convective heat transfer	Christensen test	vertical heated tube	steady state	void fraction, interfacial mass/heat transfer, wall heat flux	○ (15)	
	RBHT steam cooling tests	7x7 rod bundle (full length portion of PWR)	steady state	single-phase (steam) convective heat transfer		○ (3173A, 3216D, 3205A, 3216A, 3216G, 3205G, 3214A)
	FRIGG Tests	36 rods arranged in concentric rings	steady state	void fraction, interfacial friction, convective heat transfer		○ (313001'20, 24, 27, 30, 34, 37, 40, 43, 56, 60)
Two-phase mixture level swell, interfacial friction	ORNL THTF level swell tests	8x8 rod bundle	steady state	mixture level swell, void fraction, steam temp., interfacial friction, wall heat transfer	○ (3.09.10I)	○ (3.09.10I, J, K ,L, M, N, AA, BB, CC, DD, EE, FF)
	RBHT uncovery tests (steady state)	7x7 rod bundle (full length portion of PWR)	steady state	mixture level swell, void fraction, interfacial friction		○ (1560, 1566, 1570, 1572 1582, 1637, 1648, 1651, 1659)
	RBHT uncovery tests (transient)	7x7 rod bundle (full length portion of PWR)	transient	mixture level swell, void fraction, interfacial friction, wall heat transfer		○ (1690)
	Wilson Bubble Rise Tests	0.91 m diameter pressure vessel	steady state	mixture level swell, void fraction, interfacial friction		○ (60 tests)
Vapor generation, two-phase mixture level during blowdown	GE Level Swell Test (1ft)	vertical vessel (1ft diameter)	transient	vapor generation during blowdown, interfacial friction, mixture level	○ (1004-3)	○ (1004-3)
	GE Level Swell Test (4ft)	vertical vessel (4ft diameter)	transient		○ (5801-15)	○ (5702-16)
Critical heat flux (CHF), film boiling (FB) heat transfer during blowdown	Bennett Heated Tube Tests	vertical heated tube	steady state	CHF	○ (5358, 5294, 5394)	
	ORNL THTF steady state tests	8x8 rod bundle	steady state	CHF, film boiling	○ (3.07.9B, 3.07.9N, 3.07.9W)	○ (3.07.9B, 3.07.9H, 3.07.9N, 3.07.9W)
	ORNL THTF transient tests (blowdown)	8x8 rod bundle	transient	Heat transfer during a blowdown (e.g., CHF, film boiling)		○ (3.03.6AR, 3.06.6B, 3.08.6C)
	Royal Institute of Technology Tube Test	vertical heated tube	steady state	CHF	○ (261)	
Reflood	FLECHT SEASET	17x17 rod bundle	transient	reflood, post-CHF heat transfer, interfacial friction, entrainment	○ (31504, 31701)	○ (31108, 31203, 31302, 31504, 31701, 31805, 32013, 32114)
	GOTA	1/676-scaled BWR ECCS with 8x8 rod	transient	reflood (Run 42), radiation heat transfer (Run 27)		○ (27, 42)
	RBHT reflood tests	7x7 rod bundle (full length portion of PWR)	transient	reflood, post-CHF heat transfer, interfacial friction, entrainment, steam temp., spacer grid effects		○ (1096, 1108, 1170, 1196, 1285, 1383)
CCFL, ECC Bypass	Dukler-Smith Air-Water Flooding	vertical single pipe	transient (time-averaging data for	CCFL (Wallis)	○	
	UPTF Downcomer CCFL Test	UPTF lower plenum (full scale)	quasi steady state	CCFL	○ (6-run131)	○ (5, 6, 7, 21)
	Bankoff CCFL Tests	vertical perforated plates of various	quasi steady state	CCFL		○
Condensation	UCB-Kuhn Condensation Tests	vertical tube (downward steam/gas	steady state	condensation , non-condensible gas effect		○ (30 tests)
	Dehbi-MIT Condensation Tests	vertical vessel with copper cylinder tube inside	steady state	condensation (free convection), non-condensible gas effect		○ (21 tests)
	University of Wisconsin Condensation Tests	rectangular channel (0-90° inclination angle)	steady state	condensation (forced convection), non-condensible gas and geometry effect		○ (6 tests for vertical downward flow)
Pressurizer behavior	MIT Pressurizer Test	cylindrical vertical tanks	transient	steam condensation on the wall, interfacial mass/heat transfer	○ (ST4)	○ (ST4, Insurge-Outsurge)
	Neptunus Test	1/40th scaled pressurizer (pzr)	transient	pzr performance (flow insurges and outsurges with pzr spray operation)	○ (Y05)	
Steam generator physics (e.g., steam binding)	Westinghouse Model Boiler No. 2 (MB-2) Test	Westinghouse Model F steam generator (0.8% power scaled)	transient	steady-state steam generator behavior	○ (1712)	○ (4 tests including 2013)
	FLECHT-SEASET Steam Generator Tests	FLECHT separate effect test Phase B steam generator (32 full length U-tubes inside)	transient (PWR LOCA)			○ (22701, 21806, 23605, 23402)
Pump behavior	Darlington Nuclear Generation Station Pump Test	CANDU full scale RCP	steady state	centrifugal pump model	○	
	GE 1/6-Scale jet pump	1/6-scaled GE pump	steady state	jet pump model	○	
Accumulator injection	LOFT Experiment L3-1	50 MWt power/volume-scaled nuclear reactor		accumulator model during SBLOCA	○	

Table 5. Integral effect tests for internal code assessment of RELAP5-3D and TRACE v5.0 [7, 11]

Experimental facility	Test ID	Test scenario	RELAP5-3D	TRACE v5.0
LOFT	LOFT L2-5	LBLOCA	O	O
	LOFT L2-6	LBLOCA		O
	LP-LB 01	LBLOCA		O
	L3-1	SBLOCA		O
	L3-7	SBLOCA	O	O
LSTF (ROSA IV Large Scale Test Facility)	LSTF SBCL 01	SBLOCA		O
	LSTF SBCL 05	SBLOCA		O
	LSTF SBCL 14	SBLOCA		O
	LSTF SBCL 15	SBLOCA		O
	LSTF SBCL 16	SBLOCA		O
	LSTF SBCL 18	SBLOCA	O	O
BETHSY	6.2 TC	SBLOCA		O
	9.1 b	SBLOCA		O
CCTF (Cylindrical Core Test Facility)	C2-4 (Run 62) C2-5 (Run 63) C2-6 (Run 64) C2-8 (Run 67) C2-1 (Run 55) C2-AA2 (Run 58) C2-12 (Run 71)	Refill, reflood, ECC behavior during PWR LOCA		O
TLTA (Two Loop Test Apparatus)	Test 6425 Run 2 Test 6424 Run 1	Critical flow, CCFL, core heatup, ECC injection during BWR LOCA		O
SCTF (Slab Core Test Facility)	Runs 604, 605, 606, 607, 611, 621, and 622	Reflood (gravity reflood, forced reflood)		O
LOBI	LOBI TEST A1-04R	LBLOCA	O	
SEMISCALE Mod-2A	S-NC-1	Natural circulation (single-phase steady state)	O	
	S-NC-2	Natural circulation (single-/two-phase, reflux steady state modes)	O	O
	S-NC-3	Natural circulation (two-phase behavior)	O	O
	S-NC-10	Natural circulation (single-/two-phase behavior)	O	
FIST (Full Integral Simulation Test)	Test 6SB2C Test 6SB1	Recirculation line break (BWR)		O
SSTF (Steam Sector Test Facility)	Test EA 3.1 Run 111 Test EA 3.3-1 Run 119	CCFL, ECCS injection mixing in upper/lower plenum, parallel channel phenomena (BWR)		O

Table 6. Relevant two-phase flow phenomena to LWR safety that are not explicitly covered by the validation tasks of 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

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

Combining all the efforts described above, the requirement traceability matrix (RTM) is established. The RTM is to systematically evaluate and identify the RELAP-7 development process by associating the specific requirements with the work product. In RTM, the unique identifiers are provided for each requirement to indicate the progress/completeness of the development. Also, the matrices provide wide range of tangible and traceable items of different categories that can be examined. The requirements defined in RTM are categorized into three types: (i) general requirements, (ii) specific requirements, and (iii) code V&V requirements.

As shown in Figure 4, the "general requirements" identify the general software capabilities requested on the advanced TH system analysis code such as multi-D and multi-scale simulation capability, etc. An example of RTM for general requirements is given in Table 7. The "specific requirements" define the technical aspect of requirements for RELAP-7 especially by focusing on the legacy issues of existing system codes. The "specific requirements" are classified into 5 categories as shown in Figure 4. The "code V&V requirements" provide the necessary V&V tasks for RELAP-7, in which several test items to check the performance of unique numerical schemes employed in RELAP-7 are included besides the traditional V&V tasks applied to the previous TH system codes,. Also, the validation tasks for SET and IET are established by considering those adopted for the existing system codes as well as by accounting for the missing items revealed through this study (see section 4). The details for the first version of RTM are given in Ref. [32], but it is still continuously being updated.

6. SUMMARY AND CONCLUSIONS

This paper aims to show the INL's activity to establish Requirement Traceability Matrix (RTM) which can be used to systematically evaluate and identify the RELAP-7 code development process. The RTM provides the wide range of tangible items in different categories needed for "internal" code assessment for RELAP-7. To identify the fundamental requirements that should be covered during the code development process, international efforts are first reviewed to understand the general needs for the advanced nuclear system analysis codes, necessary V&V efforts, legacy issues, and relevant LWR safety issues, etc. Also, the specific V&V efforts made during the development phase of RELAP5-3D and TRACE are discussed with missing tasks that need to be supplemented for RELAP-7 code assessment. The RTM established through this work will play an important role in providing the software quality checkpoint for RELAP-7 during the code development process.

DISCLAIMER

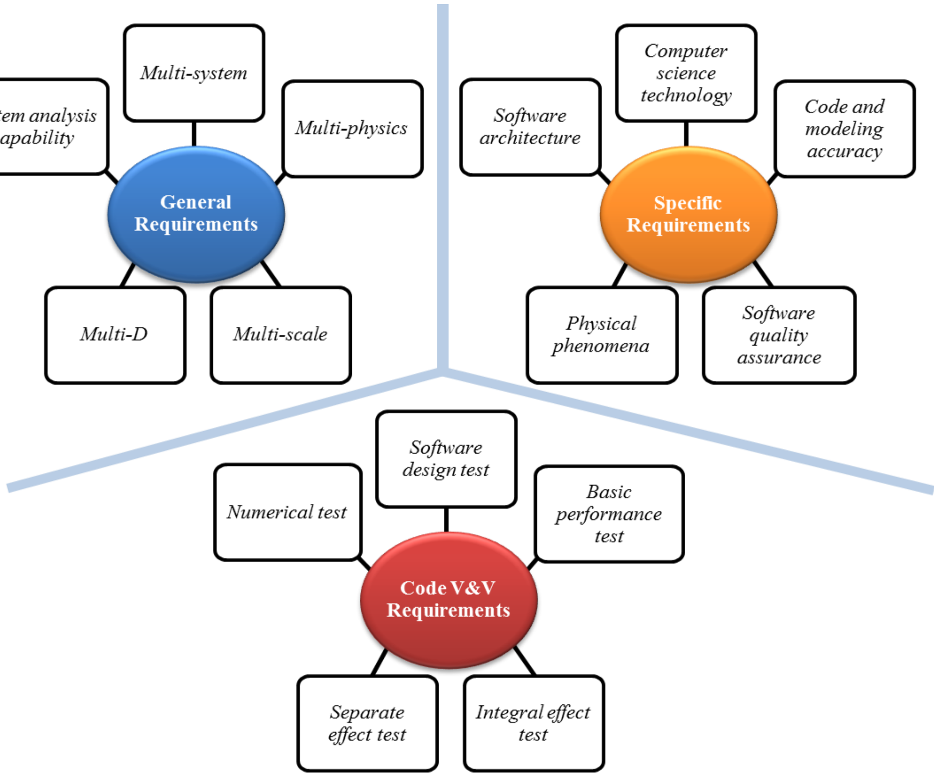

Figure 4. Three aspects of requirements for RELAP-7 code assessment

Table 7. An example of RTM for general requirements

Req #	Category	Requirement Specification	Status
GR-1	Multi-systems & Multi-components	Simulate various LWR designs including PWR and BWR	Test-45 for BWR (HEM model) Test-152 for PWR (TMI loop)
GR-2		Simulate various PWR manufacturers' designs such as Westinghouse, Combustion Engineering, and Babcock&Wilcox	PWR cores are tested but not compared for different manufactures
GR-3		Simulate various containment design such as high and low backpressure	Not tested
GR-4		Simulate reactor designs with non-water coolants	Not tested

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. Petruzzi, A. and F. D'Auria, *Thermal-Hydraulic System Codes in Nulcear Reactor Safety and Qualification Procedures*. Science and Technology of Nuclear Installations, 2008.
4. OECD/CSNI, *Proceedings of the OECD/CSNI workshop on transient thermal-hydraulic and neutronic codes requirements. NEA/CSNI/R(97)4*. 1996.
5. Aksan, N., D. Bessette, and e.a. I. Brittain, *Code validation matrix of thermo-hydraulic codes for LWR LOCA and transients*. 1987: Paris, France.
6. Aksan, N., et al., *Separate effects test matrix for thermal-hydraulic code validation: phenmena characterisation and selection of facilities and tests*. 1993.
7. *RELAP5-3D code manual - Volume III: development assessment*. 2015: Idaho National Laboratory, Idaho Falls, Idaho, USA.
8. *TRACE V5.0, Assessment Manual, Main Report*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
9. *TRAVE V5.0, Assessment Manual, Appendix A - Fundamental Validation Cases*. 2005: U. S.

Nuclear Regulatory Commision, Washington, DC, USA.
10. *TRACE V5.0, Assessment Manual, Appendix B - Separate Effects Tests*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
11. *TRACE V5.0, Assessment Manual, Appendix C - Integral Effects Tests*. 2005: U. S. Nuclear Regulatory Commision, Washington, DC, USA.
12. 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.
13. Ashley, R., et al., *Good practices for user effect reduction. NEA/CSNI/R(98)22*. 1998.
14. *IAEA Safety Report Series No. 23 (2002). Accident analysis for nuclear power plants*. IAEA, Vinenna.
15. 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.
16. D'Auria, F. and G. Galassi, *Scaling in nuclear reactor system thermal-hydraulics.* Nuclear Engineering and Design, 2010. **240**(10): p. 3267-3293.
17. *IAEA Safety Report Series No. 52 (2008). Best estimate safety analysis for nuclear power plants: uncertainty evaluation*. IAEA, Vienna.
18. 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.
19. 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.
20. 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.
21. 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.
22. 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.
23. Gaston, D., et al., *MOOSE: A parallel computational framework for coupled systems of nonlinear equations.* Nuclear Engineering and Design, 2009. **239**(10): p. 1768-1778.
24. Smith, C., et al., *Light Water Reactor Sustainability Program: Risk-informed Safety Margins Characterization (RISMC) Pathway technical program plan*. 2011, INL: Idaho National Laboratory, Idaho Falls, Idaho, USA.
25. Karwat, I.H., *CSNI international standard problems (ISP)*. 2000.
26. 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.
27. Lewis, M., *State of the art report (SOAR) on Thermalhydraulics of Emergency Core Cooling in Light Water. Reactors*. 1989, OECD-NEA-CSNI report.
28. 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).
29. 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.
30. 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.
31. 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.
32. Smith, C.L., Y.-j. Choi, and J. Yoo, *RELAP-7 Software Verification and Validation Plan*. 2015: Idaho National Laboratory, Idaho Falls, Idaho, USA.