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RELAP5-3D Seminar

Developmental Assessment and Verification and Validation of RELAP5-3D

Ethan M. Lang, Ishita Trivedi, and George L. Mesina

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Outline

- Brief Overview of Developmental Assessment
- Purpose of the conversion of “RELAP5-3D Code Manual Volume III: Developmental Assessment” to a new document titled “RELAP5-3D Code Manual: Verification and Validation”
- Project Scope
- Work Completed
- Results
- Future Work

What is Developmental Assessment?

- Developmental assessment (DA) is the systematic evaluation of a code during its development to ensure it is accurate, reliable, and capable of predicting the behavior of the nuclear systems of interest.
- Performing a developmental assessment should:
  - verify that the code correctly implements the intended mathematical models and algorithms.
  - validate the code's predictions with experimental data.
- Historically, much of the validation of system level thermal hydraulic codes have been left more to engineering judgement, rather than quantitative methods, due to the complexity of the problems, limits on computational resources, and lack of experimental data [1].

[1] Office of Nuclear Regulatory Research. (1987). Guidelines and Procedures for the International Code Assessment and Applications Program. NUREG-1271. USNRC. https://www.nrc.gov/docs/ML0710/ML071000073.pdf

Purpose

- The RELAP5-3D team maintains the code manual, including the DA, so that users can apply the code and the regulator can trust the results.
- Certain issues were identified in the DA manual:
  - Hard to navigate
  - Difficult to keep updated
  - Ambiguity in NRC acceptance criteria
  - References to uncertainties in data not found in the document
  - Not developed to meet NQA-1 standards
- The new verification and validation (V&V) document provides:
  - Addition of traceability and results matrices
  - Online documentation maintains with LaTeX
  - Allows users to add additional V&V cases with ease
  - Moves towards meeting NQA-1 standards

Purpose (cont.)

- Assessment criteria provided in [2] relies on expert judgement and quantitative values, but it is not applied consistently in the DA. The following terminology has been used historically to judge systems TH codes:
  - Excellent Agreement
  - Reasonable Agreement
  - Minimal Agreement
  - Insufficient Agreement
- See right: An example of a conclusion that refers to uncertainty data not contained within the document [3].

[2] Office of Nuclear Regulatory Research. (1993). 2D/3D Program Work Summary Report. NUREG/IA-0126. USNRC. https://www.nrc.gov/docs/ML0625/ML062570376.pdf

[3] Idaho National Laboratory. (2023). RELAP5-3D Code Manual Volume III: Developmental Assessment. INL-MIS-15-36723, Revision 4.5.

4.19.5 Conclusions and Assessment Findings

The overall results from the calculations are in reasonable agreement with the data, with no observed differences between the semi- and nearly-implicit calculations. The primary temperature drop through the U-tube bundle shows reasonable agreement, indicating that the model correctly predicts the amount of energy transferred from the primary to the secondary. Some other calculated steady-state conditions (e.g., narrow range level) lie outside the uncertainty range of the data. The level is particularly sensitive to the separator junction loss coefficients. It is likely that further adjustments to the model could be made to better match the pressure distribution on the secondary side, resulting in initial conditions that are closer to the experiment data.

4.19.6 References

4.19-1. M. Y. Young, et al., *Prototypical Steam Generator Transient Testing Program: Test Plan/Scaling Analysis*, EPRI NP-3494, NUREG/CR-3661, WCAP-10475, September 1984.

4.19-2. O. J. Mendler, K. Takeuchi, and M. Y. Young, *Loss of Feed Flow, Steam Generator Tube Rupture and Steam Line Break Thermohydraulic Experiments: MB-2 Steam Generator Transient Response Test Program*, NUREG/CR-4751, EPRI NP-4786, WCAP-11206, October 1986.

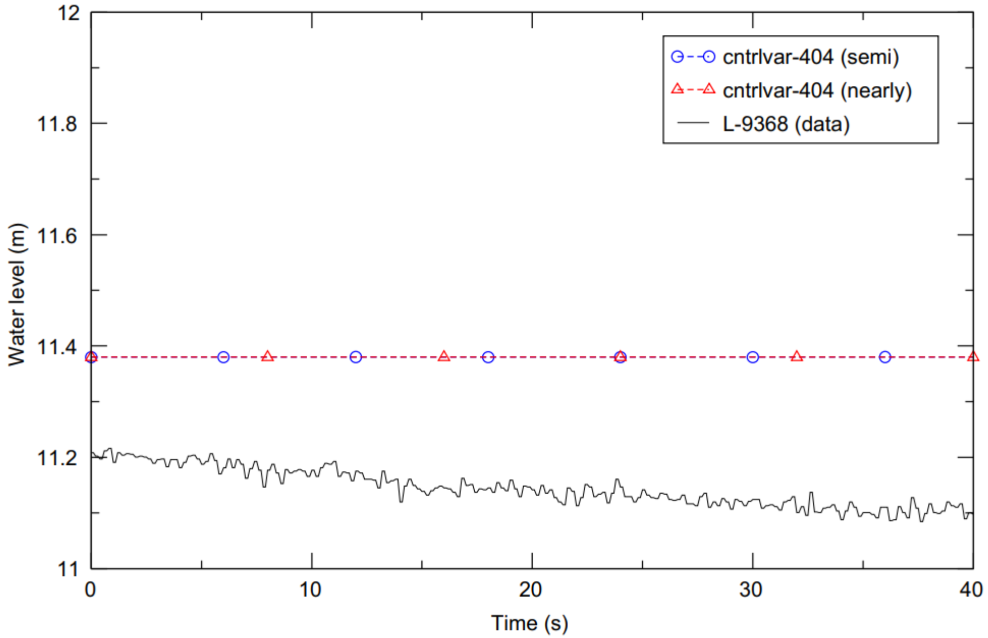


Figure 4.19-9. Measured and calculated narrow range water level during MB-2 steady state Test 1712.

Project Scope

- Update the “RELAP5-3D Code Manual Volume III: Developmental Assessment” to a LaTeX based format for ease of editing, maintaining, and adding to the document.
- Create a new LaTeX based document, “RELAP5-3D Code Manual: Verification and Validation”, derived from the DA
  - Clearly separates verification and validation cases.
  - Include traceability matrices to easily determine which cases are verifying/validating which code models.
  - Provides summary matrices containing test case results.
- Editorial updates to improve user interpretation.
- Online documentation of the manuals with real-time compilation capability.

Updates to the DA

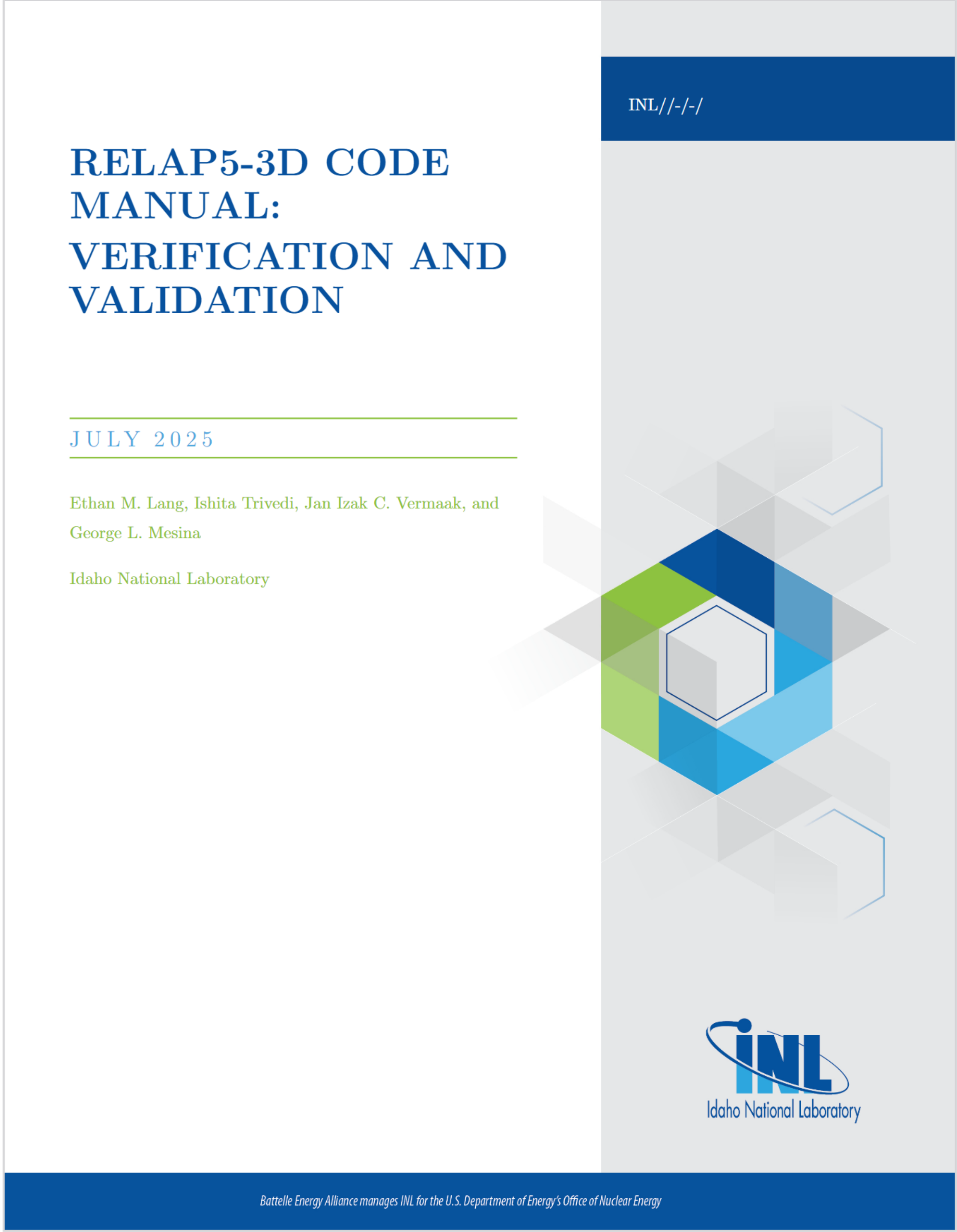
- Conversion to LaTeX based format provides
  - Ease of editing, maintaining, and adding content
- The ability to 'live-compile' the document was added so that:
  1. RELAP5-3D test system can be run on a particular computer system,
  2. Dynamically generated plots are made,
  3. And a resulting pass/fail status is declared.
- The DA manual can be compiled from that info.
- A Pass/Fail regression test table for the input files ran by the test system.

Table 2.4.3. Executed Separate Effects RELAP5-3D Input Files for Validation

Case Name	Input File	Status
Edwards-O'Brien Blowdown Test		
	edwards	Passed
	edwards-ni	Passed
Marviken Critical Flow Test 21		
	marv21	Passed
	marv21-ni	Passed
Marviken Critical Flow Test 22		
	marv22	Passed
	marv22-ni	Passed
Marviken Critical Flow Test 24		
	marv24	Passed
	marv24-ni	Passed
Marviken Jet Impingement Test 11		
	marv-jit11-ss	Passed
	marv-jit11-ss-ni	Passed
	marv-jit11-tr	Passed
	marv-jit11-tr-ni	Passed
Moby-Dick Air-Water		
	mobydck4c1	Passed
	mobydck4c1-ni	Passed
Christensen Test 15		
	chris15	Passed
	chris15-ni	Passed
GE Level Swell - 1 ft - Test 1004-3		
	ge1004-3	Passed
	ge1004-3-ni	Passed
GE Level Swell - 4 ft - Test 5801-15		
	ge5801-15	Passed
	ge5801-15-ni	Passed
	ge5801-15-lvltrack	Passed
	ge5801-15-lvltrack-ni	Passed
Bennett Heated Tube Tests 5358, 5294, and 5394		
	ben5358	Passed
	ben5358-ni	Passed
	ben5294	Passed
	ben5294-ni	Passed
	ben5394	Passed
	ben5394-ni	Passed
ORNL THTF Tests 3.07.9B, 3.07.9N, 3.07.9W and 3.09.10I		
	ornl379B	Passed
	ornl379B-ni	Passed

New V&V document

- “RELAP5-3D Code Manual: Verification and Validation” was created.
- Test cases are clearly divided into verification and validation categories.
- Creates new dynamically generated matrices
  - These indicate what code models are verified or validated by a particular test case.
- Information is summarized at the start of the document and in each section for ease of use.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RELAP5-3D CODE MANUAL:
VERIFICATION AND VALIDATION

JULY 2025

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Document Comparison

Verification and Validation

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Document Comparison (cont.)

Verification and Validation

2.2 Verification Matrices

Table 2.2.1. Traceability Matrix for Phenomenological Effects Verification Cases

Models Verified	Cases															
	3.1.1	3.1.2	3.1.3	3.1.4	3.1.5	3.1.6	3.1.7	3.1.8	3.1.9	3.1.10	3.1.11	3.1.12	3.1.13	3.1.14	3.1.15	3.1.16
1-in. small break LOCA																
6-in. small break LOCA																
Accumulator model																
CCFL																
CHF																
Cladding oxidation																X
Condensation																
Condensation heat transfer																
Conduction enclosure													X	X	X	
Critical flow																
Decay power																
Downcomer CCFL																
Entrainment					X											
Film boiling																
Flashing																
Force term							X	X								
Gravitational head		X	X													
Gravity	X															
Horizontal stratification							X	X								
Hydro numerics	X															
Interfacial drag in bubbly/slug																
Interfacial heat transfer																
Interphase drag																
Interphase evaporation																
Jet pump																
Large break LOCA																
Level tracking				X												
Liquid level		X	X			X										
Loop natural circulation																
Lower plenum refill																
Momentum equation	X															
Momentum equation (3-D)										X	X	X				
Natural circulation																

Developmental Assessment

Table 2.2-1. Phenomenological assessment cases.

Case Description	Models Validated
Bubbling steam through liquid	Entrainment, two-phase level
Cladding oxidation	Metal-water reaction
Conduction enclosure	Conduction enclosure
Conduction enclosure 1-D transient	Conduction enclosure
Conduction enclosure 2-D transient	Conduction enclosure
Core power	Decay power
Fill/drain	Level tracking
Gravity wave 1-D	Horizontal stratification, force term
Gravity wave 3-D	Horizontal stratification, force term
Manometer	Noncondensables, wall friction, liquid level, oscillations
Point kinetics ramp	Point kinetics
Pryor pressure comparison	Water packing
Pure radial symmetric flow (3-D)	3-D momentum equations
Rigid body rotation (3-D)	3-D momentum equations
R-theta symmetric flow (3-D)	3-D momentum equations
Water faucet	Hydro numerics, gravity, momentum equation
Water over steam (1-D)	Gravitational head, liquid level
Water over steam (3-D)	Gravitational head, liquid level

Table 2.2-2. Separate effects assessment cases.

Case Description	Models Validated
Bennett Heated Tube Tests 5358, 5294 and 5394	Non-equilibrium heat transfer, CHF, subcooled boiling, steam cooling
Christensen Test 15	Subcooled boiling heat transfer, void profile
Dukler air-water flooding	CCFL
Edwards' Pipe	Vapor generation, flashing, critical flow, pressure wave propagation

Document Comparison (cont.)

Verification and Validation

4.3.1 LOFT Experiment L3-7

Experiments were performed in the 1970s and 1980s in the Loss-of-Fluid Test (LOFT) facility, a 50 MWt power/volume-scaled nuclear reactor designed to investigate the response of a commercial pressurized water reactor (PWR) to loss-of-coolant accidents (LOCAs) and operational transients. Experiment L3-7 simulated a 1-in. diameter cold leg break with a maximum core linear heat generation rate of 52.8 kW/m (16.1 kW/ft).

Table 4.3.2. Code Models Validated by the LOFT Experiment L3-7 Case

Models Validated	Return to Traceability Matrix
1-in. small break LOCA	X

Table 4.3.3. Executed RELAP5-3D Input Files for the LOFT Experiment L3-7 Case

Input File	Status
l3-7_ss	Passed
l3-7_ss-ni	Passed
l3-7_tr	Passed
l3-7_tr-ni	Passed

Code Models Assessed. As an integral test facility, multiple code models are addressed. For this small break LOCA, the interest is in the overall system response, not that of the core, as there is no heatup. Parameters of significance are the break flow rates, system pressure, emergency core coolant (ECC) system response, and system mass distribution.

Experiment Facility Description. The LOFT facility is described in detail in Reeder.[1] The nuclear core was 1.68 m high with a 0.61-m diameter. The core contained nine fuel assemblies and 1,300 fuel rods that were representative of a commercial PWR. As shown in **Figure 4.3.1**, the facility contained two primary coolant loops. The intact loop represented three loops of a commercial plant, containing a hot leg, steam generator, cold leg, two primary coolant pumps, and the pressurizer. The broken loop represented a single loop, and included steam generator and primary coolant pump simulators, which modeled the flow resistance of these components. The broken loop could be configured to model either hot or cold leg breaks. Quick-opening blowdown valves (adjustable opening times of approximately 20 to 50 ms) simulated the initiation of primary coolant pipe ruptures, and orifices were used to model different break sizes. The break effluent was collected in a blowdown suppression tank.

The ECC system included a pumped high-pressure injection system (HPIS), a nitrogen-pressurized accumulator, and a pumped low-pressure injection system (LPIS). The accumulator was equipped with an adjustable height standpipe, which allowed the effective liquid volume to be varied between experiments. The ECC system was designed to allow injection to the intact loop hot leg, intact loop cold leg, reactor vessel upper plenum, lower plenum, or downcomer.

The LOFT facility was extensively instrumented. Fluid pressure, temperature, and flow rate were measured at key locations in the primary coolant, secondary coolant, and ECC systems. Three-beam gamma densitometers were used to measure fluid density at two locations in the intact and broken loops. Thermocouples measured fuel rod cladding and support tube temperatures at 196 core locations. Several fuel rod internal temperatures (fuel and plenum) were also measured. Neutron

Developmental Assessment

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5.1.2 Experiment Facility Description

The LOFT facility is described in detail in Reference 5.1-1. The nuclear core was 1.68 m high with a 0.61-m diameter. The core contained nine fuel assemblies and 1,300 fuel rods that were representative of a commercial PWR. As shown in Figure 5.1-1, the facility contained two primary coolant loops. The intact loop represented three loops of a commercial plant, containing a hot leg, steam generator, cold leg, two primary coolant pumps, and the pressurizer. The broken loop represented a single loop, and included steam generator and primary coolant pump simulators, which modeled the flow resistance of these components. The broken loop could be configured to model either hot or cold leg breaks. Quick-opening blowdown valves (adjustable opening times of approximately 20 to 50 ms) simulated the initiation of primary coolant pipe ruptures, and orifices were used to model different break sizes. The break effluent was collected in a blowdown suppression tank.

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The LOFT facility was extensively instrumented. Fluid pressure, temperature, and flow rate were measured at key locations in the primary coolant, secondary coolant, and ECC systems. Three-beam gamma densitometers were used to measure fluid density at two locations in the intact and broken loops. Thermocouples measured fuel rod cladding and support tube temperatures at 196 core locations. Several fuel rod internal temperatures (fuel and plenum) were also measured. Neutron flux was measured with four fixed detectors, which were designed to measure power transients, and four traversing in-core probes, which were designed to measure steady-state axial flux distributions at four different locations in the core.

Experiment L3-7 simulated a 1-in. diameter break in a cold leg pipe and investigated potential plant recovery methods. The break orifice was located upstream of the quick-opening blowdown valve. For this experiment, the primary coolant pumps were manually tripped following scram. HPIS injection to the intact loop cold leg was the only operating ECC system. HPIS and auxiliary feedwater flows were both

Summary

- A Github repository contains the new DA/V&V documentation system.
- Users with Platinum Level IRUG Memberships can download this repository and drop it in their RELAP5-3D installation, run the test system, and compile the document.
- Doing so effectively verifies and validates their installation of RELAP on their machine.
- The V&V document more clearly presents the outcome of the test cases in the form of traceability matrices and summary matrices
  - Users can find what they want at a glance, rather than searching the whole manual.
- The ability to easily add new verification/validation cases to the test and documentation systems using Python scripts was created; plots of RELAP data can be specified and will appear in the V&V document alongside user supplied commentary.

Future Work

This is a step in the right direction to meet NQA-1 standards. Future work includes:

1. Test results are judged based on expert opinion after inspection of plots. For validation with experimental data, uncertainties should be collected from literature or derived using inverse uncertainty quantification methods. Then a more quantitative standard can be established.
2. V&V for more processors and compilers.
3. V&V of additional fluids such as liquid metal coolants and existing RELAP code models with missing V&V cases.
4. Reduce document size by optimizing data presentation.

Acknowledgements:

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