REVISITING ISP-13 WITH A RELAP/SCDAPSIM/MOD3.5 MODEL USING CORE SCDAP COMPONENTS

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ABSTRACT

The recent accident in the Fukushima Daiichi nuclear power plant opened a discussion on severe accident management that includes the analysis of the accident by means of computational tools that can predict the core behavior in such extreme conditions.
The RELAP/SCDAPSIM/MOD3.5 code is designed to predict the behavior of Light Water Reactor (LWR) coolant systems during normal and accident conditions including severe accidents up to the point of reactor vessel failure. The code consists of two parts: the RELAP5 models calculate the overall Reactor Coolant System (RCS) thermal-hydraulic response, control system behavior, reactor kinetics and the behavior of special reactor system components such as valves and pumps, to predict the plant behavior under operational transients, Design Basis Accidents (DBAs) and Beyond DBAs; the SCDAP models calculate the behavior of the core and vessel structures under normal and severe accident conditions. Both portions of the code have been proven, separately, to accurately reproduce the response under its designed purpose, which are steady state, DBAs and BDBAs for the RELAP portion, and steady state and severe accident conditions for the SCDAP portion.
The analysis of potential scenarios does not define a priori the final state of the fuel rods, and thus the most adequate tool is a system code such as RELAP/SCDAPSIM/MOD3.5 capable of simulating accident scenarios where severe accident phenomena may or may not occur.
The present paper revisits the ISP-13 exercise, a cold leg double-ended guillotine LOCA conducted in the LOFT experimental facility, using two RELAP/SCDAPSIM/MOD3.5 models: the first one is entirely modeled with RELAP components, the second model keeps the RELAP nodalization with the exception of the core region, which is modeled with SCDAP components. The LOFT L2.5 experiment is a rather unique experiment since it features nuclear (UO2) fuel rods in a facility designed to simulate the major responses of a commercial pressurized water reactor (PWR). In addition, the fuel cladding of this experiment reached relatively high temperatures of around 1100 K. Even though this cladding temperature is far from the oxidation onset with steam, the LOFT L2-5 experiment challenges system behavior simulations by bringing the conditions close to those of severe accidents.
The final goal is to evaluate whether the use of SCDAP components in LOFT L2-5 experiment reproduces similar results to those obtained with a RELAP standalone model, and that both simulations are in good agreement with experimental data.

KEYWORDS
system codes, LOFT, LOCA

1. INTRODUCTION

The Loss-of-Fluid Test (LOFT) facility [1-3] is a 50-MW(t) pressurized water reactor (PWR) system with instrumentation that measure and provide data on the system thermal-hydraulic and nuclear conditions. The operation of LOFT system is typical of large [~1000MW(e)] commercial PWR operations.

Experiment L2-5 was performed in 1982 as part of the LOFT Experimental Program conducted by EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission (USNRC) [4]. The LOFT facility was configured to simulate a double-ended offset shear of a cold leg in the primary coolant system concurrent with a loss of offsite power. The peak fuel cladding temperature achieved was 1078±13K at 28s. and no evidence of core damage was detected. The L2-5 experiment was designated as the International Standard Problem 13 (ISP-13) by the OECD [5].

The system code RELAP/SCDAPSIM/MOD3.5 (from now on referred as RS/MOD3.5) is designed to run a wide range of conditions from normal operating conditions up through severe accidents. RS/MOD3.5 uses the publicly available RELAP5/MOD3.3 [6] and SCDAP/RELAP5/MOD3.2 [7] models, developed by the USNRC in combination with advanced numerics, advanced programming, and SDTP member-developed models and user options.

Both RELAP5/MOD3.3 and RS/MOD3.5 can approach pseudo 3D-modelling of annulus- and cylindrical- geometries by connecting vertical pipes with cross-flow junctions, using the appropriate junction areas. This approach has been used in previous analyses in order to accurately simulate the radial fluid temperature and velocity distribution in the core and therefore to accurately reproduce the clad-to-fluid heat transfer [8].

The L2-5 experiment provides a suitable environment to develop a pseudo-3D model of the core vessel and to perform a first step verification of SCDAP core components under non-severe accident conditions, versus the RELAP heat structures. The specific reasons for L2-5 choice are:

- LOFT L2 series are nuclear experiments performed with real fuel
- The peak cladding temperature (PCT) value 1078±13K is close to the onset of rapid oxidation (which under certain conditions it has been found to be as low as 1073.15K [9]) and the resulting undamaged core, provide the limiting conditions between a severe and non-severe accident conditions to verify SCDAP core components.
- L2-5 fuel was fresh, and therefore the burnup dependency of the power is not relevant. This allows a simplified approach in the RS/MOD3.5 model of imposing the power from a power-time table. This is the reason why the present work may be considered as a first, but necessary step of the verification of SCDAP components under non-severe accident conditions. The full verification should include a kinetics model including both high and low burnup fuel.

The present work focuses on:

(1) revisiting of Experiment L2-5 using an enhanced pseudo-3D vessel model; and
(2) a first step verification of SCDAP components under non-severe accident conditions by comparing the experiment results using a RELAP standalone model and a RELAP&SCDAP model.

In the present work, a pseudo 3D nodalization of the reactor pressure vessel of the LOFT facility has been generated with RELAP components. In parallel, a core model has been built by making use of SCDAP components. Post-test calculations have been carried out with RS/MOD3.5 (with and without SCDAP components) and RELAP5/MOD3.3.

2. TOOLS AND MODELS

The present section provides a brief introduction to the RELAP/SCDAPSIM code, with focus on the main modeling differences between RELAP heat structures and SCDAP core components (Section 2.1); a description of the hydrodynamic RELAP model for L2-5, with focus on the pseudo 3D-modelling of the vessel (Section 2.2); and a description of the core model using RELAP standalone heat structures and the combined model using RELAP heat structures to simulate the passive structures and SCDAP components for the active core structures (Section 2.3).

2.1. RELAP/SCDAPSIM code

The RELAP/SCDAPSIM code is being developed as part of the international SCDAP Development Training Program (SDTP). SDTP, a cooperative program, includes more than 90 organizations in 30 countries [10] supporting the development of technology, software, and training materials for the nuclear industry. The program members and licensed software users include universities, research organizations, regulatory organizations, vendors, and utilities located in Europe, Asia, Latin America, and the United States. Innovative Systems Software (ISS) administers the program and is responsible for the configuration control and distribution of the RELAP/SCDAPSIM system thermal hydraulic code version, see www.relap.com for more detailed information.

RELAP/SCDAPSIM has been used to analyze a variety of nuclear power plant designs. The applications have included RELAP-only input models for normal operating or transient conditions where core damage is not expected as well as combined RELAP-SCDAP input models that included the possibility of transients with the loss of core geometry.

The RELAP models calculate the overall RCS thermal-hydraulics, control system interactions, reactor kinetics, and the transport of noncondensable gases. The RELAP code is based on a two-fluid model allowing for unequal temperatures and velocities of the fluids that is solved by either a semi-implicit or nearly-implicit numerical scheme to permit economical calculation of system transients.

System structures can be modeled with RELAP heat structures or SCDAP core components. The RELAP heat structures are one-dimensional models with slab, cylindrical, or spherical geometries. The SCDAP core components include representative light water reactor (LWR) fuel rods, silver-indium-cadmium (Ag-In-Cd) and B4C control rods and/or blades, electrically heated fuel rod simulators, and general structures.

The SCDAP code models the core behavior during a severe accident: heatup and damage progression in the core structures and the lower head of the reactor vessel. SCDAP solves a two-dimensional heat conduction equation to calculate the temperature response for the SCDAP components. The calculations of damage progression include calculations of the fuel rod heatup, ballooning and rupture, fission product release, rapid oxidation, zircaloy melting, UO2 dissolution, ZrO2 breach, flow and freezing of molten fuel and cladding, and debris formation and behavior. The code also models control rod and flow shroud behavior.

A RELAP-only input deck models the core components (vessel, shroud, fuel rods) using the so-called heat structures. This component simulates the heat transfer from the fluid to a solid material and the temperature across the solid structure solving a one-dimensional heat conduction equation and given built-in or user-provided material properties. The heat structures allow the simulation of passive heat transfer, i.e. no heat generation, and active heat structures such as fuel rods. The code allows to impose boundary conditions or to use the built-in correlations for a variety of boundary conditions, from single phase liquid to single phase vapour, including non-condesnables effect. The limitations of the default RELAP standalone heat transfer model are that the fuel rod behaviour is not dynamically simulated.

2.2. Pseudo 3D nodalization for L2-5

The present section focuses on the pseudo 3D modelling of the vessel. The rest of the plant nodalization has been taken from [11] with minor modifications.

The reactor pressure vessel has been fully renodalized to account for the different geometrical aspects in each radial plane. The core region has been nodalized with nine parallel pipes connected with crossflows, each pipe representing a single fuel assembly (see left figure in Fig.1). Areas and hydraulic diameters have been calculated independently for each pipe according to the facility drawings. The size and distribution of holes is heterogeneous in some radial planes, therefore the need for a fine nodalization with several parallel pipes. Each fuel assembly is linked to heat structures with a different linear heat generation rate. The core bypass is modelled by a single pipe. This core nodalization allows reproduction of a heterogeneous velocity distribution in the radial plane.

The LOFT facility has two independent downcomers, each of them has been nodalized with 4 annulus components connected with crossflows (see the right drawing in Fig.1).

This nodalization approach does not significantly improve the results in terms of system behavior but it provides more detailed information, especially for the heterogeneity of the cladding temperatures.

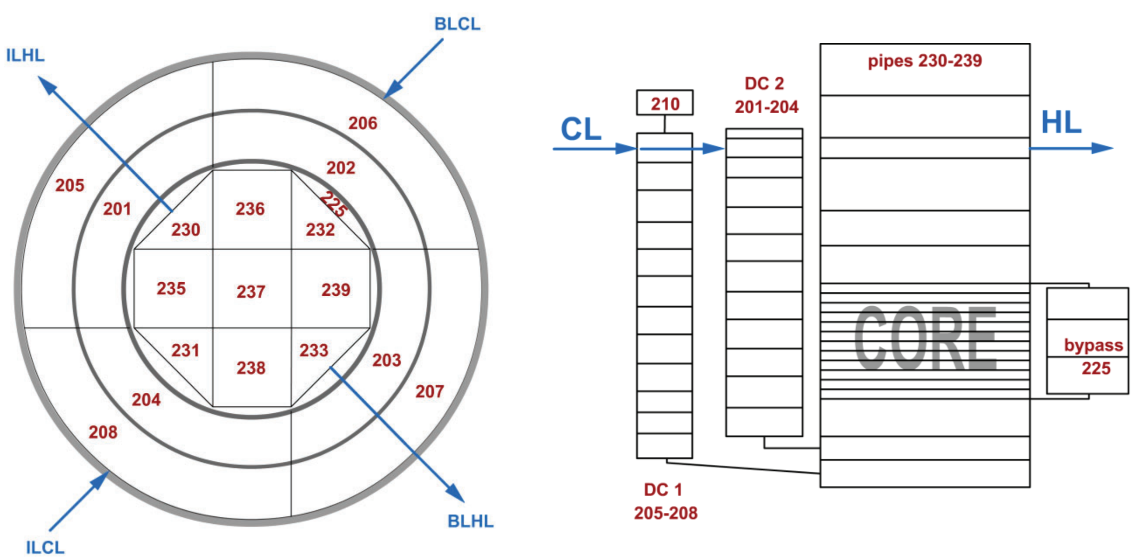


Figure 1. Scheme of the nodalization of the Reactor Pressure Vessel for the LOFT facility (left: top view, right: axial view)

2.3. LOFT L2-5 core model

The LOFT L2-5 core is modeled using (a) RELAP-only heat structures and (b) a combination of RELAP heat structures and SCDAP core components:

- Both models (a) and (b) use RELAP heat structures to model the passive heat structures.
- Model (a) uses RELAP heat structures to simulate fuel rods
- Model (b) uses SCDAP components to simulate control and fuel rods

The features of the two models are summarized in Table I and II: Table I details the correspondence between the RELAP heat structures and the SCDAP components, and specifies their location in the hydrodynamic model; Table II details the features of each power zone.

The formal main differences are that SCDAP does not use the approach of the single "hot rod", instead it models 10 "hot rods"; and that the RELAP standalone model does not have control rod components. However, the latter should not affect in a non-severe accident scenario: the effect of the control rod components in a transient where melting does not take place is merely the heat that may be absorbed or delivered to the system, which is negligible compared to the other structures present in the core.

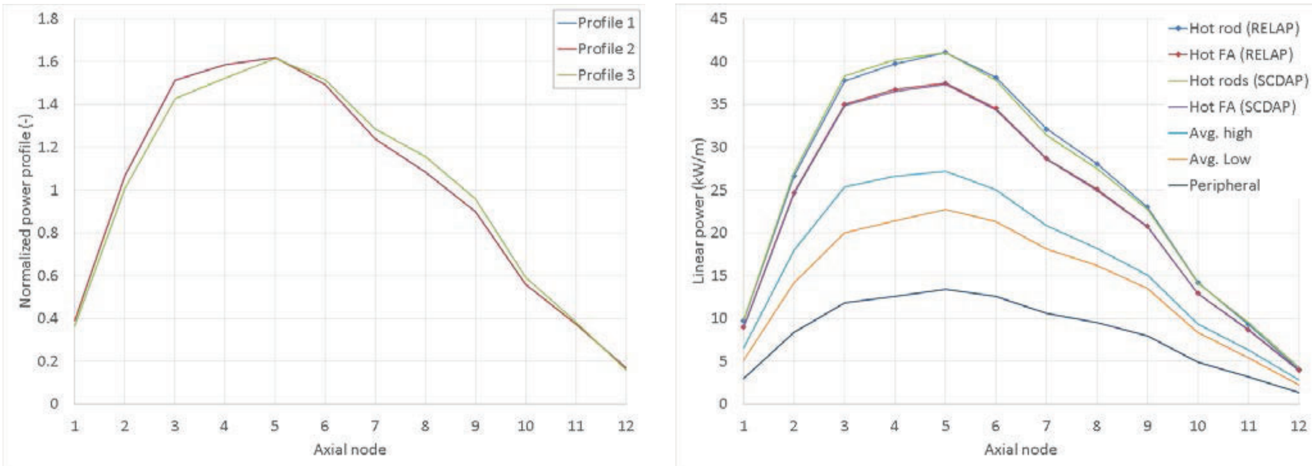
Power profiles, taken from Table 2-4 in ref. [4], are shown in Fig.2 (a), and the linear power for each rod type is depicted in Fig.2 (b).

Table I. RELAP and SCDAP active core components

<table>
<tr><th rowspan="2">Component type</th><th colspan="2">Number</th><th rowspan="2">Location
Hydrodynamic channel CCC</th><th rowspan="2">Description
Rod type</th></tr>
<tr><th>Model (a)
RELAP CCCG</th><th>Model (b)
SCDAP CCC</th></tr>
<tr><td rowspan="14">Fuel</td><td>2310</td><td>023</td><td rowspan="2">237</td><td>Hot rod/s</td></tr>
<tr><td>2320</td><td>001</td><td>Hot channel</td></tr>
<tr><td>2330</td><td>002</td><td rowspan="2">235</td><td>Average high</td></tr>
<tr><td>2350</td><td>006</td><td>Peripheral</td></tr>
<tr><td>2331</td><td>003</td><td rowspan="2">236</td><td>Average high</td></tr>
<tr><td>2351</td><td>007</td><td>Peripheral</td></tr>
<tr><td>2332</td><td>005</td><td rowspan="2">238</td><td>Average high</td></tr>
<tr><td>2352</td><td>009</td><td>Peripheral</td></tr>
<tr><td>2333</td><td>004</td><td rowspan="2">239</td><td>Average high</td></tr>
<tr><td>2353</td><td>008</td><td>Peripheral</td></tr>
<tr><td>2340</td><td>010</td><td>230</td><td>Average low</td></tr>
<tr><td>2341</td><td>011</td><td>231</td><td>Average low</td></tr>
<tr><td>2342</td><td>012</td><td>232</td><td>Average low</td></tr>
<tr><td>2343</td><td>013</td><td>233</td><td>Average low</td></tr>
<tr><td>Control</td><td>-</td><td>014,015,016,
017,018,019,
020,021,022</td><td>One SCDAP component
"control" per fuel assembly</td><td>-</td></tr>
</table>

Table II. Rod characteristics

Rod type	Number of rods	Radial peaking factor	Maximum linear power (kW/m)	Profile
Hot rod for model (a)	1	1.533	41.09	1
Hot channel for model (a)	203	1.4	37.49	2
Hot rods for model (b)	10	1.533	41.04	1
Hot channel for model (b)	194	1.394	37.31	2
Average high	720	1.015	27.18	2
Peripheral	96	0.85	22.72	3
Average low	280	0.5	13.37	3



(a) (b)

Figure 2. (a) Power profiles; and (b) Linear power per each fuel rod type

3. RESULTS

The present section briefly summarizes LOFT configuration for L2-5 and describes the experiment (Section 3.2) and presents the results of the simulations (Section 3.3).

3.1. The L2-5 experiment

The LOFT system configuration for L2-5 is shown in Fig.3. The major components of the LOFT system are: reactor vessel including a core with 1300 unpressurized nuclear fuel rods with an active length of 1.67 m; an intact loop (triple loop) with a pressurizer, steam generator, two pumps arranged in parallel, and piping connected to the break plane orifice; a broken loop (single loop) with a simulated pump, simulated steam generator, two break plane orifices, two quick opening blowdown valves (QOBVs), and two isolation valves; an emergency core cooling system consisting of two accumulators, a high pressure injection system (HPIS) and a low pressure injection system (LPIS); and a blowdodwn suppression system consisting of a header and a suppression tank.

At operation reactor power of 36 MW, experiment L2-5 was initiated by opening the two QOBVs, in the broken loop hot and cold legs. The primary coolant pumps were tripped by the operators and were not connected to their flywheels during the coastdown to simulate a loss of offsite power. In response to the loss of offsite power the high- and low- pressure were delayed to 24s and 37s, respectively.

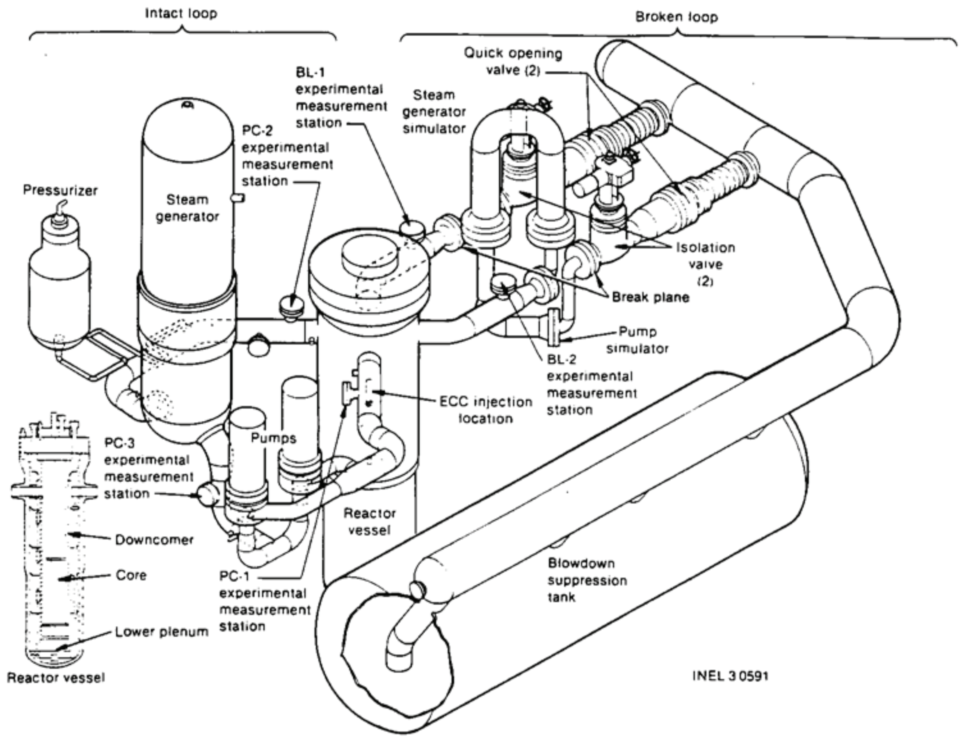


Figure 3 LOFT system configuration for cold leg break

3.2. L2-5 simulation

Three simulations have been completed using:

- RELAP5/mod3.3
- RS/mod3.5 w/o SCDAP components,
- RS/mod3.5, using SCDAP components

Table III summarizes the steady state results for each calculation. The three calculations provided almost the same results. All parameters are within the uncertainty ranges of the measurements.

Table III. Steady state qualification

Parameter	Experimental	RELAP5_NRC	RELAP_ISS	RELAP/SCDAP
IL Mass flow rate (kg/s)	192.4±7.8	189.97	190.79	191.06
IL HL pressure (MPa)	14.94±0.06	14.99	14.97	14.97
IL CL temperature (K)	556.6±4.0	556.7	556.90	556.90
IL HL temperature (K)	589.7±1.6	590.95	591.11	591.07
Core Power (MW)	36.0±1.2	36.00	36.00	36.00
Pressurizer Level (m)	1.13±0.18	0.99	0.99	1.00
BL CL temperature (K)	554.3±4.2	556.4	556.60	556.60
BL HL temperature (K)	561.9±4.3	562.96	563.06	563.51
SG Pressure (MPa)	5.85±0.06	5.85	5.84	5.84
MFW mass flow rate (kg/s)	19.1±0.4	19.04	19.05	19.05

The transient results of the three calculations and the experimental data are shown in Fig. 4 to Fig. 9. In terms of system behavior, the three calculations provide practically identical results. Differences are only observed in the cladding temperatures of the different parts of the core.

A LOCA scenario can be subdivided in three distinct phases: blowdown (first greyed area in the plots), ECC/bypass (whitened section) and reflood (second greyed area). In the L2-5 experiment, the blowdown phase lasts for about 18 s when the HPIS is initiated. The ECC/bypass is ended approximately at 30 s when the break flow turns from two phase flow to single phase steam flow. At this point all cladding temperatures start to decrease. The reflood phase ends at around 60 s when the core is completely quenched. The timing of these three phases is well reproduced by the three calculations.

- Blowdown phase: The blowdown phase is reproduced with accuracy by the three calculations. Primary pressures, break flows and cladding temperature are in very close agreement with the experiment. The maximum cladding temperature during the blowdown phase in the experiment was 1062.6 K and in each calculation is 1049.5 K (RELAP5/mod3.3), 1047.1 K (RS/mod3.5 w/o SCDAP components) and 1032 K (RS/mod3.5 w SCDAP). Towards the end of this phase, there is a rapid quench of some parts of the core due to the loop seal clearing in the intact loop (see Fig.9). In the three calculations, loop seal clearing occurs as well at this time although the effect is not as visible in the cladding temperatures (specifically for the hot rod).
- ECC/bypass phase: During this phase the PCT in the three calculations remain almost constant, whereas in the experiment the PCT increases. Therefore, one can assume that the ECC/bypass phenomenon is more intense in the experiment, where the PCT climbs to 1077 K whereas the three calculations remain close to 1020 K. During this phase the pressure in the hot leg is slightly underpredicted by the three calculations. In addition, the break flow in the cold leg is overestimated
- Reflood phase: The reflood phase starts similarly in the three calculations and in the experiment and ends also with similar timing. The most noticeable difference during this phase is that the quenching of the top of the core occurs earlier in the experiment indicating a deeper penetration of coolant from the upper plenum. This might be due to an overestimation of the CCFL phenomenon at the top plate.

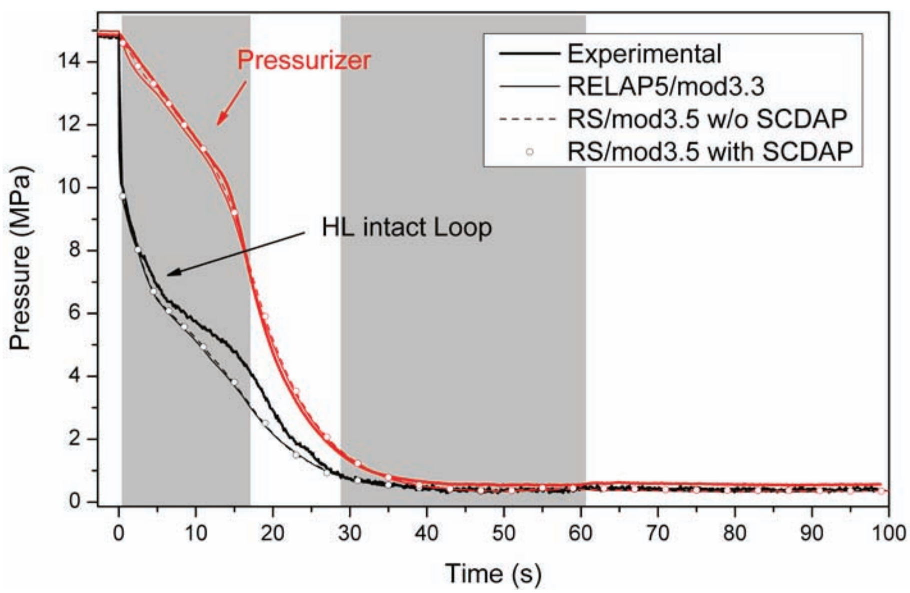


Figure 4 Intact loop HL and Pressurizer pressures

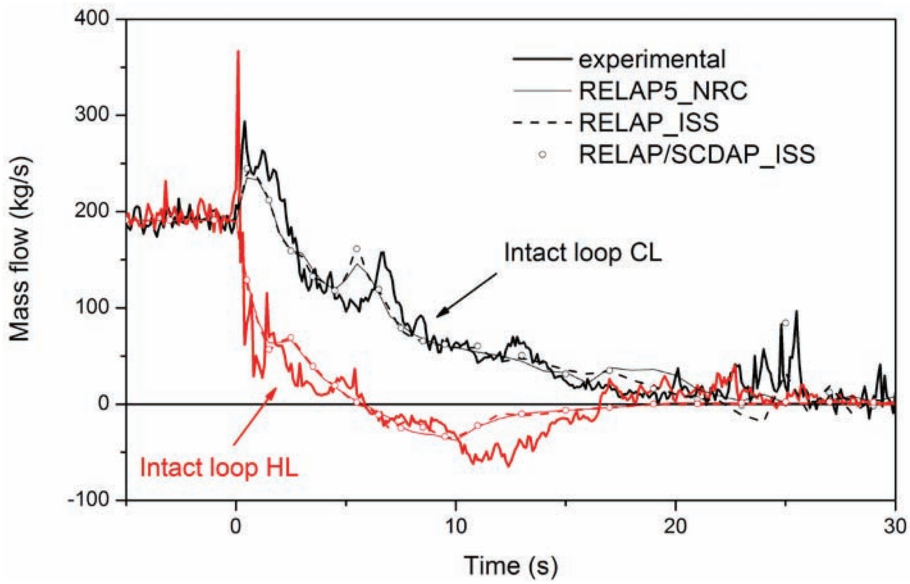


Figure 5 Intact loop HL and CL mass flows

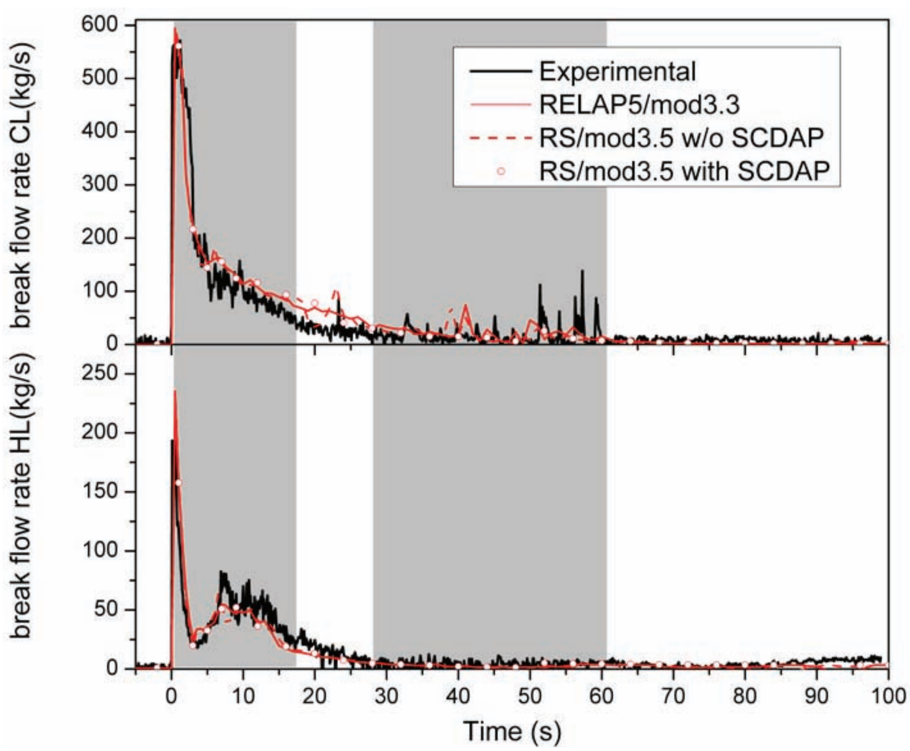


Figure 6 Break flow rates

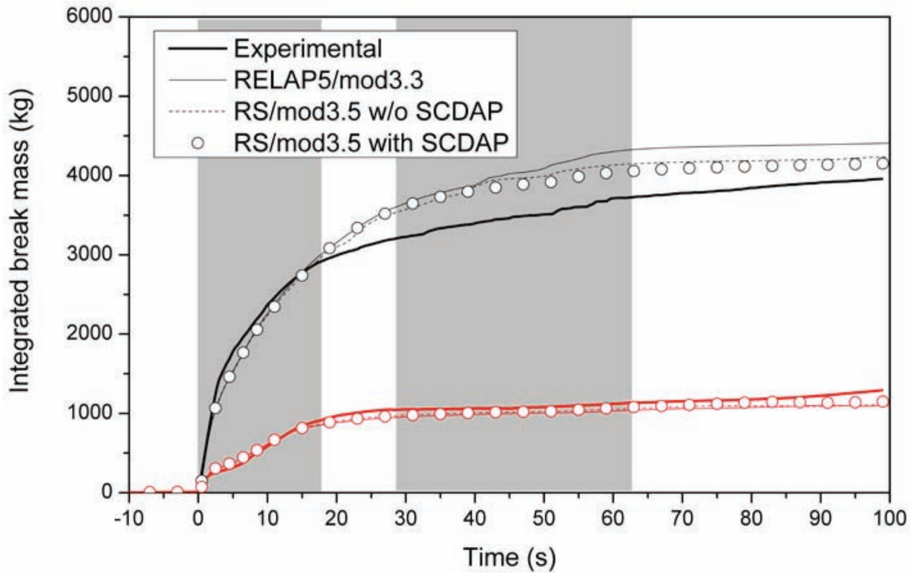


Figure 7 Integrated discharged mass

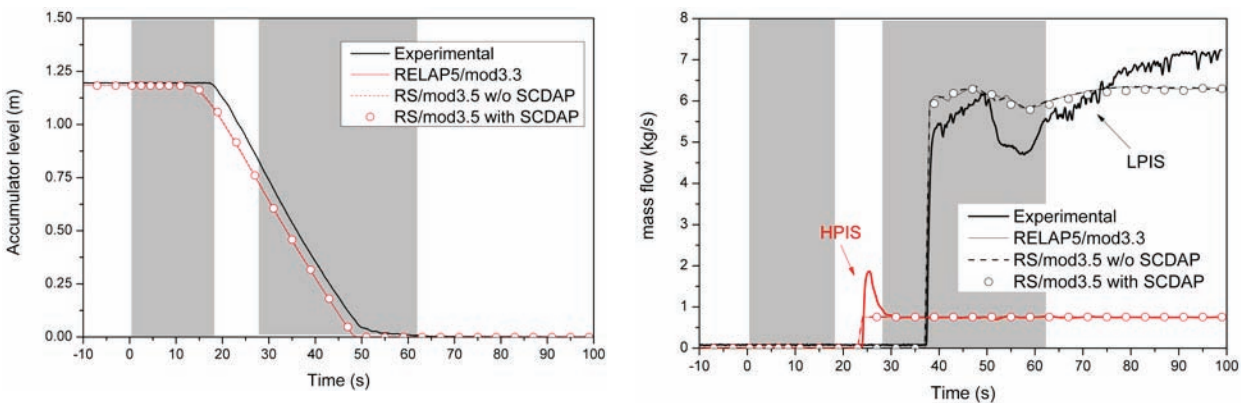


Figure 8 Left: Accumulator level. Right: HPIS and LPIS mass flows

Figure 9 shows the cladding temperatures measured across the core region in comparison with the calculated results. There were three types of rods with thermocouples: a hot rod, an average rod and a peripheral rod. Even though, the exact peaking factor for these rods is unknown, they should be close to the values defined in Section 2.2. In each sub-graph in Figure 9 the legend specifies the location of the measurement, the value in inches refers to the elevation of the measurement from the bottom. It can be observed that very different cladding temperatures are calculated at the same elevation but in different core zones. The three calculations provided very similar results of the hot rod throughout the transient. For the average and periphery rods, the three calculations provided slightly different results.

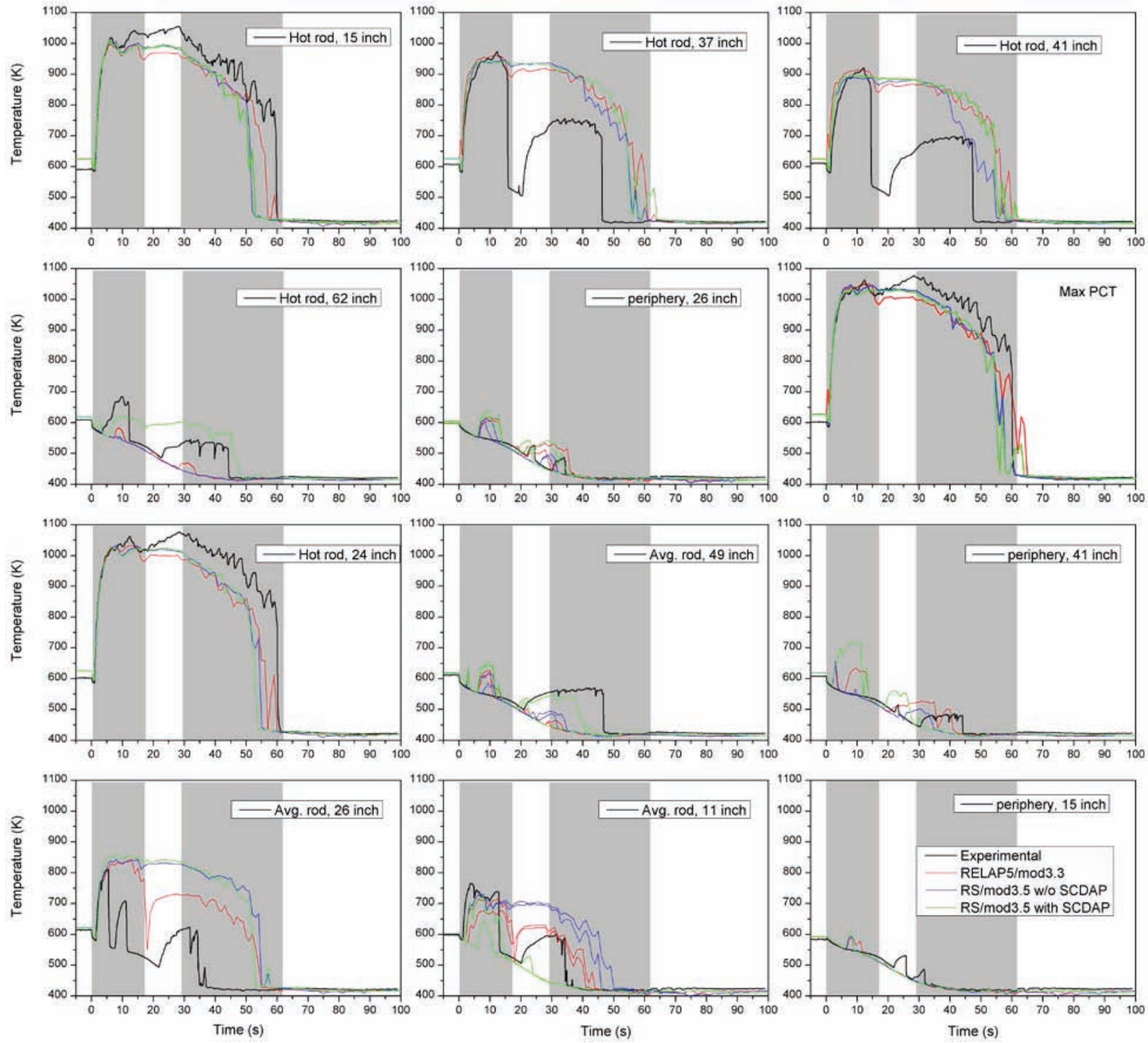


Figure 9 Cladding temperatures in different parts of the core.

4. CONCLUSIONS

The LOFT experiment L2-5 (OECD ISP-13) has been revisited using a pseudo 3D dimensional vessel for the hydrodynamic model, and using two approaches to model the solid core vessel structures:

- Model (a) has a core vessel modeled using RELAP heat structures
- Model (b) has a core vessel model using RELAP heat structures to model the passive core structures, and SCDAP components to model the fuel and control rods.

The calculations have been carried out using two version of the RELAP code: the RELAP5/MOD3.3 for RELAP-standalone model (a); and RELAP/SCDAPSIM/MOD3.5 for both models (a) and (b).

On the one hand, the agreement of the calculation results with the experimental data allows a user to conclude that the pseudo-3D model of the core vessel is an accurate approach to accurately reproduce the heterogeneity of the core region, especially for the radial core power distribution but also to take into account the non-homogeneous size and location of flow paths at some axial levels.

On the other hand, the agreement of the calculation results for both models and codes allow concluding that a first step, i.e. verification for low fuel burnup, of the SCDAP components verification versus the RELAP heat structures during a non-severe accident scenario process has been carried out successfully. It has also been shown that the calculated results from the two RELAP versions RELAP5/MOD3.3 and RELAP/SCDAPSIM/MOD3.5 show no relevant differences.

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