NURETH14-142

COMPUTER CODES VALIDATION FOR CONDITIONS OF CORE VOIDING

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Abstract

Void generation during a Loss of Coolant Accident (LOCA) in a core of a CANDU reactor is of specific importance because of its strong coupling with reactor neutronics. The use of dynamic behaviour and computer code capability to predict void generation accurately in the temporal and spatial domain of the reactor core is fundamental for the determination of CANDU safety. The Canadian industry has used the RD-14M test facilities for its code validation. The validation exercises for the Canadian computer codes TUF and CATHENA were performed some years ago. Recently, the CNSC has gained access to the USNRC computer code TRACE. This has provided an opportunity to explore the use of this code in CANDU related applications. As a part of regulatory assessment and resolving identified Generic Issues (GI), and in an effort to build independent thermal hydraulic computer codes assessment capability within the CNSC, preliminary validation exercises were performed using the TRACE computer code for an evaluation of the void generation phenomena.

The paper presents a preliminary assessment of the TRACE computer code for an RD-14M channel voiding test. It is also a validation exercise of void generation for the TRACE computer code. The accuracy of the obtained results is discussed and compared with previous validation assessments that were done using the CATHENA and TUF codes.

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) protects the health, safety and security of Canadians as well as the environment, and respects Canada's international commitments on the peaceful use of nuclear energy. The CNSC regulates the use and development of nuclear technology in Canada. To accomplish this mission successfully, it is important to continually develop its capabilities to perform safety assessment and analysis of nuclear installations in Canada.

The Canadian nuclear power generation industry is based on CANDU technology. CANDU power plants have been developed and used to generate electricity in CANADA and other countries for more than 40 years. The CANDU reactor is moderated and cooled with heavy water. The coolant water is under pressure and therefore a CANDU is a Pressurized Heavy Water Reactor (PHWR).

Safety assessment of CANDU reactors was developed on a similar framework as other types of reactors elsewhere in the world. It is based on safety analysis of the power plants for postulated accidents and events. One of the most important events that is analysed within Design Basis Events is a postulated large Loss of Coolant Accident (LOCA).

CANDU reactors use natural uranium fuel and a heavy water moderator and coolant. These design characteristics are important with respect to the behaviour of this reactor during a LOCA event. Specifically, CANDU reactors have a positive void coefficient. That is, power will increase with the development of void in the coolant. Therefore, one of the important phenomena to assess in the context of a LOCA scenario is void generation.

This paper presents work performed by CNSC staff to analyse important phenomena related to LOCA accidents for CANDUs. This assessment was driven by a few major objectives. One objective was to further develop independent assessment capabilities of the CNSC. The other was to perform a preliminary assessment of the TRACE computer code for potential use in CANDU applications and compare the results against current processes for resolving generic safety issues.

2. Safety assessment of large LOCA for CANDU reactors

As mentioned above, one of the most important events from the perspective of safety analysis of CANDU reactors is the large LOCA. It has similarities with light water reactor LOCA events but it also has specific characteristic phenomena only related to CANDU reactors.

2.1 Large LOCA description of phenomena

A large LOCA event is caused by a break in the Heat Transport System (HTS) pressure boundary that generates thermalhydraulic and neutronic conditions such that the Reactor Coolant System (RCS) and Reactor Regulating System (RRS) are incapable of maintaining heat removal and reactivity balance. Because of the inherent positive void coefficient characteristic of the CANDU reactor core, an immediate reactor power excursion occurs. The large LOCA event is characterized by the following most important phenomena:

- An immediate power excursion caused by coolant voiding in the horizontal core channels;
- A reactor trip that terminates the power excursion;
- A degradation of fuel cooling because of the large rate of coolant discharge into containment.
- Potential for pressure tube deformation and fuel failure.
- A requirement of an available Emergency Core Coolant System (ECCS) for injection to re-establish fuel cooling; etc.

It is evident that coolant voiding has significant importance in the safety assessment of a LOCA. In early 2000s, CNSC staff noted that the lack of experimental support for channel voiding rates was becoming increasingly important as newer large LOCA assessments were showing reduced safety margins.

In February 2001, CNSC staff raised Generic Action Item (GAI) 00G01 "Channel Voiding During a Large LOCA". Among other requirements, two significant parameters that determine the magnitude of the power pulse were identified: the rate and extent of coolant voiding and reactivity changes due to voiding. These parameters were predicted by computer models however they have not been adequately validated. This GAI has been intended to deal with the channel voiding issue; the void reactivity is being dealt separately under another GAI (95G04). There had been no direct void fraction measurements applicable to CANDU fuel channel conditions.

As closure criteria, the licensees were expected to carry out following three principle items:

- channel void measurements during large LOCAs relevant to reactor conditions; (including assessments of the heat transfer rate effects and scaling on the voiding);
- validation exercises with the relevant safety analysis computer codes against the channel void data;
- an impact assessment of the safety margins in the safety report.

The industry committed to:

1. develop special measurement instrumentation,
2. perform a test in the RD-14M test facility
3. perform a validation exercise for the CATHENA and TUF codes
4. address scaling of the test facility, and,
5. determine the impact of this evaluation on safety margins of power plants.

2.2 Computer codes

Safety analysis of CANDU reactors is carried out by computer codes. The Canadian industry has developed a number of computer codes that are CANDU specific and that are used in safety assessment. The main thermalhydraulic codes used in assessments are CATHENA and TUF. For the GAI 00G01 these two codes were intended to be validated by the experiments carried out in the RD-14M test facility.

CATHENA (**C**anadian **A**lgorithm for **THE**rmalhydraulic **N**etwork **A**nalysis), [3], is a one-dimensional, non-equilibrium, two-phase, two fluid network analysis code that has been in use for over two decades for safety assessment and design of nuclear power plants. CATHENA was developed by AECL (Atomic Energy of Canada Limited).

TUF (**T**wo **U**nequal **F**luids), [4], is a computer programme for simulation of the thermalhydraulics of the piping circuits of heat transport systems and the secondary side light water systems of a CANDU reactor. The code is used in design, commissioning, operational support and the safety analysis of the CANDU reactors of Ontario Power Generation, OPG (former Ontario Hydro). OPG is also the developer of the TUF code.

Both codes are supposed to be realistic (best estimate) codes that are used in the thermalhydraulic part of safety assessment. They are based on two fluid one dimensional non-equilibrium models with non-condensable gasses capability.

The CNSC has not developed its own thermalhydraulics code and has performed only limited independent in-house assessment using industry developed codes. While the main responsibility for performing such analyses will always reside with industry, the CNSC has recently been investigating the potential benefits and possibility of performing more analyses independently.

Along those lines, for the past several years the CNSC has participated in the USNRC run CAMP (Code Applications and Maintenance Program). This has provided CNSC staff with the opportunity to explore other computer codes and approaches to safety evaluations. CAMP codes offer wide and highly developed experimental thermalhydraulic databases (for validation) plus transparent and open feedback among users. The CNSC could see many positive benefits that could come from direct code to code comparison with Canadian codes.

Thus it was decided to explore the capabilities of the TRACE code for application in CANDU safety related assessments. There were implicitly defined objectives of this work of addressing coolant voiding in CANDU channels during large LOCA. This work should be considered as being very preliminary.

TRACE (**TR**AC/RELAP **A**dvanced **C**omputational **E**ngine), [9], [10], is a modernized thermal-hydraulics code designed to consolidate and extend the capabilities of NRC's 3 legacy safety codes - TRAC-P, TRAC-B and RELAP. It is able to analyze large/small break LOCAs and system transients in both pressurized- and boiling-water reactors (PWRs and BWRs). The capability exists to model thermalhydraulic phenomena in both one-dimensional (1-D) and three-dimensional (3-D) space. This is the NRC's flagship thermalhydraulics analysis tool.

It should be noted that TRACE code has already been used in Canada for assessment of CANDU specific and other thermalhydraulic problems: [5], [6], [7], [8].

3. RD-14M test facility

3.1 Test facility description

The RD-14M, [2], is a major thermalhydraulic integral test facility to support safety research of CANDU reactors, Figure 1. It was built in the early 90s and it is successor to the previously constructed thermalhydraulic test facilities (RD-12, RD-14). The RD-14M presents a typical configuration of CANDU "figure-of-eight" flow configuration that consists of two HTS pumps, two steam generators and two groups of reactor core channels.

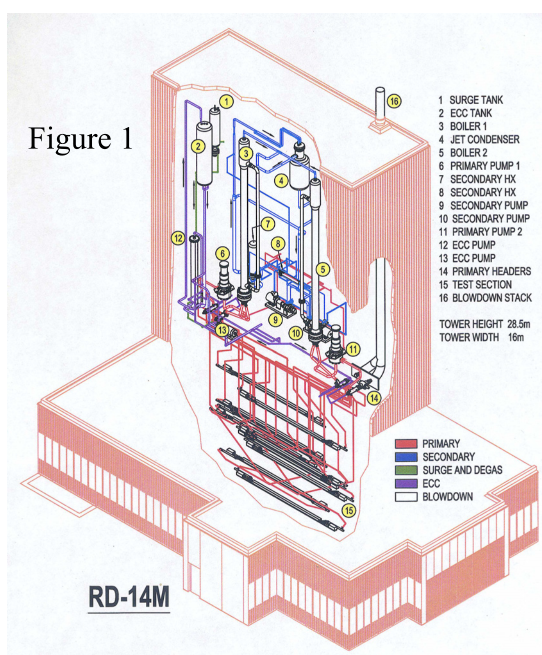

Figure 1

The RD-14M test facility is operated under CANDU HTS pressure, and it maintains the full height of CANDU HTS components. It was designed to reproduce CANDU reactor typical conditions, such as fluid mass flux, temperature, pressure, enthalpy and transit time. The RD-14M facility can achieve typical conditions for forced and natural circulation in the primary-side. The facility consists of five horizontal channels per pass and a 1:1 scaling of the vertical elevations throughout the loop, see Figure 2. Each six-meter-long channel contains 7 electrically heated Fuel Element Simulators (FES), connected to end-fitting simulators, Figure 2b. The feeder volumes, areas, and metal masses are appropriately scaled to the channels (7-pin FES). The thermal characteristics of the FES are similar to CANDU fuel in terms of power density, heat flux and heat capacity. The above header piping is sized to give reactor-typical pressure drops.

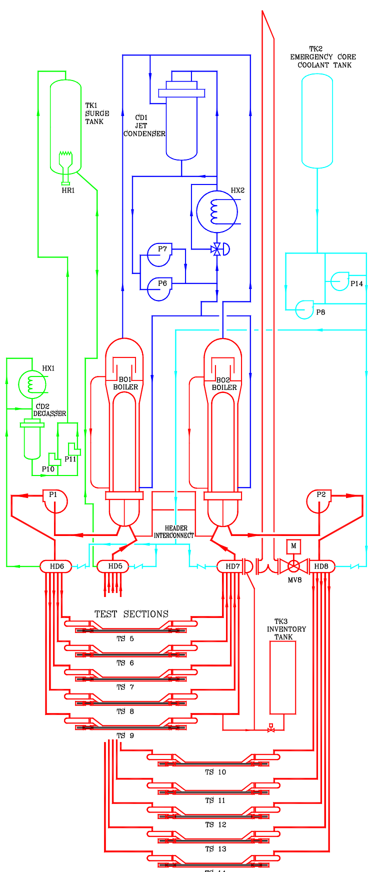

Figure 2

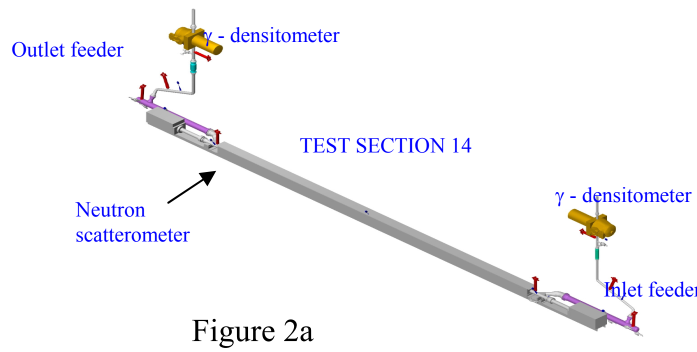

Figure 2a

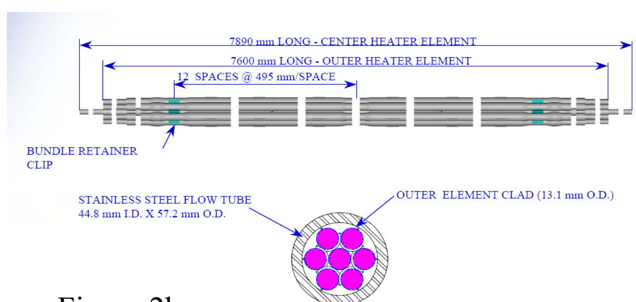

Figure 2b

The heated section is a 6-m long channel between the inlet and outlet end-fitting simulators, composed of a flow tube containing a 7-pin FES. Figure 2b presents the cross-section of a typical RD-14M heated section. The FES pins are divided into 12 axial segments, each having a length of 495 mm and separated by short unheated sections to simulate fuel bundle spacers. The FES pins consist of a central core of magnesium oxide surrounded by an electrically heated Inconel 625 tube of 7.62 mm outside diameter. The tube is insulated from the 13.18 mm outside diameter stainless steel (type-304) sheath by a 2 mm thick annulus of boron nitride.

The steam generators are scaled approximately 1:1 with typical CANDU steam generators, in terms of tube diameter, mass flux, and heat flux. The secondary-sides of the steam generators contain an internal preheater and an external downcomer.

Primary fluid circulation is provided by two centrifugal pumps. The pumps deliver full reactor typical head (about 225 m) at mass flow rates similar to a single reactor channel (about 24 kg/s). Primary circuit pressure is maintained by a loop pressurizer that contains an electrical heater.

The RD-14M facility is equipped with an Emergency Core Cooling System (ECCS) system that provides cooling to the FES under postulated LOCA conditions. ECCS was not used in the test modeled with TRACE in this paper.

More detailed information on the RD-14M facility can be found in [11].

3.2 Facility/Test instrumentation

The RD-14M facility is extensively instrumented for data collection during experiments. The data acquisition system consists of several input multiplexers each connected to an analog-to digital converter, and a data acquisition computer capable of scanning up to 768 channels at a maximum rate of about 50 ms per scan (20 Hz).

Measurement accuracies are the following: fluid and FES temperature ±2.0^{o}C, heated section power ±1.5%, Pressure and differential pressure ±0.5%, flow rates ±0.5%, void fraction in channel ±0.1.

The void fraction at the inlet and outlet of each boiler, at the pump discharge, and at the inlet and outlet of each test section was measured using three-beam, two-beam, and single-beam gamma densitometers, respectively.

For the test B0113, the data-acquisition system sampled all instrument signals at a rate of 20 scans per second. Over 200 temperatures and other parameters at various locations around the primary and secondary loops were measured.

AECL has developed special measurement technique that includes hardware and software for measurement void fraction in the channel. It is based on neutron scatterometer readings. The neutron scatterometer is located on channel heated section 14 (4740 mm from the inlet end). It consists of neutron source and eight detectors. As shown in Figure 2a, this corresponds to segment number 10 of the FES, and the heated section nodalization.

3.3 Test Setup

Assessment of coolant voiding was carried out for two series of tests. The first series used the test full channel configuration (10 channels). The second used only two channels. This paper presented the test B0113 from the second series which had only two heating channels (FS9, and FS14).

Test B0113 was designed to measure void fraction in the channel under conditions of power pulse and to represent a supposed loss of Class IV power. For this test series it was the modified RD-14M test facility which used two horizontal channels. The power pulse was specified to occur in channel 14 reaching its peak magnitude of 1.7 times higher than nominal power in the first second of simulate large LOCA break. Pumps run down was representative of a supposed Loss of Class IV power. For this test, mass flow rate of the break discharge was not measured.

3.4 Test Procedures

The loop was warmed at moderate power and full reference readings for the neutron scatterometer were taken at that power level. At this power setting the temperatures in HS14 were well subcooled. Heated-section power and pump speeds were then adjusted to bring the loop to the desired steady-state starting conditions. Full reference readings were taken for all gamma densitometers. The pressurizer was isolated shortly before the discharge valve simulating the break was opened (when the primary-side pressure reached 5.5 MPa(g)).

The following table gives the sequence of events for the B0113 test.

Time	*Event (Boundary conditions for the test B0113)*
0.00	Data collection started
50	Surge tank isolated
60.00	Signal sent to break valve (MV8) to open Signal sent to power supply 3 (PS3) to start simulated power pulse
60.80	MV8 starts to open
62.00	Signal sent to step power down to decay levels and to start primary pump rundowns
…	…

Shortly after the blowdown, primary-pump rampdowns, emulating a loss of class IV power, was initiated. The isolation of the pressurizer, before the blowdown, differed from the procedure used during most Phase I experiments where the pressurizer was isolated after the blowdown.

4. Computer codes assessment

For the assessment of channel voiding of the test B0113, the TRACE code (V5.0, p02), [9], [10], with support SNAP (Symbolic Nuclear Analysis Package) was used. The configuration of B0113 test was required to develop the nodalization of the test flow configuration.

4.1 RD-14M nodalization

As mentioned before, the RT-14M test facility was reduced and operated in the configuration that used only two channels, channel HS9 and HS14. The nodalization development was done by SNAP with the assistance of nodalization information for the previously developed inputs for CATHENA, TUF, and TRACE. The nodalization is presented in Figure 3. It consists of two heating channels HS9 and HS14, 4 headers, 2 steam generators U-tube sections, two pumps and four inlet/outlet feeders. The secondary side consists of preheaters, boiler evaporation space, separators dome and corresponding steam lines. The break model is attached to header 8. Heat structures are set to channel fuel sections separately modelling POWER elements for each channel, [9], [10].

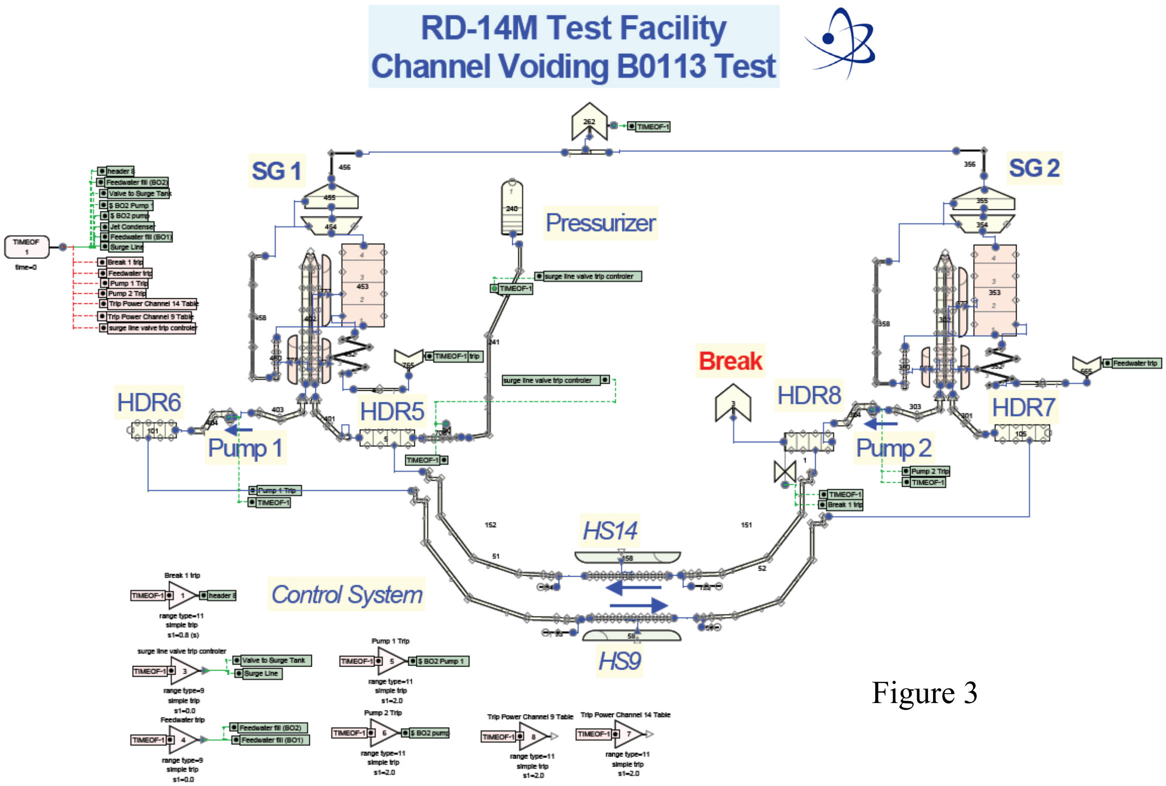

Figure 3

The pressurizer and its surge line are also represented even though the pressurizer is isolated before the break discharge is initiated. This is because the pressurizer is used in this TRACE assessment for achieving a proper steady state. Control models are based on a set of TRIPs mainly to control initial and boundary conditions. The trips are also presented in the nodalization diagram.

4.2 Initial and boundary conditions

Initial conditions were set up using the standard steady state procedure (SSP) and assumptions of loop initial thermalhydraulic parameters. Besides the well documented test conditions, there were additional parameters that allowed some freedom to set up initial conditions close to those measured in the test. Because of the preliminary character of this exercise, all the possible measurements values were not systematically checked with those obtained with the steady state code assessment. It is strongly recommended that these be evaluated in any future phase of this work. From the regulatory point of view, initial values should be set in accordance with developed criteria for acceptance of steady state. Criteria could be based on, mathematically speaking, an optimization problem solution with a criterion of a maximum accuracy vector for the selected thermalhydraulic parameters set for this test.

Boundary conditions were set in the standard way describing activation and changes to the boundary parameters as a function of time (table form). The following set of boundary conditions were specified: break opening time and time function, 2 pump speed functions of time, 2 channel power functions (representing power pulse and shutdown), 2 feed water flow function and pressurizer isolation function.

The steady state was obtained by separate calculation(s). As an illustration of the assessment presented here, the following table provides comparison between measurement and the TRACE obtained steady state:

Pressure [kPa(a)]	***Measurement***	***TRACE***
Header 5	10111.0	10070.1
Header 6	10989.9	11120.7
Header 7	10091.0	9985.1
Header 8	10963.6	11032.8
Coolant Temperature (℃)		
Header 5	299.0	300.6
Header 6	251.6	252.6
Header 7	297.4	299.0
Header 8	255.3	254.0

4.3 Test B0113 - code simulation results

For this test, RD-14M data from ~240 measurement channels were collected. Comparison between measurement and TRACE code prediction is given for the key parameter: void fraction in channel 14 as measured by neutron scatterometer. Additionally, pressure was compared in the

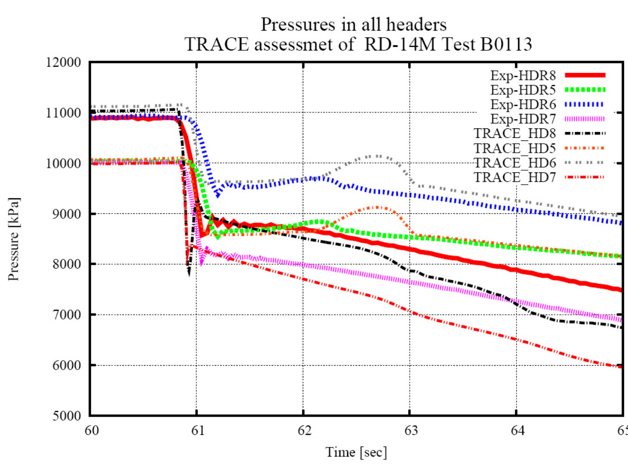

Figure 4

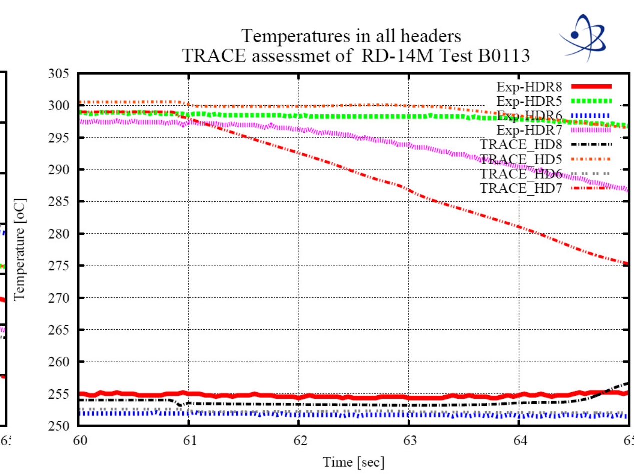

Figure 5

four headers, as cornerstone components that can indicate the integral character of the facility transient. Headers temperatures and break mass flow rate are also given. It should be again noted that break mass flow rate was not measured in the test and the TRACE calculated mass flow rate was compared to break flow calculated by CATHENA and TUF.

Figure 4 presents a comparison between pressures in all four headers measurements with the TRACE prediction. All results have produced correct trends and the TRACE calculation enveloped the result space by the calculated pressures in HD6 and HD7. The header pressures recovery and their maximum is captured but with a delay and a higher maximum. The depressurization rate in the subcooled phase, calculated with TRACE is steeper than obtained

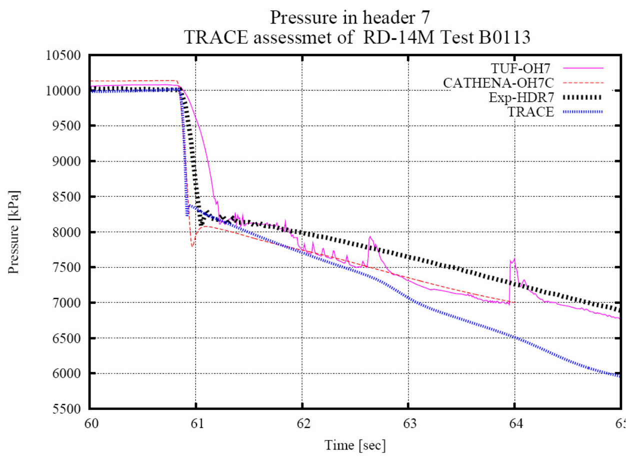

Figure 6

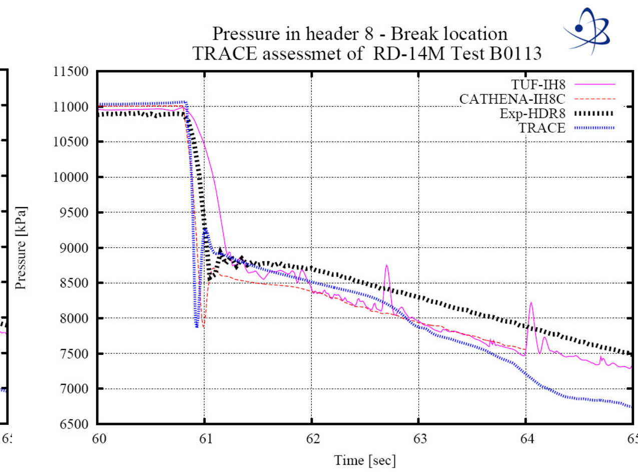

Figure 7

from the test. Figure 5 compares temperatures in the headers. TRACE produces correct predictions. Only the calculated temperature in header 7 dropped at a higher rate than measured temperature.

Figure 6 presents pressures in header 7. TRACE prediction is good during the first transient period. During the two-phase discharge period the pressure is under predicted and divergent related to the measurement. Figure 7 presents a comparison of the pressure in header 8 (break location) with pressure measurement and calculation done with CATHENA and TUF. It is

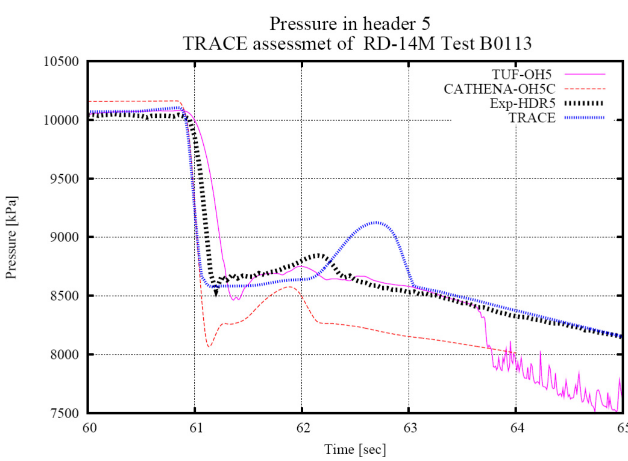

Figure 8

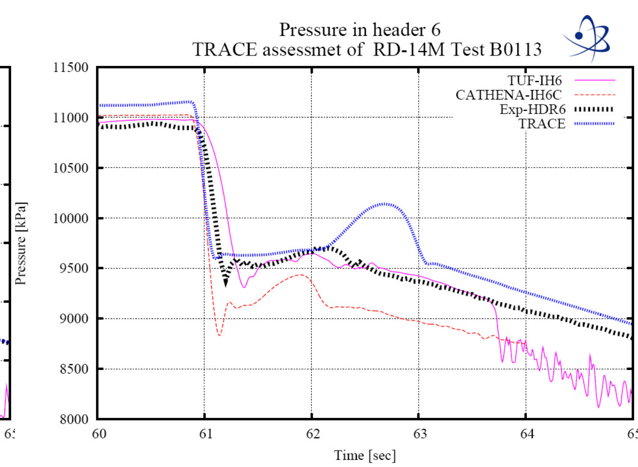

Figure 9

evident that TRACE, after a phase of subcooled depressurization, recovers to the pressure level obtained by the measurement. All three codes under predict pressures in headers 7 and 8. TRACE pressures diverged and reached a lower point at the end of observation period. This could be an indicator that break flow is over predicted. It should be noted again that a standard model for break flow and default choking coefficients were used.

Figure 8 and Figure 9 present TRACE pressures in headers 5 and 6 compared with measurements and also with CATHENA and TUF. The pressures obtained by TRACE are well predicted with exception of the subcooled depressurization phase where it is under predicted similar to CATHENA. All three codes reproduced the pressure bell shape recovery as a trend but with different time and peak pressure levels.

During the two phase discharge the TRACE prediction is very good and pressure is practically identical to the measured at the end of the observation period for header 5 while in the header 6, TRACE calculates pressure with a small bias.

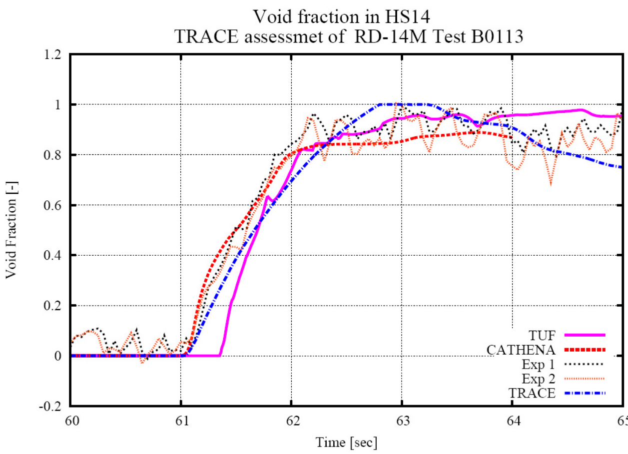

Figure 10

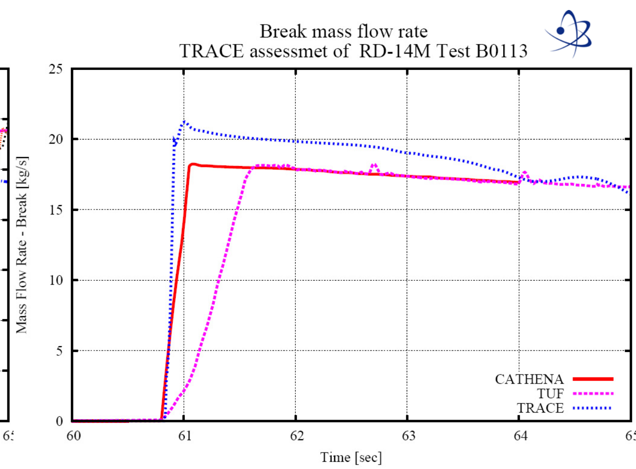

Figure 11

Void fraction in the channel HS14, the key parameter of this test exercise, is presented in Figure 10. The calculated result is compared with two channels of measurements and predictions obtained by TUF and CATHENA. In general all three codes prediction is good. The TRACE prediction is between the CATHENA and TUF calculations. The void is generated early, like in the CATHENA case, but later the voiding generation rate is not as intensive as in CATHENA. It is interesting to note that TRACE shows full voiding of the channel (~62.8 sec), while two other codes do not.

The last diagram, Figure 11, presents break flow rate calculated by TRACE compared with CATHENA and TUF. It is clear that TRACE calculates slightly higher break discharge flow than other two codes. Also it is evident that all three codes produced different discharge flow change in the initial phase of the transient. It should be noted that the opening valve time function is different for CATHENA and TUF and it was set by industry in their validation exercise. The TRACE code used a linear function of valve opening, the same as in the CATHENA assessment. Also it is interesting that CATHENA and TUF produced identical break discharge flow after full break flow area is reached.

5. Conclusion

The TRACE code assessment of the channel voiding presents an additional valuable exercise in the process of closing GAI 0G01. It provides an additional comparison for assessment of computer codes ability to model complex two phase behaviour important for CANDU reactor safety. From the point of view of the regulator, such an assessment has added to the ability of the organization for independent assessment and evaluation of safety issues.

This assessment contributes directly to TRACE validation and indirectly to validation of CATHENA and TUF code.

In a global assessment level, all three codes presented satisfactory performance in simulating RD-14M test B0113. That means the codes results have captured void fraction behaviour as the key parameter.

It is also noted that pressures in the headers in the first phase of transient when system was subcooled, were not as accurate as expected. The TRACE and CATHENA codes under predicted the pressures while TUF over predicted the pressure. It can be also concluded that accuracy of the generated void is dependent on discharge opening time function.

It is recommended that a more comprehensive assessment of the steady-state and transient be performed using more available experimental data as practical. This would be to investigate the accuracy of other variables and to investigate the reasons for any inaccuracies that may exist while the key calculated variables demonstrate high levels of accuracy.

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