LEI QUENCH TESTS MODELLING EXPERIENCE AND THE PRELIMINARY MODELLING RESULTS OF QUENCH-20 TEST USING RELAP/SCDAPSIM

Elsalamouny N.*, Kaliatka T., Kaliatka A.
*Author for correspondence
Laboratory of Nuclear Installation Safety,
Lithuanian Energy Institute,
Kaunas,
Lithuania,
E-mail: noura.elsalamouny@lei.lt

ABSTRACT

Many international programs and experiments were conducted to investigate the phenomenon of hydrogen generation during severe accidents such as QUENCH program. QUENCH-20 test was conducted as the latest test to investigate the hydrogen production phenomenon, relocation of damaged fuel rods, and boron carbide (B_4C) reactions in BWR.

This paper presents the LEI QUENCH tests modelling experience and based on gained experience development of numerical model for QUENCH-20 test using severe accident code RELAP/SCDAPSIM. Numerical analysis was provided using mod3.4 and mod3.6 code versions. The preliminary calculation results of the first attempt are presented for total hydrogen generation and cladding temperatures. The performed modelling using RELAP/SCDAPSIM code results demonstrated the advantages of mod3.6 code version, comparing to the mod3.4 version. However, the modelling of a BWR control blades still remains challenging.

INTRODUCTION

During an accident in a nuclear power plant, even after stopping the chain reaction in the nuclear reactor, it is very important to ensure adequate cooling of the reactor core. The reactor core consists of nuclear fuel assemblies assembled from fuel rods and control rods which are different for PWRs and BWRs. Due to the release of residual heat in the nuclear fuel assemblies, the failure of the main emergency cooling systems can lead to a severe accident with severe damage or melting of the nuclear fuel rods. The consequences of such an accident can be as severe as those at the Chernobyl or Fukushima NPPs. To avoid such consequences, cooling the core by flooding it with water is necessary. However, when the water cools the overheated fuel rods, the supplied water causes an intense exothermic oxidation reaction of the fuel cladding made from zirconium alloy. This leads to the release of fission products from the fuel rods into the environment and the hydrogen gas generated as a product of the steam – zirconium oxidation reaction.

The occurrence of the last severe accident in Fukushima Daiichi NPP increases the need for the review of safety regulation in Nuclear Power Plants (NPP), especially for the development of Severe Accident Management (SAMG) guidelines. Usually the computer codes, developed for the severe accident analysis (such as ASTEC, RELAP/SCDAPSIM and others), are used for the SAMG preparation. However, the validation of computer codes for all severe accident phenomena should be performed in advance. In this paper the LEI experience in validation of quenching of overheated nuclear fuel assembly phenomena is presented. Such phenomenon was selected, because the most important accident management measure during a severe accident is the injection of water to cool down the reactor core.

Hence, many international experimental programs and experiments were conducted such as CORA, QUENCH, etc. The QUENCH experimental facility is located at the Forschungszentrum Karlsruhe (Karlsruhe Research Center). Schematic view of the facility is presented in **Figure 1**. The focus of the QUENCH program is the investigation of typical light water reactors fuel assembly response during the flooding of the uncovered core with cold water [1]. According to such conditions, the oxidation of heated fuel simulators in a steam atmosphere causes the phenomenon of hydrogen generation. As hydrogen is considered a flammable gas, its rapid increase could lead to an explosion.

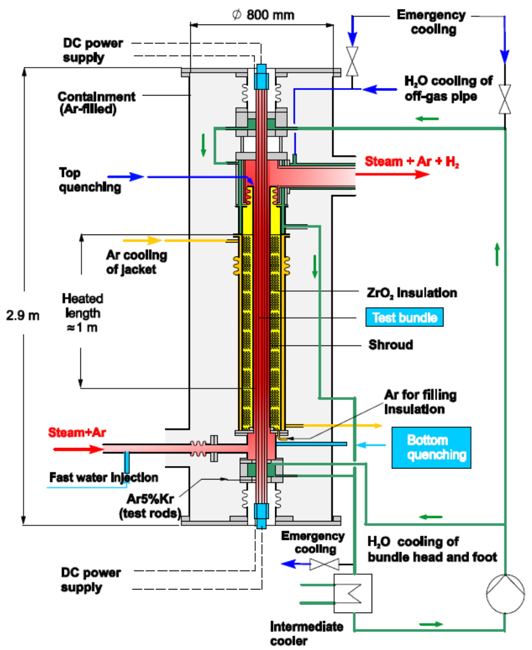

Figure 1 QUENCH test facility

Scientists of LEI have experience in modelling different QUENCH tests using severe accident computer codes ASTEC and RELAP/SCDAPSIM:

- Modelled QUENCH-03 and 06 tests [2, 3]. These tests are related to the PWR fuel bundle with the oxidation phase in the water steam ambient. This situation corresponds to the severe accident conditions in the reactor core.
- Modelled QUENCH-10 and 18 tests [6, 7, 8, 9]. The specific of these tests is the air ingress before the quenching phase. This corresponds to the severe accident conditions in the reactor core when the reactor cavity is not intact, or the processes in the spent fuel pools during loss of coolant.

QUENCH-20 test [4, 5] was conducted as the latest test to investigate the hydrogen production phenomenon, relocation of damaged fuel rods, and boron carbide (B_4C) reactions in BWR. The QUENCH-20 test was equipped with a quarter of the fuel bundle SEVA-96 Optima-2 including the absorber blades part and fuel and water channel boxes to study their oxidations and degradation under the quenching conditions.

In this article the experience gained from the previous modelled QUENCH tests is used for the development of the numerical model for the QUENCH-20 test.

Comparison of QUENCH-06 and QUENCH-20 tests

Based on the QUENCH matrix [1] some similarities could be found between QUENCH-06 and QUENCH-20 tests (Table1), specifically for the boundary conditions, power distribution, and operational test phases.

Table 1 similarities of QUENCH-06 and QUENCH-20 tests.

	QUENCH-06	**QUENCH-20**
Steam flow rate (g/s)	3 g/s	3 g/s
Argon flow rate (g/s)	3 g/s	3 g/s
Water injection (g/s)	40 g/s	50 g/s
Peak Power kW	18.2 kW	18.2 kW
Test phases	Pre-oxidation, Transient, Quenching	Pre-oxidation, Transient, Quenching

However, despite similarities in QUENCH-6 and QUENCH-20, these tests have significant difference – test bundle. The QUENCH-06 test bundle was constructed to investigate processes in PWR type reactors. Bundle consists of 20 heated rods, 1 unheated central rod (which is used for the measurement's devices and sensors or as a control rod) and four corner rods (Figure 2). All 20 fuel rod simulators have a total length of ~2.5 m, and the electrically heated length is about 1 m length.

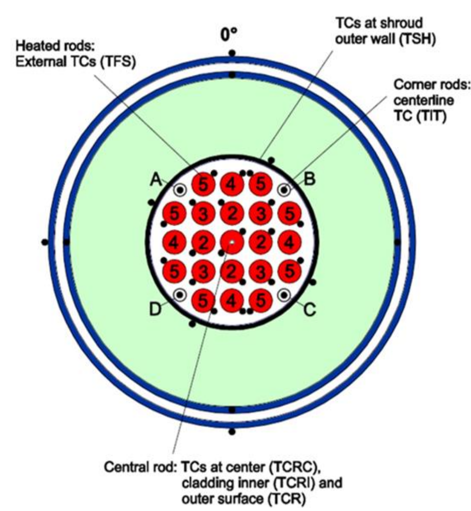

Figure 2 QUENCH-06 test bundle

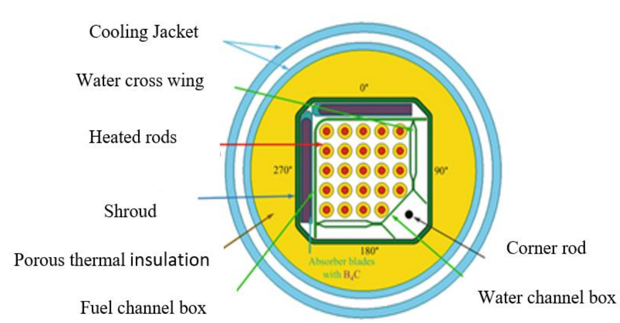

Figure 3 QUENCH-20 test bundle

The QUENCH-20 test bundle is quarter of the SEVA-96 Optima-2 BWR fuel bundle (**Figure**). It consists of 25 fuel rods simulators, water channel box, fuel channel box, absorber blades filled with B_4C pins, power supply, electrical tungsten heaters, water, and steam supply systems. The rod cladding was identical to light water reactors fuel rods` cladding material (Zricaloy-4). 24 heated fuel rods simulators up to 1.024 m in length and one unheated rod located in the corner of the bundle.

The QUENCH-20 test consists of 3 operational phases (**Figure 4**), which represent the calculation domain of the test analysis [4]:

- The pre-oxidation phase is the first operational phase that occurs when the temperature was kept constant up to the time at which the maximum oxide layer reached the experiment designed value.
- The transient phase is the second operational phase that occurs when the temperature increased up to the experiment-designed value for the onset of the quenching phase.
- Quenching phase is the last operational phase that occurs when the steam supply was stopped and water was added, simulating the reflood.

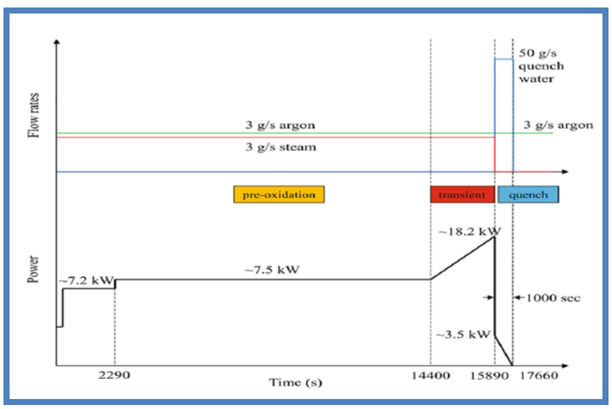

Figure 4 QUENCH-20 power and flow distributions

The consequences of reflooding, oxidation and hydrogen generation are more severe in the Boling Water Reactors than in Pressurized Water Reactors. The higher concentration of zirconium and stainless steel in the core of BWR compared with the PWRs increases the probability of producing a bigger amount of hydrogen and higher energy release rate in case of a severe accident scenario. In addition, it is necessary to evaluate the oxidation of boron carbide and its reactions, that is also a source of hydrogen generation, energy release, and melting relocations.

QUENCH-20 MODEL DEVELOPMENT

A new numerical model has been developed for QUENCH-20 test using severe accident code RELAP/SCDAPSIM. This computer code is an integral severe accident code designed to make a simulation for the overall reactor coolant system thermal-hydraulic response and core behaviour under normal operating conditions or under design basis or severe accident conditions. The RELAP/SCDAPSIM [10, 11] code includes two different parts.

1. The RELAP part, which is used to perform the calculations of the overall reactor coolant system thermal-hydraulic response (temperatures, pressure distribution, etc.), control system behaviour (changing of the operating power settings, etc.), reactor kinetics, and the behaviour of special reactor system components such as valves and pumps.
2. The SCDAP part, which is used to perform the calculation of the core behaviour and vessel structures under normal and severe accident conditions. The SCDAP part also includes models to simulate the later stages of severe accidents. These SCDAP models are invoked automatically by the code.

The development of the QUENCH-20 model is based on LEI experience on modelling QUENCH-03, 06 [2, 3], 10 [6, 7] & 18 [8, 9] using RELAP/SCDAPSIM.

Figure 5 shows the general RELAP/SCDAPSIM nodalization scheme which was used for modelling the QUENCH test facility (QUENCH-03, 06, 10 and 18 tests). The space between heated rods and the outer cooling loop of the QUENCH test facility was modelled using RELAP components: pipe, time-dependent volumes and junctions, single junctions, and branches. In addition, a time-dependent volume for the air ingression was connected to branch 007 in the case of QUENCH-10 and 18 tests [8, 9]. The electrically-heated rod simulators and surrounded shroud were modelled using SCDAP components (FUEL, CORA, and SHROUD), which were in total 5 components.

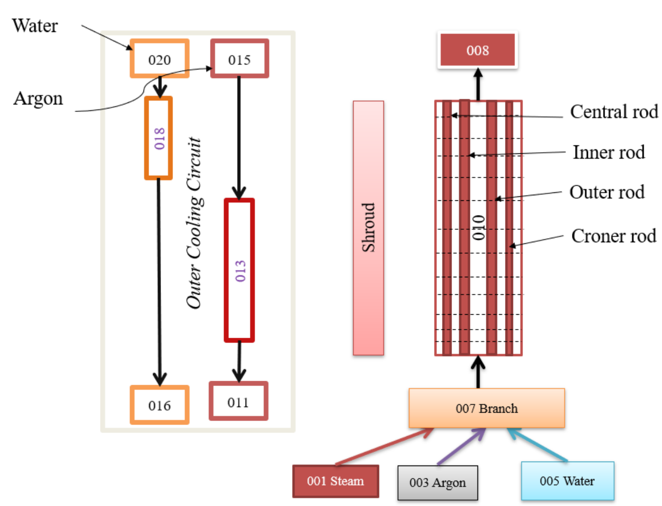

Figure 5 Nodalization scheme of QUENCH-06 test

As it was mentioned, QUENCH-06 and QUENCH-20 have similarities in boundary conditions, power distribution, and operational test phases. Thus, for the first attempt, it was decided to use the model developed for QUENCH-6 with some adaptations for QUENCH-20 test bundle.

RELAP part of the model was left almost without the modifications, only component 010 was modified according to the bundle geometry of QUENCH-20 test (**Figure 6**).

For the SCDAP part the structure was kept as for QUENCH-06, only updating geometrical data according bundle specifics. 5 SCDAP components (**Figure 7**) were used:

- Component 1: CORA component (1 heated fuel rod simulator).
- Component 2: CORA component (9 heated fuel rod simulators).
- Component 3: CORA component (14 heated fuel rod simulators).
- Component 4: FUEL component (corner rod).
- Component 5: SHROUD component (shroud with isolation).

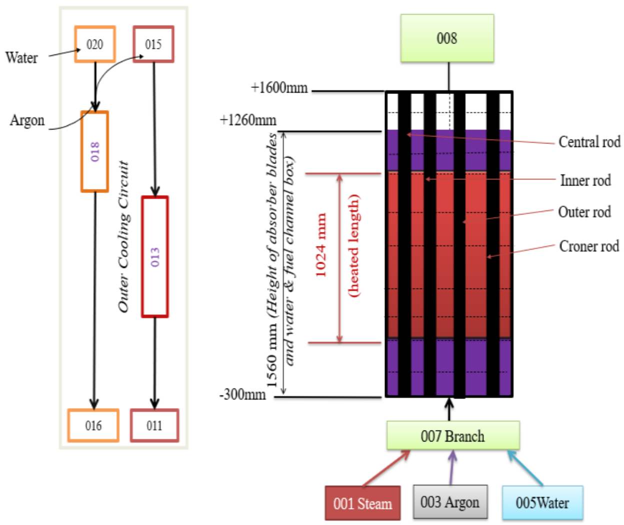

Figure 6 Nodalization scheme of QUENCH-20 test

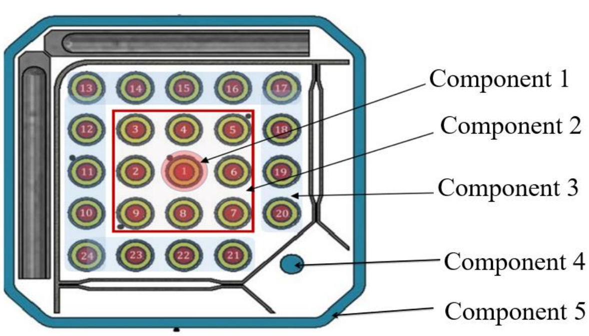

Figure 7 QUENCH-20 test bundle geometry

PRELIMINARY RESULTS FOR MODELLING QUENCH-20 TEST

The QUENCH-20 test was modelled using two different RELAP/SCDAPSIM code versions 3.4 and 3.6. The new version RELAP/SCDAPSIM 3.6 has many improvements and models in the SCDAP part. The new modelling options for SCDAP part are [14, 15, 16]:

- improvements of fuel gap conductance model;
- improvements in the model of the electrically heated rod simulator;
- shroud model improvements.
- models to treat the influence of air ingression.

There are some modifications, which have been added to the electrically heated fuel rod simulator model and applied to the new version RELAP/SCDAPSIM 3.6 [14, 15, 16]:

- improvements and consideration of inner gap between annular pellet and heater;
- improved an option to use the measurement of total electrical resistance;
- the ability to do setting for a different constant resistance value at each node.

Four different cases were performed using RELAP/SCDAPSIM code versions 3.4 and 3.6, to investigate the calculation results of QUENCH-20 test:

- **Case1**: calculations performed using RELAP/SCDAPSIM mod 3.4.
- **Case2**: calculations performed using RELAP/SCDAPSIM mod 3.6.
- **Case3**: calculations performed using RELAP/SCDAPSIM mod 3.6. In addition to the Cathcart-Pawel oxidation model, the Urbanic-Heidrick model could be applied for modelling of the oxidation in steam environment. The last model (which is optional) gives better understanding of the cladding degradation process [17].
- **Case4**: calculations performed using RELAP/SCDAPSIM Mode 3.6, with activated improvements which allows to consider the inner gap between annular pellet and heater, use measurement of total electrical resistance, and to have the ability to do setting for a different constant resistance value at each node. [15]

The calculation results for total hydrogen generation (**Figure 8**) showed high oxidation and high amounts of hydrogen generation for Case1, Case2, and Case4 compared with the experimental data. As seen from Figure 8, the calculation results of the amount of produced hydrogen in Cases 1, 2 and 4 are more than 2 times higher compared to the experiment (0.0575 kg). However, Case3 showed a very small amount of hydrogen compared with experimental data - less than 2 times than the experimental measurements.

Figure 9 presents the cladding temperature of the four cases at 950 mm elevation with the experimental data. As shown in Figure 9, Case1, Case2, and Case3 have slightly lower temperature values compared with the experimental data measurements. The calculated cladding temperature values in Case4 are slightly higher than the experimental data during peroxidation and transient phases. During the quenching phase, the cladding temperature at 950mm for Case4 is significantly higher than the experimental measurements.

The most specific for QUENCH-20 test with BWR fuel assembly is the existence of absorber blades (B_4C), thus it is needed to consider in the modelling. RELAP/SCDAPSIM have a special BWR Control Blade/Channel Box Component which could be used for the modelling of QUENCH-20 test. According to the developers a new model of B_4C model is developed, that allows to select Ag-In-Cd or B_4C. However, in this work this model was not tested, because in the current code version this model is not activated. Other option is to use second shroud component as stainless steel, but in that case the B_4C material properties will be not considered.

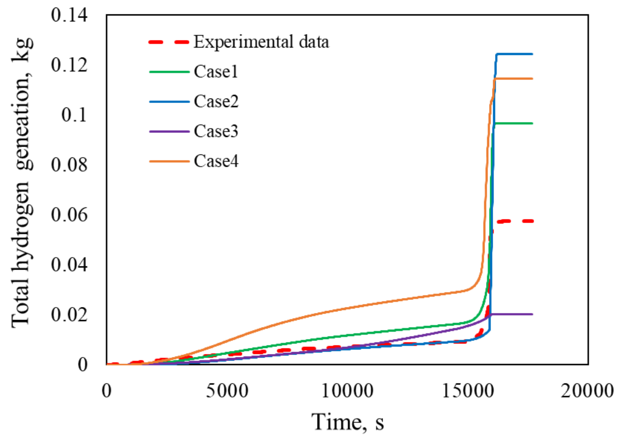

Figure 8 Total hydrogen generation

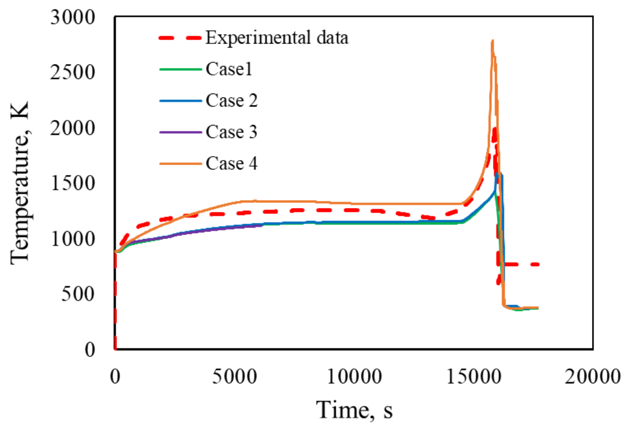

Figure 9 cladding temperature at 950mm

Second nodalization scheme was developed to check these cases (**Figure10**).

- **Case5**: control blades were modelled as BWR Control Blade/Channel Box component.
- **Case6**: control blades were modelled as shroud component. The shroud material is stainless steel.

Figure 11 shows the total hydrogen generation for Case5 and Case6. In case "modelling control blades as BWR control blades/channel box component", the calculated amount of hydrogen has lower values (0.033kg) compared with the measured values (0.057 kg). Analogously, Figure 12 shows low values of cladding temperature at 950 mm elevation for Case5 compared with experimental data measurements, however, Case 5 has a good agreement with the experimental data measurements. For Case6 the total hydrogen generation is in a good agreement with experimental data until quenching phase. Case 6 calculation results shows ~2 times higher calculated values, compared with the experimental data measurements (Figure 11). Also, Figure 12 showed that the cladding temperature at 950 mm for Case6 is in good agreement with experimental data comparing the Case5.

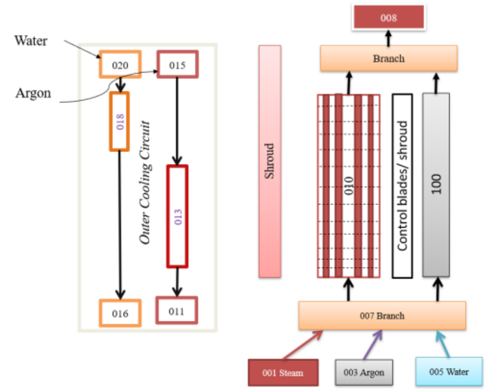

Figure 10 New nodalization scheme

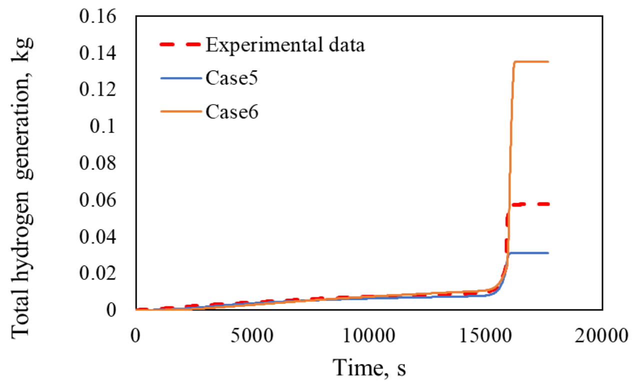

Figure 11 Total hydrogen generation.

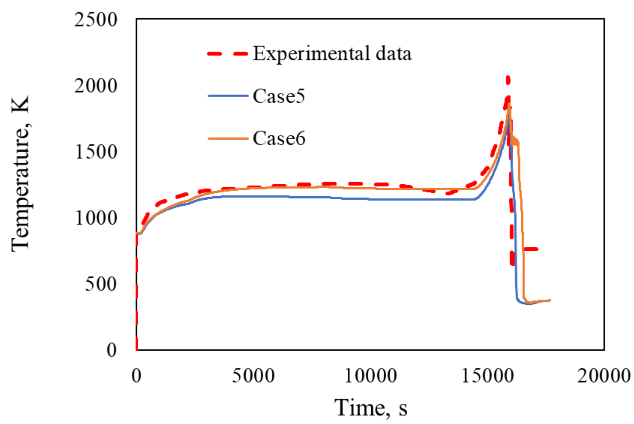

Figure 12 Cladding temperature at 950 mm elevation (first approach)

Acknowledgement

The authors would like to thank Mr. Chris Allison (General manager at Innovative System Software "ISS") and

Mr, Jonathan Birchley for sharing their knowledge and support during this work.

CONCLUSION

The geometrical arrangement of QUENCH-20 bundle test section is very challenging for modelling for these reasons:

- Severe accident codes (RELAP/SCDAPSIM, ASTEC, MELCOR, AC2, ect…) use a modelling approach based on concentric rings to simulate fuel.
- Challenge in modelling control blades and B_4C reactions with steam.
- Possible large uncertainties in calculation results.

The LEI modelling experience of QUENCH tests was used for the development of new numerical model for QUENCH-20 test. It was found from QUENCH matrix that there are similarities in boundary conditions, power distribution, test phases between QUENCH-06 and QUENCH-20. According to these similarities it was decided to use the same nodalisation scheme as QUENCH-06 test with modifying the test bundle according to QUECH-20 bundle geometry. As a first attempt, calculations of four different cases were performed using RELAP/SCDAPSIM versions mod3.4 and mod3.6. The best agreement with experimental measurements was received, when RELAP/SCDAPSIM Mode 3.6 was used with activated improvements allowing to consider the inner gap between annular pellet and heater and considering a different constant resistance value at each node. As B_4C modelling is very challenging, two different Cases were made: in first case the control blades were modelled as BWR Control Blade/Channel Box component, while in second - these control blades were modelled as shroud component. The calculations were performed using RELAP/SCDAPSIM mod3.6. First case gave lower values of hydrogen generation (~0.033kg), however, second gave 2 times higher values compared with experimental data (~0.15kg). Also, for the cladding temperature at 950 mm elevation, the calculation result of second case is in a better agreement with the experimental data then the first case.

Thus, the performed first modelling using RELAP/SCDAPSIM code results demonstrated the advantages of mod3.6, comparing to the mod3.4 version. However, the modelling of a BWR control blades still remains challenging. This work will be continued in future.

REFERENCES

1. QUENCH matrix < https://quench.forschung.kit.edu/24.php>
2. Kaliatka, T.; Kaliatka, A.; Vileiniškis, V. Best estimate approach for QUENCH-03 and QUENCH-06 Experiments // 20th International QUENCH Workshop Karlsruhe Institute of Technology, Campus North, November 11-13, 2014, p. 20
3. Kaliatka, T.; Kaliatka, A.; Vileiniškis, V.; Ušpuras, E. Modelling of QUENCH-03 and QUENCH-06 Experiments Using RELAP/SCDAPSIM and ASTEC Codes // Science and Technology of Nuclear Installations. Dx.doi.org/10.1155/2014/849480 ISSN 1687-6075. 2014. Vol. 2014, Article ID 849480, p. 1-13.
4. Murat, O.; Sanchez-Espinoza, V.; Wang, Sh.; Stuckert, J. Preliminary validation of ASTEC V2.2.b with the QUENCH-20 BWR bundle experiment, nuclear engineering and design, 2020.
5. Stuckert, J.; Peters, U.; Stegmaier, U. Results of metallographic analysis of the QUENCH-20 bundle with B4C absorber, QWS-26, KIT, 2021.
6. Kaliatka, T.; Ušpuras, E.; Allison, CH. M. Modeling of quench 10 experiments on air ingress using relap/scdapsim mod 3.5 // Proceedings of the 2017 25th International Conference on Nuclear Engineering ICONE25 July 2-6, 2017, Shanghai, China. P. 1-7
7. Kaliatka, T. Implementation of QUENCH-10 experiment modelling experience for the modelling of severe accidents in spent fuel pools // Proceedings 23rd International QUENCH Workshop Karlsruhe Institute of Technology, Germany (doi:10.5445/IR/1000076201) October 17-19, 2017. P. 366-395
8. Hollands, T.; Bals, C.; Tiborcz, L.; Beuzet, E.; Vasiliev, A.; Kaliatka, T.; Birchley, J.; Steinbrück, M. Pre-and Post-Test Simulation of the QUENCH-18 Bundle Experiment in the frame of the NUGENIA QUESA Project,2019.
9. Kaliatka, A.; Kaliatka, T.; Vileiniškis, V.; Ušpuras, E. Best estimate approach for the simulation of reactor core overheating and quenching experiments // Best Estimate Plus Uncertainty International Conference. Multi-Physics Multi-Scale Simulations with Uncertainty (BEPU 2018) May 13-18, 2018, Lucca, Italia. 12 p.
10. Allison, C.; Wagner, R. J. "RELAP5/SCDAPSIM/MOD3.2(am+) Input Manual Supplemental," Innovative Systems Soft-ware, LLC, 2020, http://www.relap.com/.
11. Kaliatka, T. Modelling of quench experiment using RELAP/SCDAPSIM code // 9th annual conference of young scientists on energy issues CYSENI 2012: international conference, Kaunas, Lithuania, 24-25 May 2012. Kaunas: LEI, 2012. ISSN 1822-7554, p. 668-677.
12. RELAP5 Code Development Team, 2001. RELAP5/MOD3.3 Code Manual. Vol 1–8, NUREG/CR-5535/Rev1.
13. SCDAP/RELAP5 Development Team, 1981. SCDAP/RELAP5/MOD3.2 Code Manual, Vol. 1–5, NUREG/CR-6150, INEL-96/0422.
14. Madokoro, H.; Sato,I. Estimation of the core degradation and relocation at the Fukushima Daiichi Nuclear Power Station Unit 2 based on RELAP/SCDAPSIM analysis, Nuclear Engineering and Design, 2020.
15. Madokoro, H.; Okamoto, K.; Allison, C.; Siefken, L.; Hohorst, J.; Hagen, S. Assessment of RELAP/SCDAPSIM/MOD3.5 against the BWR core degradation experiment CORA-17, Conference: The 10th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation, and Safety (NUTHOS-10) At Okinawa, Japan 2014.
16. Allison, C.; Hohorst, J.; Allison, B.; Konjarek, D.; Bajs. T. Preliminary Assessment of the Possible BWR Core/Vessel Damage States for Fukushima Daiichi Station Blackout Scenarios Using RELAP/SCDAPSIM, Hindawi Publishing Corporation Science and Technology of Nuclear Installations, 2012.
17. Birchely, J.; Fernandez-Moguel, L. Simulation of air oxidation during reactor accident sequence: part 1-phenomenology and model development, *Annals of Nuclear engineering*, 2012.