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Quench Analysis Using RELAP/SCDAPSIM/MOD3.5

G. Gerova[1], **C. Allison**[2]

[1] *Department of Thermal and Nuclear Power Engineering, Technical University of Sofia, 8 Kliment Ohridski Blvd., Sofia 1000, Bulgaria*

[2] *Innovative Systems Software LLC, 3585 Briar Creek Ln., Ammon, ID 83406, USA*

Abstract. To perform nuclear reactor safety analysis related to accidental conditions that can lead to severe accidents it's necessary to perform some experiments before that. QUENCH-06 is exactly that kind of experiment. It is related with the consequences that can arise from severe accidents and subsequent water injection in Light Water Reactors. The main goal of the present paper is to compare measured by the QUENCH-06 experiment and calculated by the RELAP/SCDAPSIM/MOD3.5 temperatures of twelve rods and shroud at different elevations of the test bundle.

Keywords: QUENCH-06, RELAP/SCDAPSIM/MOD3.5, severe accident, water injection.

1 Introduction

Termination of transients which could lead to severe accidents in Light Water Reactors (LWR) is one of the most important accident management measure. This measure is realized by water injection to cool down the uncovered core [1,2]. Analysis that was performed after the Three Mile Island Unit 2 accident, as well as results of out-of-pile and in-pile experiments like CORA and LOFT respectively, have shown that before the water succeeds in cooling the fuel there will be a sharp temperature increase, hydrogen production, and fission products release. These phenomena are result from Zircaloy cladding enhanced oxidation [2,3].

Moreover, water injection to the uncovered degraded core of nuclear reactor or so called quenching is considered the worst case accidental scenario regarding hydrogen release in the containment building. Furthermore, the hydrogen that is generated due to Zircaloy claddings oxidation by the steam occurs when the core temperature exceeds 1000 K [4]. Performing both in-vessel and ex-vessel safety analyses is necessary to prove that hydrogen release rates and its total amount should not exceed limits for the nuclear power plant that is analyzed. Therefore, hydrogen amount must be known in order to design adequate measures to mitigate the effects of accidents that may arise from its presence.

However, there are other factors which play an important role in the safety analyses. The physicochemical phenomena related to the hydrogen release are still not well enough understood as well as presently available Zircaloy-Steam oxidation correlation are not sufficient to determine increased hydrogen generation during the quenching phase. Also there are certain phenomena that lead to enhanced oxidation of fuel claddings materials and therefore to hydrogen production. These phenomena are melt oxidation, steam starvation conditions, and crack surface oxidation [3].

2 International Standard Problem Program and QUENCH-06 Experiment Relationship

International Standard Problem(ISP) Program is a series of comparative exercises that are related with the predictions or recalculations of given physical problem with different best-estimated computer codes. Predictions obtained by these codes are compared with each other and above all with the results of a carefully specified experimental study. International Standard Problems are carried out as two different type problems – "open" and "blind". In an open Standard Problem exercise the results obtained of the experiment are available to the participants before performing the calculations. On the other hand, in a blind Standard Problem exercise the experimental results are locked until the calculation results are made available for comparison [5].

As a part of this program, International Standard Problem No.45 is dedicated to study the behavior of heat-up and delayed reflood of fuel elements in nuclear reactor during a hypothetical accident. ISP No.45 is related to the out-of-pile bundle experiment QUENCH-06 where special attention was paid to the hydrogen generation. The ISP No.45 is initiated to extend the database for hydrogen generation, material behavior and bundle degradation during water injection into uncovered core of LWRs as well as to identify the key phenomena for such situations.

The main objective of the QUENCH-06 experiment is to investigate fuel rod bundle behavior up to and during reflood/quench conditions without severe fuel rod damage prior to reflood initiation. In particular, the conditions of the Design Basic Accident plus an additional failure, leading to delayed activation of Emergency Core Cooling system, were investigated up to total reflood of the heated section of the bundle with water, starting with conditions representative for normal reactor operation [4].

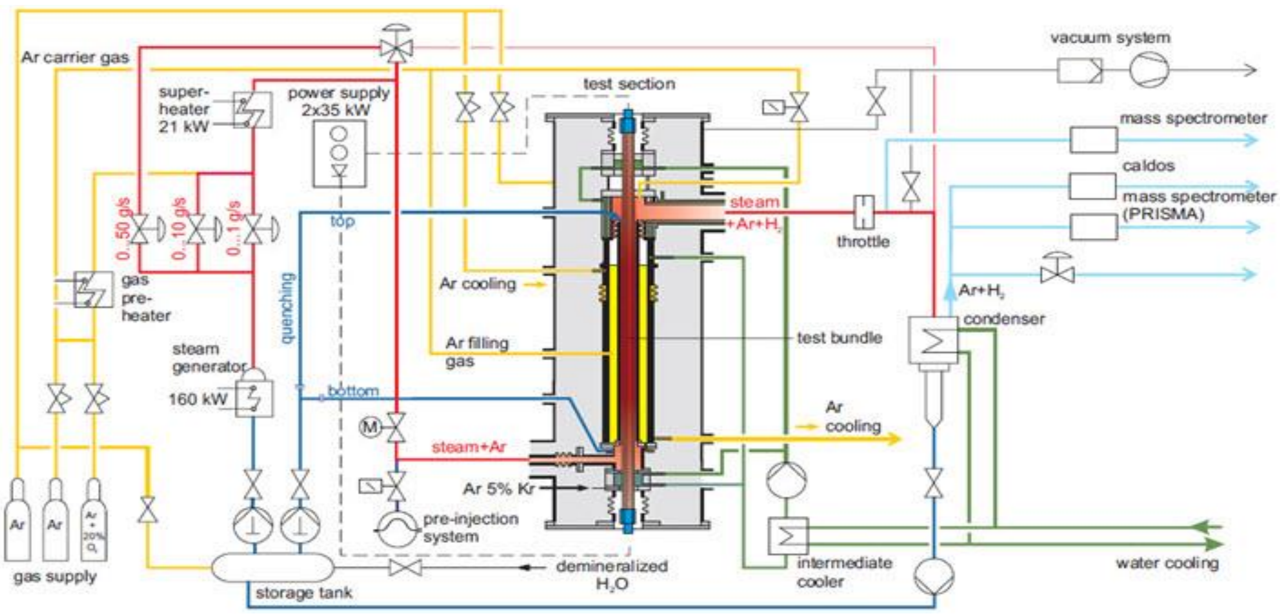

Figure 1. Simplified flow diagram of the QUENCH test facility [3,4].

3 Short Description of QUENCH facility and QUENCH-06 Test Bundle

The QUENCH-06 experiment is performed in the QUENCH facility at Karlsruhe Institute of Technology (former Forschungszentrum Karlsruhe) on 13 December, 2000 [3] and is performed as a blind Standard Problem, i.e. only the experimental initial conditions and boundary conditions are given to the participants to perform their calculations [6]. The purpose of this experiment is to investigate the behavior of pre-oxidized LWR fuel rods when they are cooled down with water from the bottom [3].

3.1 QUENCH facility

The QUENCH test facility is composed of several subsystems – the test section that consists of 21 fuel rod's; an electric power supply for the bundle heating; water, steam and argon gas supply systems; temperature, pressure, mass flow, and hydrogen measurement devices; process control and data acquisition systems. The fuel rods in QUENCH facility are replaced by imitators or simulators with tungsten heaters with 6 mm outer diameter, surrounded by annular pellets which are made from ZrO_2 simulating the real fuel pellets, bordered by Zircaloy-4 claddings with outer diameter of 11 mm [2]. Twenty fuel rod simulators are heated electrically over a length of 1024 mm and the total heating power available is 70 kW, distributed among the two groups of heated rods with 35 kW each. The tungsten heaters are connected to electrodes made of molybdenum and copper at each end of the heater [3].

Figure 1 shows a simplified flow diagram of the QUENCH-06 test facility. The superheated steam flowing first trough the steam generator and then trough a superheater together with argon, which is used as a carrier gas for the hydrogen detection systems, enter the bundle at the bottom end. The steam that is not consumed along with the argon gas and the hydrogen that is generated by the zirconium-steam reaction flow through a water-cooled off-gas pipe from the bundle to the condenser. In this condenser the non-condensable gasses argon and hydrogen are separated from the steam. The quenching phase is started by turning off the flow of superheated steam while the argon flow rate remains unchanged as well as gas inlet position is switched to the upper plenum of the test section. At the same time, through another separate line the quench water is injected into the bottom end of the test bundle.

3.2 QUENCH-06 test bundle

In radial direction the QUENCH-06 fuel rod bundle is composed of one unheated rod located at the center, an inner ring of eight heated rods connected to one

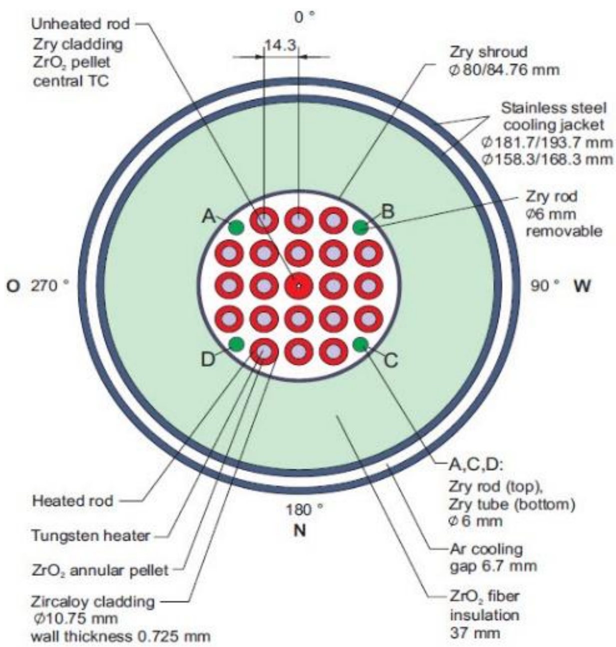

Figure 2. QUENCH-06 Fuel rod simulator bundle [3,4].

electric power supply system, an outer ring of twelve heater rods connected to a second power supply system and a set of four corner rods.

The 21 fuel rod simulators are filled with a mixture of 95% argon and 5% krypton at pressure slightly above fluid pressure in the bundle [4]. The length of the fuel rod imitators is approximately 2.5 m. Cross-section of the test bundle is depicted in Figure 2. Materials and dimensions of fuel rod simulator claddings are exactly the same as that used in Pressurized Water Reactors.

3.3 QUENCH-06 test conduct

The results obtained from the QUENCH-06 experiment are extensively documented in [3] so that only a brief description will be given here. Generally, each one of QUENCH test series consists of several different phases: heat-up, peroxidation, transient when the bundle is cooled by saturated steam, and the quenching phase when the bundle is reflooded by water. The test phases of QUENCH-06 experiment are represented in Figure 3, while Table 1 shows times (in seconds) of various events and phases of the experiment.

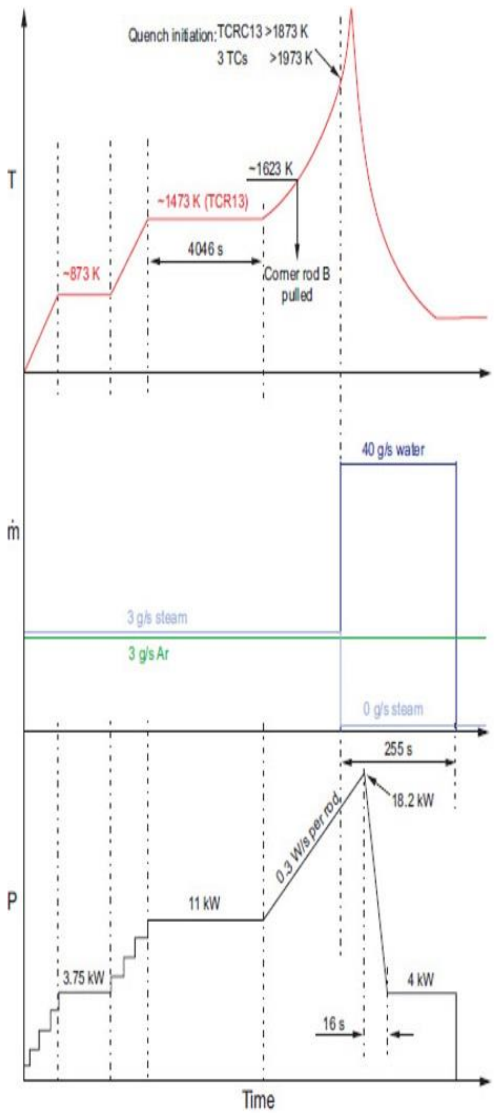

Figure 3. Test phases (schematic) [3].

3.3.1 Preparatory and heat-up phase [$0 \sim 1960$ s]

As in previous QUENCH experiments, during the preparatory phase the test bundle was heated initially by a series of stepwise increases of electrical powerof about 4 kW. This electrical power increase leads to a temperature increases from room temperature to $\sim$ 873 K in an atmosphere of argon (3 g/s) and steam (3 g/s) [3,7,8].

In the heat-up phase the bundle was ramped by stepwise increases in power up to about 11 kW to reach 1473 K (axial maximum) which is the target temperature for pre-oxidation.

3.3.2 Pre-oxidation phase [$1960 \sim 6010$ s]

The bundle was stabilized at this temperature (1473 K) for about two hours and the electrical power being approximately 4 kW [7]. This is achieved by control of the electrical power throughout the phase constant [3]. During this time the operation of the various systems was checked and shortly before the end of this phase data acquisition was started. At the end of the stabilization period the bundle

Table 1. Sequence of events and phases [1,3]

Phase	Time (s)	Event
Heat-up	0	Data acquisition start
	20	Power 4 kW. Power stepwise increase started
	1960	Reached power 10.5 kW. Max temperature 1400 K
Pre-oxidation	1960	Beginning of bundle oxidation at about 1400 K
	1965	Pre-oxidation at about 1500 K
	6010	Transient phase initiation
Transient	6010	Power transient initiation
	6620	Pull-out of corner rod (B) initiation
	6640	End of pull-out of corner rod transient
	7179	Quench phase initiation
Quenching/reflood	7179	Steam supply shut down
	7179	Onset of fast water injection
	7179	Onset of quench water injection by quench water pump
	7179	Cladding failure detection
	7179	Fast temperature drops at TFS 2/1
	7181	Zero steam mass flow rate
	7204–7205	Onset of electric power reduction from $\sim$ 18.2 kW down to $\sim$ 4 kW
	7221–7222	Decay heat level reached
	7429–7430	Onset of final power reduction quench/reflood
	7431	Quench water injection shut down
Post-reflood	7431–7437	Electric power $<$ 0.5 kW
	7435	Quench mass flow zero
	11420	Data acquisition end

was ramped by stepwise increases in power up to approximately 11 kW to reach an appropriate temperature for pre-oxidation. The heat-up rate of the bundle was 0.32 K/s between 1450 K and 1750 K. The temperature level was maintained for about 1 hour by controlling the electrical power in order to reach the desired oxide layer thickness [7]. After the bundle was heated to approximately 1500 K a pre-oxidation phase was used to establish a specific oxide layer thickness [4].

3.3.3 Transient phase [$6010 \sim 7179$ s]

During this phase, the bundle temperature was increased to the designed experimental value for the quench phase onset and the electrical power was ramped at 0.3 W/s per rod to start the transient phase in the same way as in QUENCH-05 experiment. During the transient phase at 6620 s and ~ 1606 K a corner rod was withdrawn to check and measure the amount of oxidation at that time of the experiment [3,7,8]. The analysis performed after the experiment via metallographic examination resulted show that the maximum oxide layer thickness of $\sim\ 210\ \mu$m is at 950 mm elevation [3].

During the transient phase electrical heating, and chemical power released due to the oxidation, led to a maximum cladding temperatures of $\sim\ 2200$ K. The higher temperatures would cause dissolution of the fuel pellets by the liquid zirconium with subsequent melt relocation prior to reflood. Water injection was initiated at that temperature and most of the measured temperatures dropped nearly immediately to 400 K due to fast cooling caused by water evaporation [4].

3.3.4 Reflood/quenching phase [$7179 \sim 7435$ s]

At reflood initiation cladding failure, and a little later a shroud failure, were detected. Approximately 250 s after reflood initiation the temperatures up to the level of the off-gas pipe decreased to saturation. Prior the reflood phase $\sim\ 32$ g of hydrogen were produced as well as additional 4 g during the reflood [4].

The quenching phase was initiated when pre-defined criteria similar to previous experiment were reached [5]. For the quenching sequence to begin, the following precondition is required: a minimum of three rod thermocouples should have exceed 1973 K as well as the central rod thermocouple TCRC 13 which is located is 950 mm elevation [3,8].

To initiate the quench phase, the flow of 3 g/s superheated steam was tuned off at 7181 s, the argon flow was switched over to the bundle head as well as the valve of the fast injection system was opened for 5 s allowing approximately 4 kg of quench water rapidly to fill the pipes and the lower plenum of the test section. At the same time the quench pump was started to inject water from the bottom of the test section for 255 s at a rate of ~42 g/s which corresponds to a flooding velocity of 1.4 cm/s [3,7,8].

26 s after the water injection had started (7205 s) the electrical power was reduced from 18.2 kW to 3.9 kW within 16 s. This was done to simulate decay heat levels in nuclear power reactor [3]. During the quench phase the argon injection was switched to the upper plenum to continue providing carrier gas for quantitative hydrogen detection [7,8]. The quench water and electrical power were turned off 252 s after water injection, terminating the experiment.

4 Code Version Used for the Analysis

The RELAP/SCDAPSIM is a computer code, designed to predict the overall reactor coolant system thermal hydraulic response and core behavior during normal operational conditions as well as under design basis or severe accident conditions [1,9,10].

RELAP/SCDAPSIM uses the publicly available RELAP/MOD3.3 as well as SCDAP/RELAP5/MOD3.2 models developed by the US Nuclear Regulatory Commission (NRC) in a combination with proprietary advanced programing and numerical methods, user options, and models developed in frame of the International SCDAP Development and Training Program (SDTP) [1,9]. The administrator for the SDTP program and main developer of specific models for the RELAP/SCDAPSIM is a private, limited liability company Innovative Systems Software (ISS) and their enhancements allow the code to run faster and more reliable than the original US NRC codes [9].

In RELAP/SCDAPSIM/MOD3.5 the overall thermal hydraulics of the reactor coolant system, control system behavior, reactor kinetics as well as behavior of several special reactor system components such as valves and pumps are being calculated by the RELAP5 portion of the code and its models [11]. On the other hand, the behavior of the core and vessel structure under normal as well as accident conditions are being calculated by SCDAP. SCDAP is the part of code that includes user-selectable reactor component models for LWR fuel rods, Ag-In-Cd and B4C control rods, BWRcontrol blade/channel boxes as well as electrically heated fuel rod simulators, general vessel and core structures. SCDAP also has models to treat the later stages of a severe accidents with debris and molten pool formation, debris/vessel interactions and the structural failure (creep rupture) of vessel structures[10].

5 QUENCH Facility Modeling

The QUENCH test facility and QUENCH-06 experiment was modelled using RELAP/SCDAPSIM/MOD3.5 code.

The RELAP components of the QUENCH-06 nodalization scheme (pipe, time dependent volume, time dependent junction, single junction and branch) is presented in Figure 4 while SCDAP component nodalization scheme of fuel rods imitators and surrounding shroud (axial nodes and radial spacing mesh) for the all five components that are being modeled (central unheated rod, inner and outerring of heated rods, instrumentation tubes and shroud) is presented in Figure 5.

The boundary conditions for the RELAP components and how the SCDAP components are modeled is explained below.

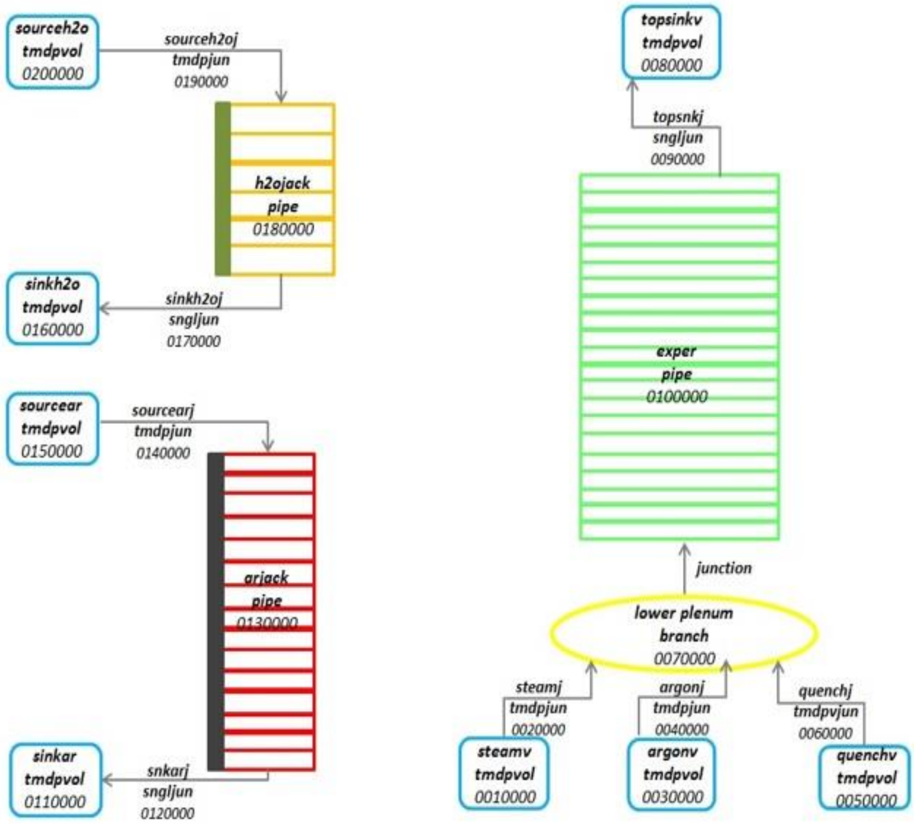

Figure 4. RELAP5 nodalization scheme

At RELAP5 portion of the code the element 0010000 (time dependent volume), 0030000 (time dependent volume), and 0050000 (time dependent volume) which representing steam, argon, and quench water inletbottom boundary conditions are time (s), pressure (Pa), temperature (K) at the bundle inlet as well as for elements 0020000 (time dependent junction), 0040000 (time dependent junction), and 0060000 (time dependent junction) boundary conditions are time (s), mass flow rate (kg/s) of steam, argon, and quench waterrespectively at the bundle inlet. Top boundary conditions of the bundle are time (s), pressure (Pa), temperature (K) for the element 0080000 (time dependent volume) and 0090000 (time dependent junction). Element 0100000 (pipe) is representing the test section.

Cooling jacket – argon section is composed of the following elements: 0110000 (sinkar) and 0150000 (sourcear) with bottom and top boundary conditions respectively as time (s), pressure (Pa), and temperature (K) as well as for elements 0120000 (snkarj) and 0140000 (sourcearj) with time (s) and mass flow rate (kg/s) as boundary conditions for element 0130000 (arjack). For the cooling jacket – water section boundary conditions are the same as those of cooling jacket – argon section.

At SCDAP part of the programComponent No.1 – one central unheated rod, is modeled as central "fuel" component that is composed of ZrO_2 pellets in the center, a gas-filled gap, and a cladding of Zircaloy. Component No.2which includes eight heated rods to simulate fuel from inner heated ring is modeled as inner "cora" component. This component is composed of tungsten heating elements in the center, ZrO_2 pellets, gas-filled gap, and cladding of Zircaloy. Component No.3 is composed from twelve heated rods simulating outer heated ring is modeled as outer "cora" component. Component No.4 is modeled as instrumentation "fuel" component representing the four rods in the corner. Component No.5 – shroud of the bundle is modeled as shroud "shroud" component, which consists of the inner Zircaloy layer, a ZrO_2 insulating layer, and an Inconel layer.

SCDAP components nodalization is different for all of five components that are being modeled. SCDAP nodalization scheme for the five elements includes 22 axial nodes for

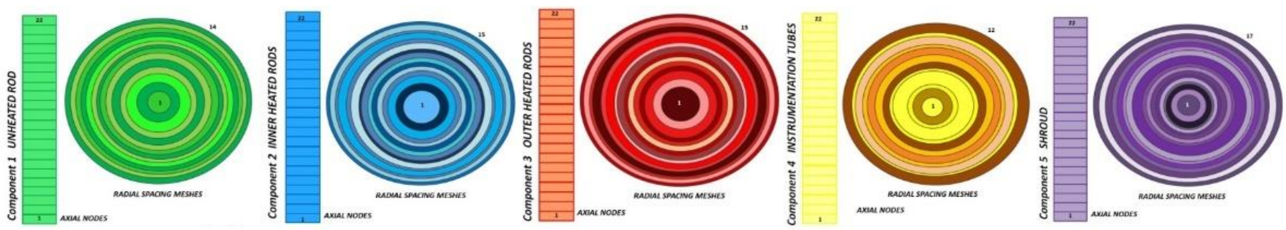

Figure 5. SCDAP nodalization scheme.

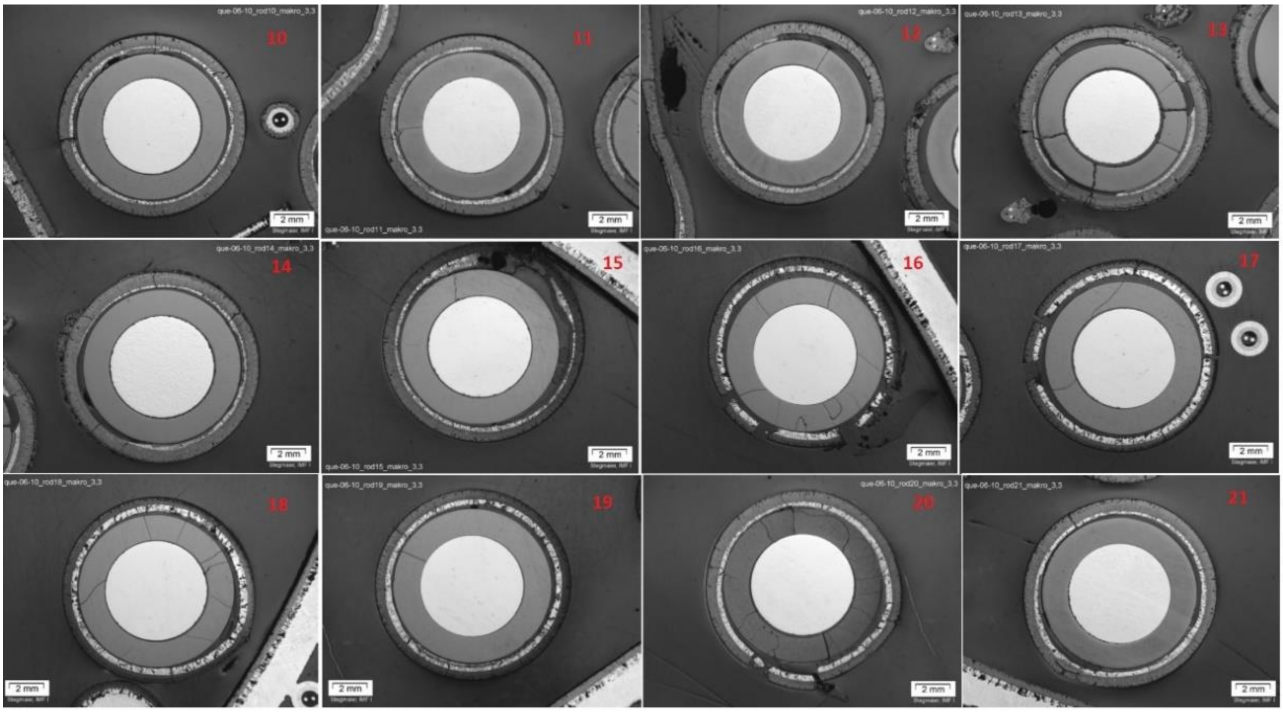

Figure 6. Cross section QUENCH-06 at 950 mm showing separately test rods 10 – 21 [3].

the all of elements and 14 radial space meshes for the Component No.1, 15radial space meshes for the Component No.2, 15 radial space meshes for the Component No.3, as well as 12 radial space meshes for the Component No.4 and 17 radial space meshes for the Component No.5 respectively.

6 RELAP5/SCDAPSIM/MOD3.5 Accuracy Assessment against QUENCH-06 Experimental Data

For the RELAP5/SCDAPSIM/MOD3.5code assessment comparison between measured by the QUENCH-06 experiment temperatures and those that are calculated by the code was done.

Figure 6 shows separately cross-sections of fuel rods 10 to 21 representing the test bundle outer ring at 950 mm elevation. For those twelve fuel rod simulators comparison is made.

In all of the graphs with *cntrlvar* are marked temperatures measured by the QUENCH-06 experiment thermocouples. Figure 10, Figure 14, and Figure 18 show bundle and fuel rods cross-sections at 550, 850, and 950 mm elevation, whereas Figure 12, Figure 16, and Figure 20 show bundle and shroud cross-sections at the same elevations. Figure 7 and Figure 8 show comparisons between measured by the experiment and calculated by the RELAP/SCDAPSIM/MOD3.5 code temperatures for outer

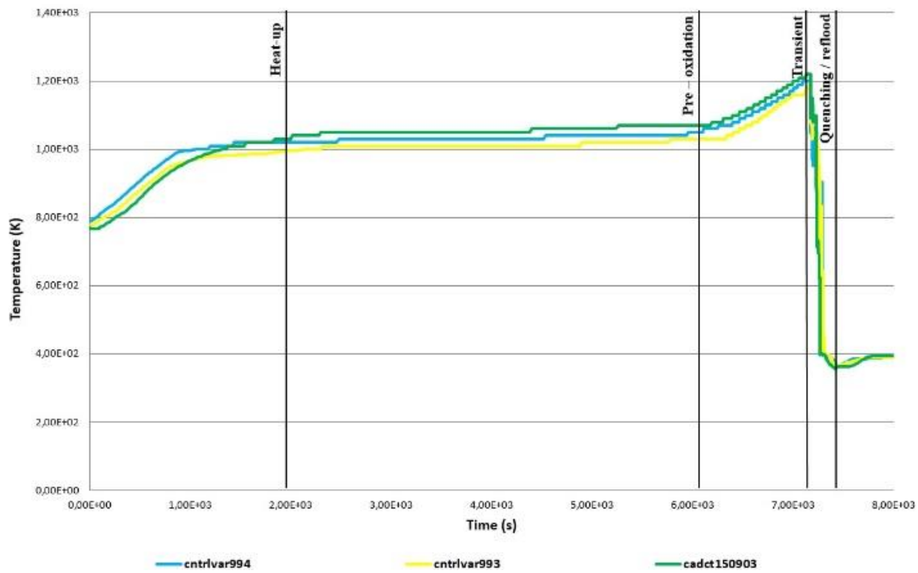

Figure 7. Measured vs. calculated cladding outer surface temperatures for outer ring rods at 350 mm elevation.

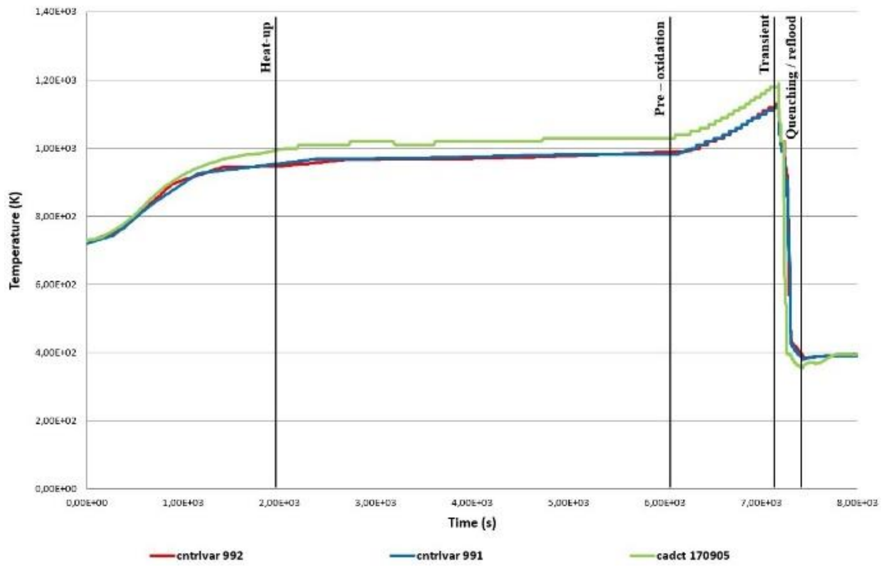

Figure 8. Measured vs. calculated shroud outer surface temperatures for outer ring rods at 350 mm elevation.

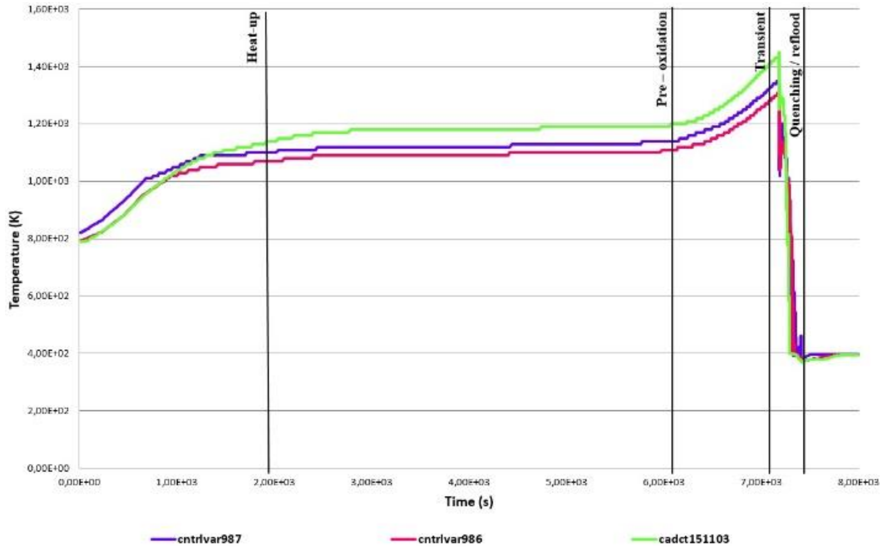

Figure 9. Measured vs. calculated cladding outer surface for outer ring rods at 550 mm elevation.

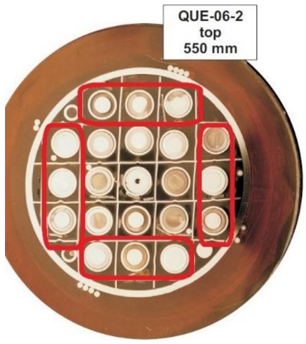

Figure 10. Bundle and fuel rods cross-section at 550 mm elevation [3].

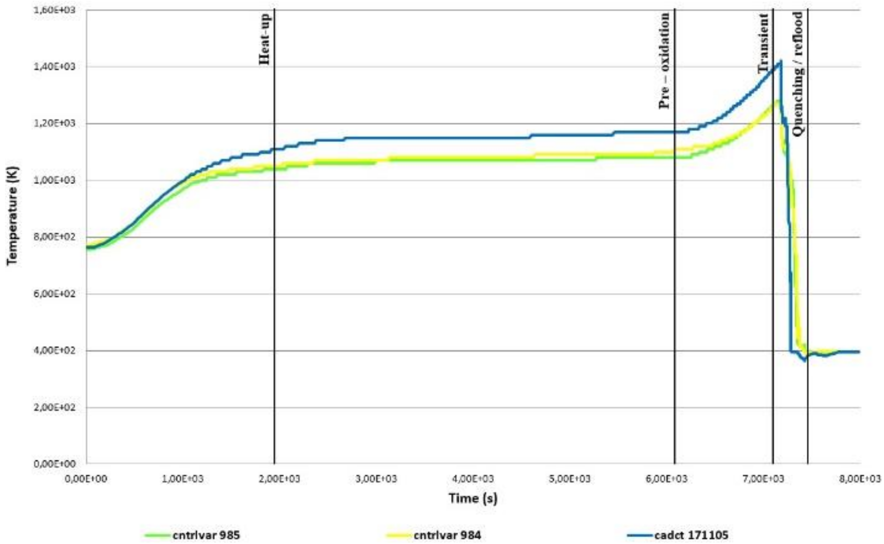

Figure 11. Measured vs. calculated shroud outer surface temperatures at 550 mm elevation.

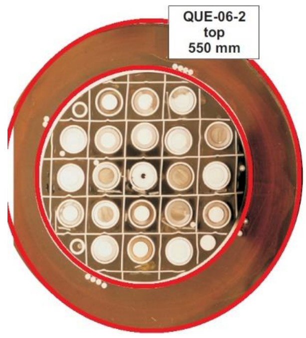

Figure 12. Bundle and shroud cross-section at 550 mm elevation [3].

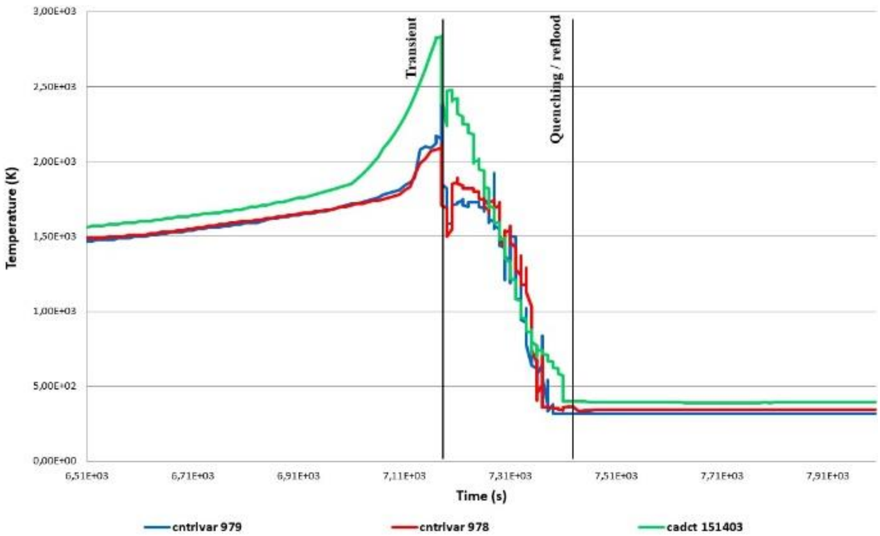

Figure 13. Measured vs. calculated cladding outer surface temperatures for outer ring rods at 850 mm elevation.

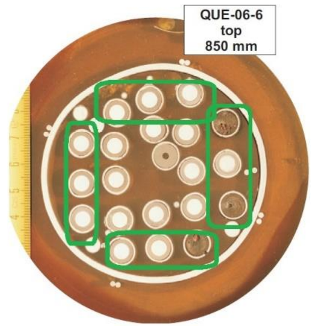

Figure 14. Bundle and fuel rods cross-section at 850 mm elevation [3].

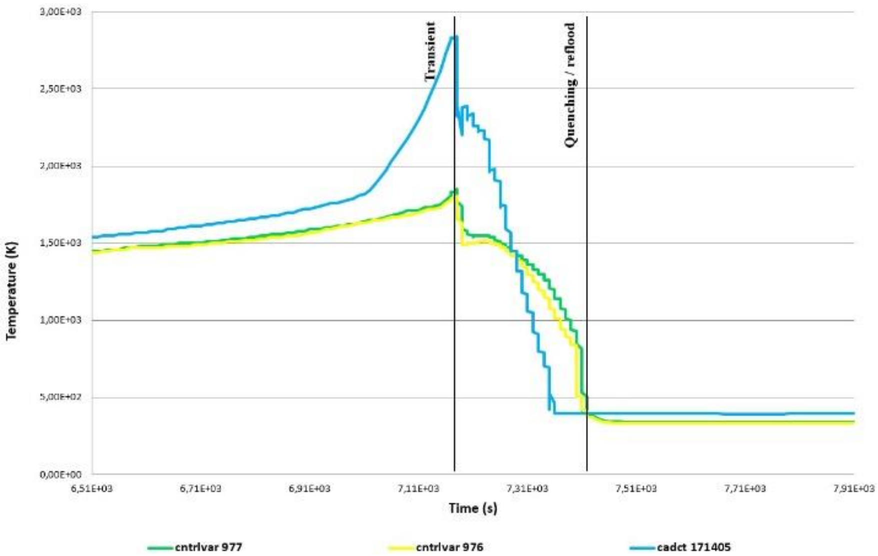

Figure 15. Measured vs. calculated shroud outer temperatures at 850 mm elevation.

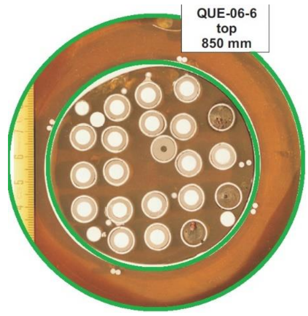

Figure 16. Bundle and shroud surface cross-section at 850 mm [3].

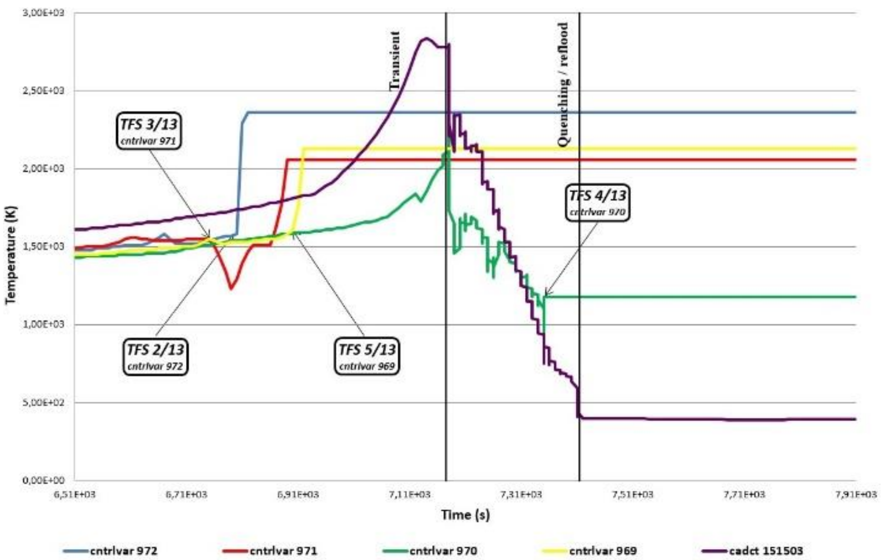

Figure 17. Measured vs. calculated cladding outer surface temperatures for outer ring rods at 950 mm elevation.

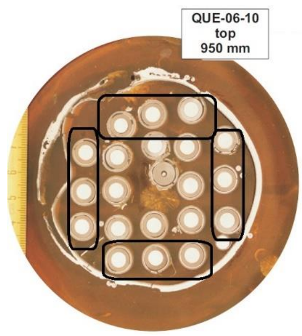

Figure 18. Bundle and fuel rods cross-section at 950 mm elevation [3].

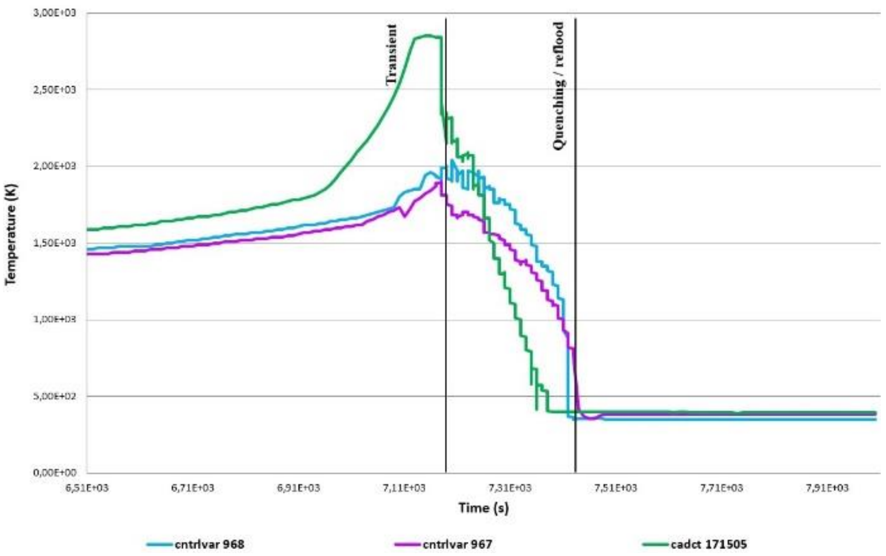

Figure 19. Measured vs. calculated shroud outer surface temperatures at 950 mm elevation.

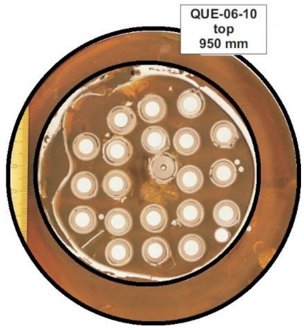

Figure 20. Bundle and shroud cross-section at 950 mm [3].

ring rods and shroud at 350 mm elevation. At Figure 9 and Figure 11 are depicted comparisons between measured and calculated temperatures for cladding outer surface for rods 10 to 21 and shroud at 550 mm elevation. Figure 13, Figure 15 show temperatures for outer ring rods and shroud of the test bundle at 850 mm elevation whereas at Figure 17 and Figure 19 are depicted temperatures for the same elements at 950 mm elevation.

On the other hand, with *cadct* are represented calculated by RELAP/SCDAPSIM/MOD3.5 temperatures at the same places as those that are being measured by thermocouples described above. Cadct is a parameter that describes temperatures at radial node (spacing mesh) *No.ii*, axial node *No.kk*, and component *No.jj* [12].

7 Conclusions

As can be seen from the graphson Figure 7, Figure 8, Figure 9, and Figure 11 there is a very good match betweenmeasured and calculatedvalues, while Figure 17 shows cladding outer surface temperature for rods 10 to 21 as well as the places where four thermocouple failures take place. In Figure 13, Figure 15, and Figure 19 there are some discrepancies between measured and calculated values of temperatures. This is mainly due to the influence of the radiation exchange in a gas gab because it is not being accounted for in original input file. If radiation exchange is not accounted, the argon gas in the gap between inner liner and outer structure in the shroud (above the tungsten) is acting as a very good insulator which makes the calculated temperatures way too high. If the radiation exchange is taken account then at high temperatures the gas becomes a good conductor keeping the calculated temperatures much lower and if the user defined density of the gap is set to a value less than 10 kg/m^3, the code will automatically account for the radiation exchange.

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References

[1] Kaliatka T., Kaliatka A., Vileiniskis V. (2016) Application of Best Estimate Approach for Modelling of QUENCH-03 and QUENCH-06 Experiments. *Nuclear Engineering and Technology* **48** 419-433.

[2] Kaliatka T., Kaliatka A., Vileiniškis V., Ušpuras E. (2014) Modelling of QUENCH-03 and QUENCH-06 Experiments Using RELAP/SCDAPSIM and ASTEC Codes. *Hindawi Publishing Corporation Science and Technology of Nuclear Installations* Volume 2014: Article ID 849480; Available from: http://dx.doi.org/10.1155/2014/849480.

[3] Sepold L., Hering W., Homann C., Miassoedov A., Schanz G., Stegmaier U., Steinbrück M., Steiner H., Stuckert J. (2004) Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45). Scientific report FZKA-6664, Karlsruhe, February 2004; Available from: http://quench.forschung.kit.edu/.

[4] Hering W., Homann Ch., Lamy J.-S., Miassoedov A., Schanz G., Seplod L., Steinbrück M. (2002) Comparison and interpretation report of the OECD International Standard Problem No.45 exercise (QUENCH-06). Scientific report FZKA-6722, Karlsruhe, 2002; Available from: http://quench.forschung.kit.edu/.

[5] CSNI INTERNATIONAL STANDARD PROBLEMS (ISP) Brief descriptions (1975-1999), NEA/CSNI/R(2000)5, Organization for Economic Co-operation and Development.

[6] Stanojević M., Leskovar M. (2001) Simulation of the QUENCH-06 experiment with MELCOR 1.8.5. In Proceedings of International Conference Nuclear Energy in Central Europe 2001 Portorož, Slovenia, September 10-13, 2001.

[7] Hering W., Homann Ch., Lamy J.-S. (2002) Comparison report on the blind phase of the OECD International Standard Problem No.45 exercise (QUENCH-06). Scientific report FZKA-6677, Karlsruhe, 2002; Available from: http://quench.forschung.kit.edu/.

[8] Marina Perez Ferragut, *RELAP/SCDAPSIM/MOD3.5 user training QUENCH-06 experiment*, Innovative Systems Software

[9] Allison C.M., Hohorst J.K. (2010) Role of RELAP/SCDAPSIM in Nuclear Safety. *Hindawi Publishing Corporation Science and Technology of Nuclear Installations* Volume 2010: Article ID 425658; Available from: http://dx.doi.org/10.1155/2010/425658.

[10] Trivedia A.K., Allison C., Khanna A., Munshi P. (2014) RELAP5/SCDAPSIM/MOD3.5 analysis of the influence of water addition during a core isolation event in a BWR. *Nuclear Engineering and Design* **273** 298-303.

[11] RELAP5/MOD3.3 CODE MANUAL VOLUME I: CODE STRUCTURE, SYSTEM MODELS, AND SOLUTION METHODS, Nuclear Safety Analysis Division December 2001 Information Systems Laboratories, Inc. Rockville, Maryland Idaho Falls, Idaho.

[12] Hohorst J. (2012) RELAP/SCDAPSIM Input Manual MOD 3.4 3.5 & 4.0, November 2012, Innovative Systems Software.