Experimental Measurements for Plate Temperatures of MTR Fuel Elements at Sudden Loss-of Flow Accident and Comparison with Computed Results

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1. Introduction

The aim of this study is to generate experimental data to be used for sensitivity analysis and assessment calculations on the thermal-hydraulic codes written for Loss of Flow Accident (LOFA) analysis of research reactor with MTR- type (plate type) fuel elements.

In an open pool research reactor with downward coolant flow, an accident such as shaft breaks between pump and flywheel can lead to sudden loss of flow. Flow reversal occurs in many MTR-type research reactors during transient after shut-down (flow scram). It can be mismatched with the decay heat removal by natural convection when the coolant flow rate becomes so low. A critical heat flux may occur even at a low heat flux. It is essential to demonstrate that such a sudden loss of flow does not lead to excessive temperature increase at the fuel plate and consecutively not cause fuel melting.

In the TR-2 Research Reactor, Sudden Loss of Flow Accident which happens by shaft break between pump and flywheel is simulated by closing the pool inlet valve (Figure-1).

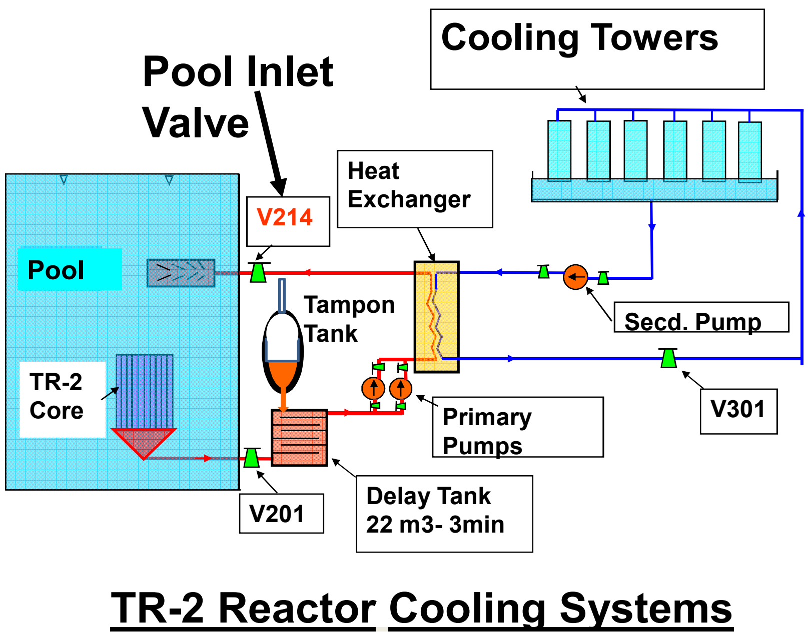


Figure-1

There is a butterfly type valve at the pool inlet and total closing time is about 30 seconds but effective flow decrease occurs in 10 seconds. Beginning of the flow decrease by closing of the valve was accepted as time zero and the decrease of flow rate versus time was plotted (Figure-2).

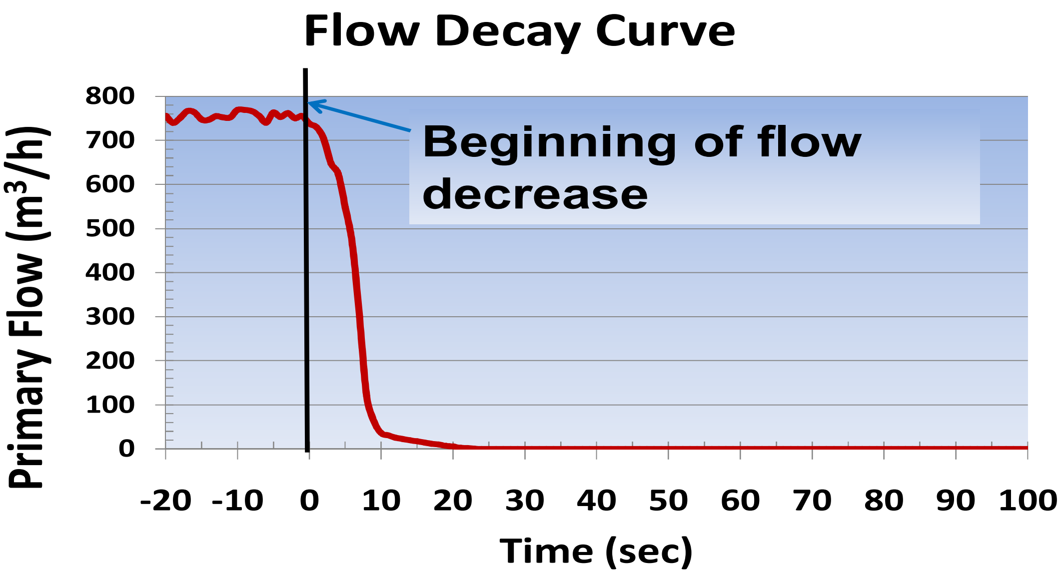


Figure-2

An instrumented fuel element, which has five thermocouples along the vertical length of the fuel plate, was used for the experiments (Figure-3a, 3b).

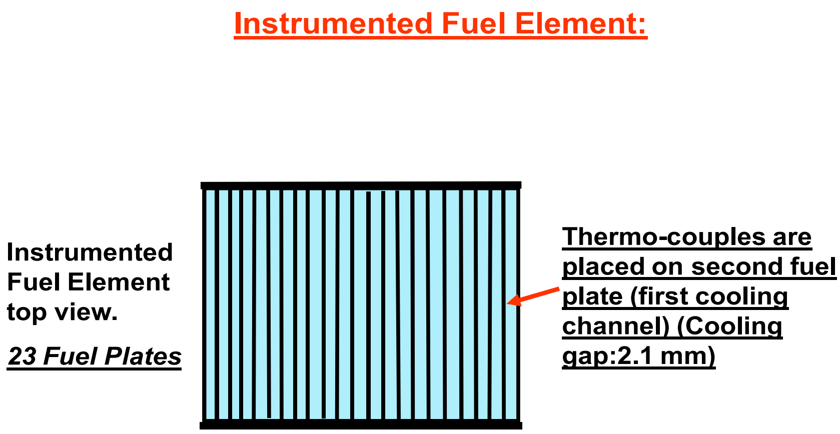


Figure-3a Instrumented Fuel Element Top View

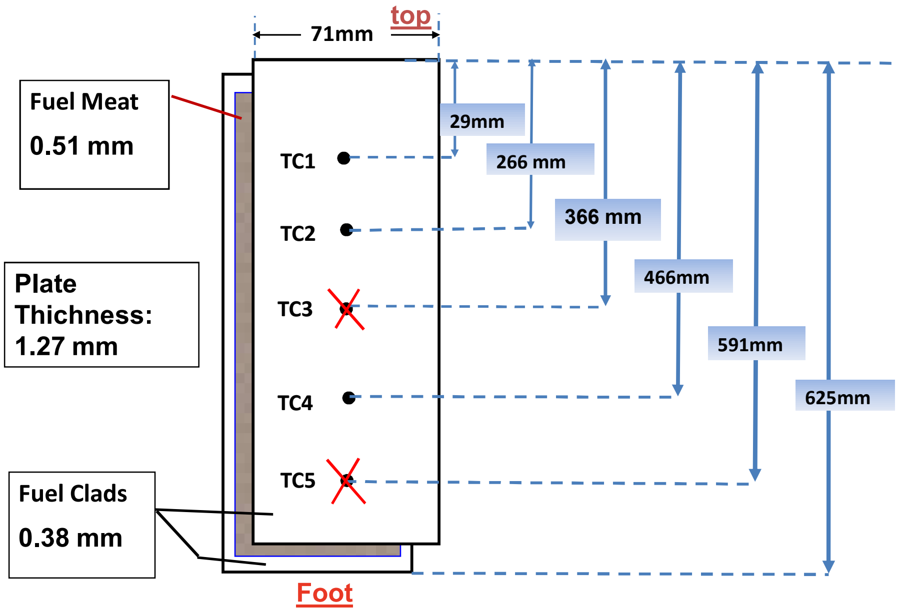


Figure-3b Location of the Chromel –Alumel Thermocouples on Fuel Plate of the Instrumented Fuel Element

This fuel element was placed in a certain position of the TR-2 core (Figure-4) and the plate temperatures were measured and recorded by a digital recorder.

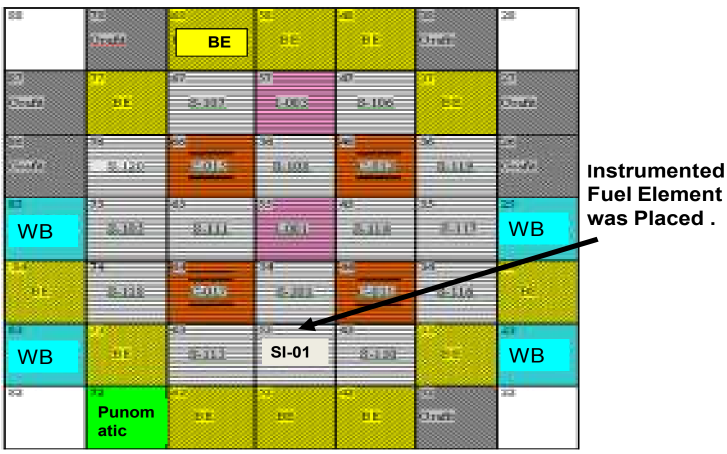


Figure- 4 TR-2 Core Configuration

At the same time primary flow rate was measured and recorded. Power distributions of the fuel elements in the reactor core were determined experimentally by using copper wire activation technique. According to the existing core loading of the TR-2 Reactor, core positions, U-235 weights, relative neutron flux distributions and power distributions of the fuel elements were given as the inputs of the calculations to simulate the loss of flow accident at the computer codes.

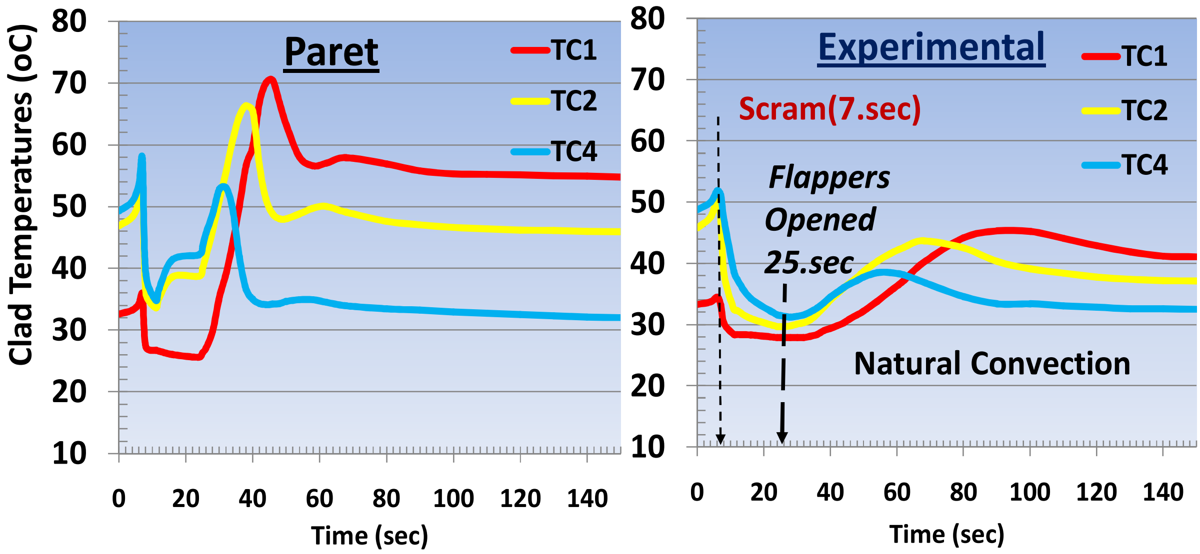
2. Experimental Measurements

Four experiments were performed for sudden loss of flow accident. Three of these experiments were repeated with different initial conditions such as different core inlet temperatures and different reactor operation times at 5 MW nominal power level and 750 m^3/h primary flow rate. The reactor was shutdown by flow scam at these three experiments.

<u>Experiment I:</u>

The TR-2 Reactor had been operated at 5 MW for 5 minutes. The primary flow rate was $750 m^3/h$ and core inlet (pool) temperature was 23°C. Primary coolant flow rate began to decrease from $750 m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 7 seconds and natural convection flappers opened after 25 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

Comparision of Results for The 1 Experiment



P=5 MW ***Total reactor operation time is 5 minute before scram***

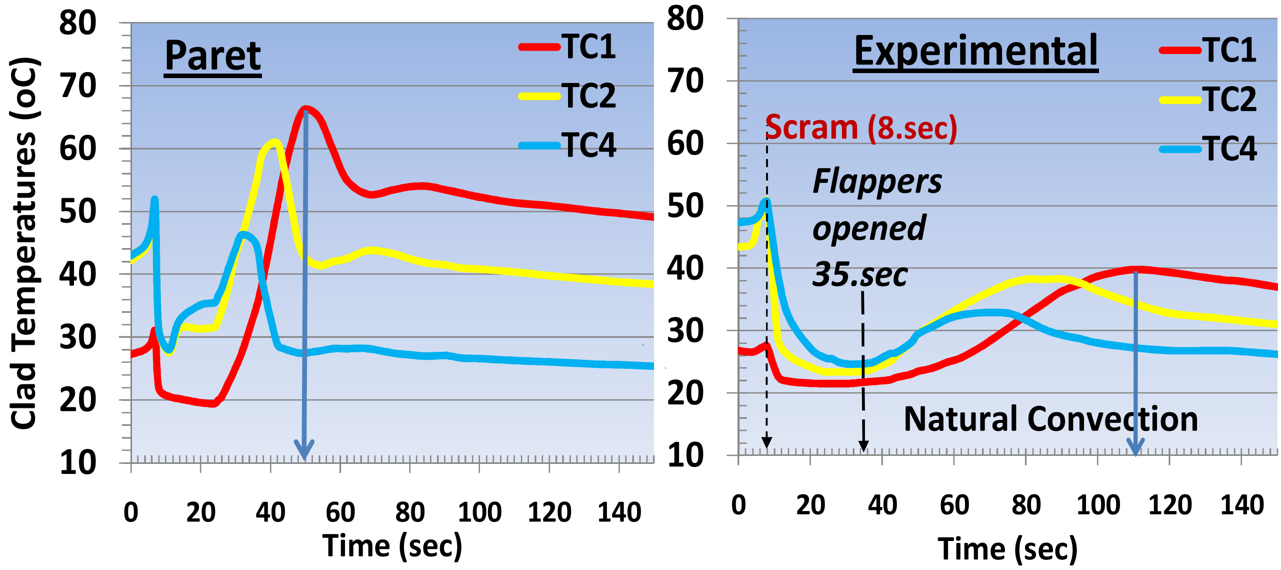
Initial flow rate: 750 m^3/h (reverse flow)

Pool Water temperature : 23 °C (Core inlet)

Experiment II:

The TR-2 Reactor had been operated at 5 MW power for 5 minutes. The primary flow rate was $750 m^3/h$ and core inlet (pool) temperature was 17^oC. Primary coolant flow rate began to decrease from $750 m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 9 seconds and natural convection flappers opened after 35 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

Comparision of Results for The 2 Experiment



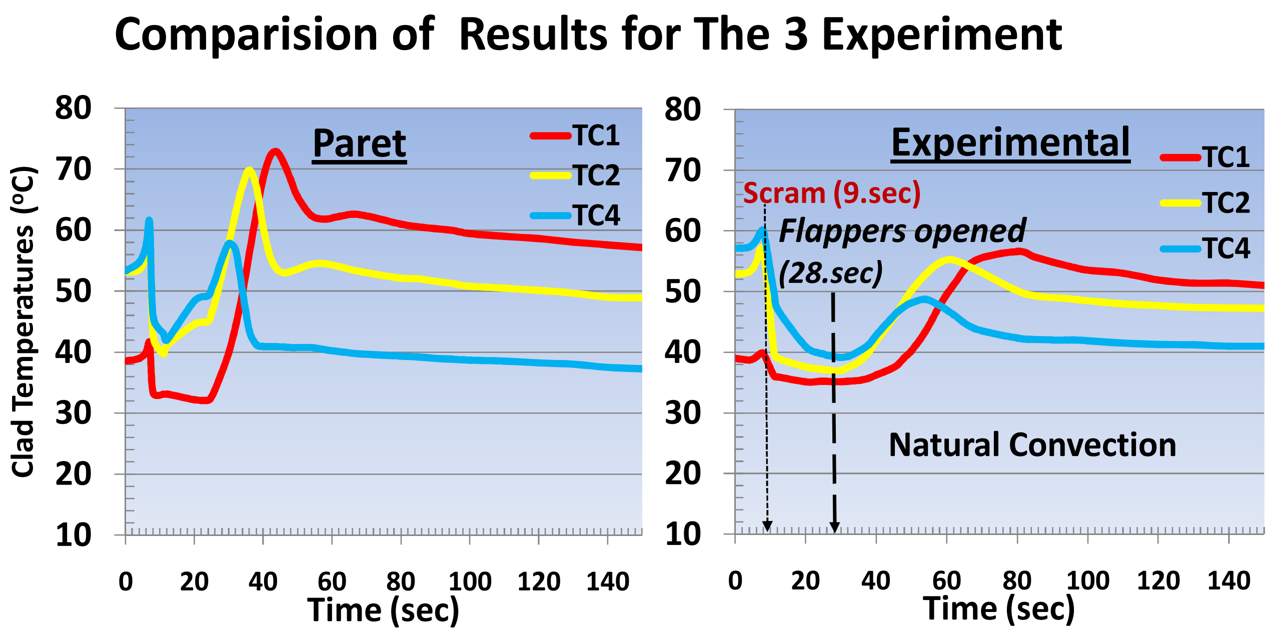
P=5 MW Total reactor operation time is <u>*5 minute*</u> before scram

Initial flow rate: 750 m^3/h *(reverse flow)*

Pool Water temperature : 17 °C *(Core inlet)*

Experiment III:

The TR-2 Reactor had been operated at 5 MW power for 2 hours. The primary flow rate was $750m^3/h$ and core inlet (pool) temperature was 29.5^oC. Primary coolant flow rate began to decrease from $750m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 9 seconds and natural convection flappers opened after 28 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

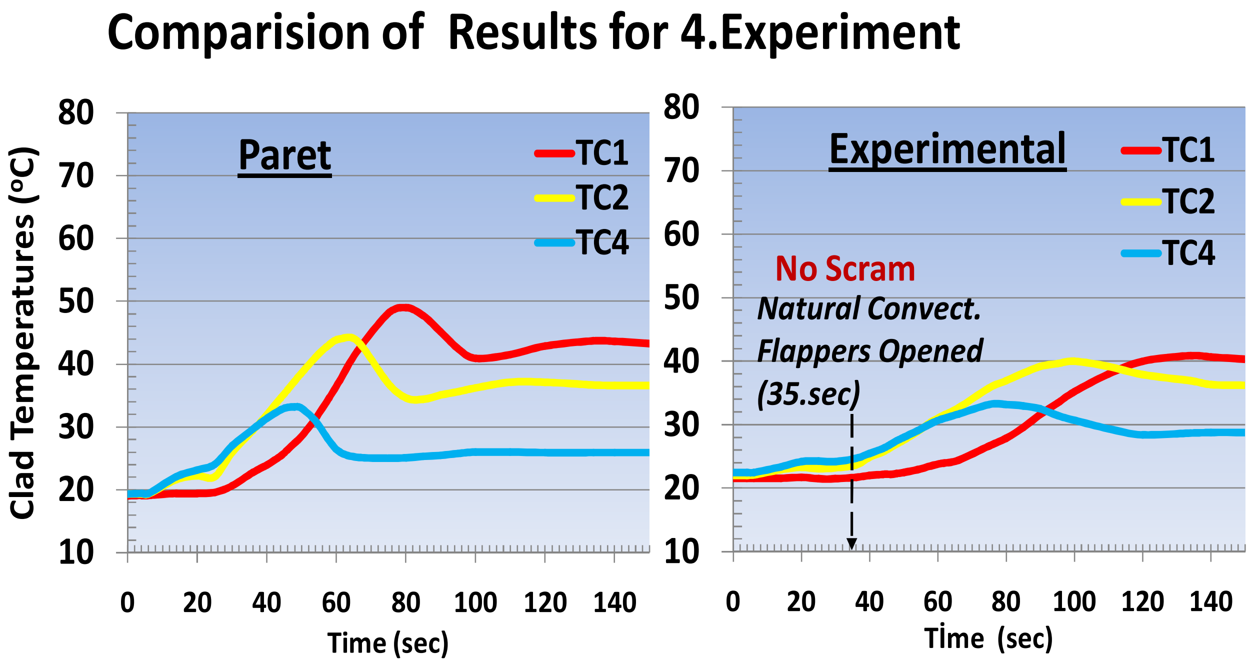


P=5 MW **Total reactor operation time is *2 hours* before scram**
Initial flow rate: $750\ m^3/h$ *(reverse flow)*
Pool Water temperature : $29.5\ ^oC$ *(Core inlet)*

The sudden loss of flow accidents performed in these three experiments were simulated and analyzed by using PARET computer code. The measured plate temperatures were less than the calculated plate temperatures. The differences may come from the decay heat calculation or modeling of the natural convection for narrow, rectangular flow channels at PARET code. To investigate the source of these differences a fourth experiment was performed at 50kW power level, which allows the reactor to operate with natural convection without primary flow scram.

Experiment IV:

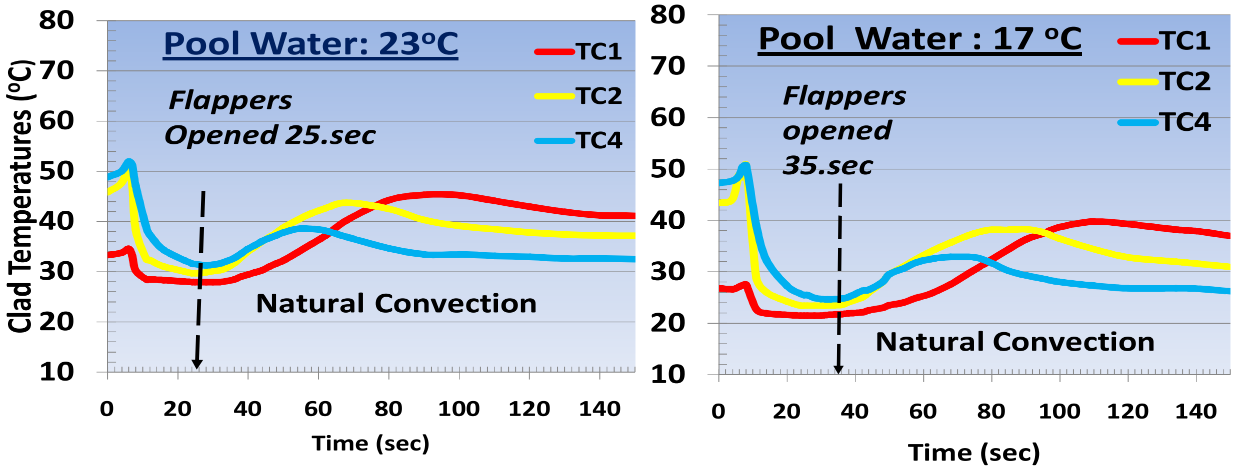
The TR-2 Reactor had been operated at 50kW power for 5 minutes. The primary flow rate was 750m^3/h and core inlet (pool) temperature was 19^{o}C. Primary coolant flow rate began to decrease from 750m^3/h (time zero) by closing of the core outlet valve. The primary flow scram not occurred (50kW neutronic power) and natural convection flappers opened after 35 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.



P= 50 KW Neutronic Power - Total reactor operation time is ***5 minutes***
before flow decrease **Without SCRAM**
Initial flow rate: 750 m^3/h *(reverse flow)*
Pool Water temperature : 19 ^{o}C *(Core inlet)*

3. The Cross Comparisons of the Experimental and Calculated Results

Comparision of the 1. & 2. EXPERIMENT



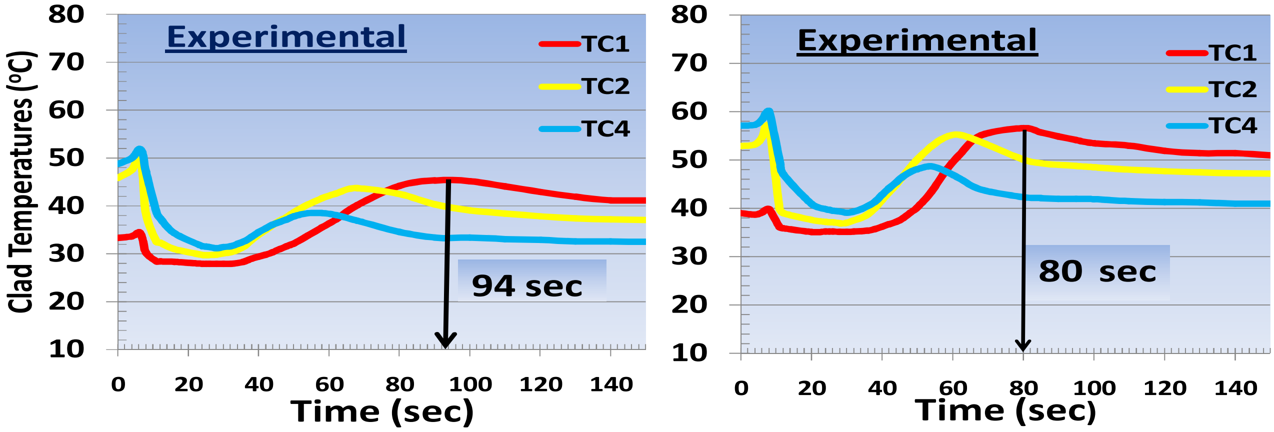
P=5 MW *Total reactor operation time is* ***5 minute*** *before scram*

Initial flow rate: ***750 m³/h (reverse flow)***

Comparision of the 1. & 3. EXPERIMENT

5 Minutes 5 MW Power Operation
Pool Water: 23 °C

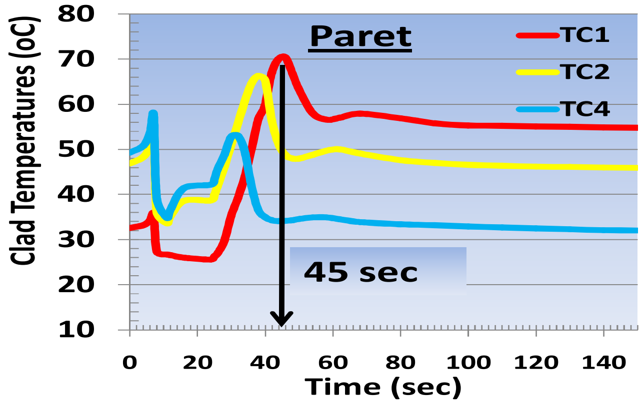
2 Hours 5 MW Power Operation
Pool Water: 29.5 °C



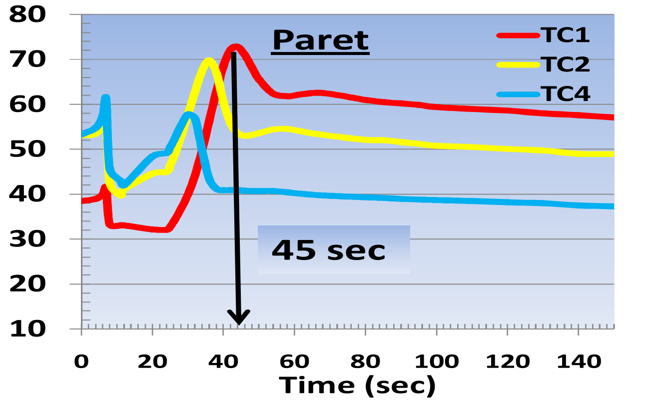
They are compatible

Comparision Of the 1. & 3 Calculations:

5 Minutes 5 MW Power Operation
Pool Water :23 °C

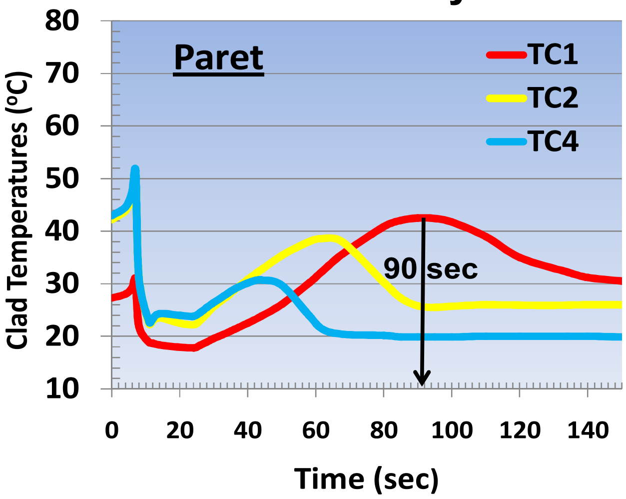


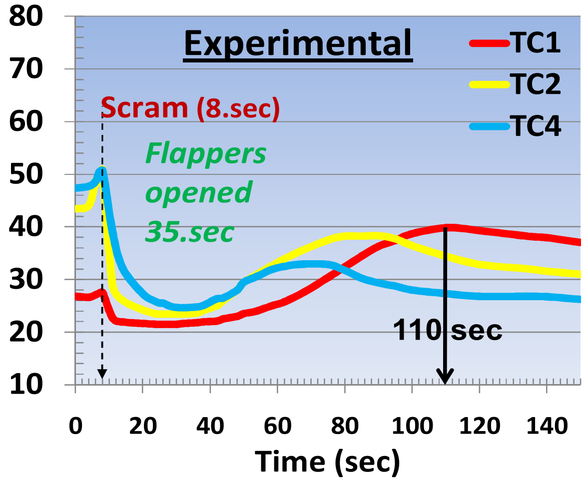
2 Hours 5 MW Power Operation
Pool Water :29.5 °C



The Calculation by PARET was repeated for 5 minutes, 1 MW power operation before scram and just before the scram 5 MW power operation to simulate the calculations for decreased decay heat. And the results are compared as follow:

Decreased Decay Heat





For Paret Calculations : P=5 MW , Reactor operation time before scram 5 minutes at **1 MW** , **(to decrease decay heat at the calculations)**
Initial Flow Rate: **750 m³/h**
Pool Water Temperature: **17 °C**

4. Conclusions:

- It was determined that the residual decay heat calculation and the natural convection modeling for narrow rectangular channels of PARET code are not quite realistic.

- PARET uses the decay heat generation rate within the reactor core based on the standard fission-product decay heat curve for uranium-fuelled thermal reactors published by the American Nuclear Society as a proposed standard (ANS-5.1/N18.6). ANS decay heat curve is too pessimistic.

- **For the calculation of the residual decay heat, special attention must be given to the fact that "*Approximately one-half of the decay heat is due to gamma radiation with energies in the range of about 0.2-2.0 MeV. The e-folding length for 1 MeV gamma radiation in aluminum is 6 cm, i.e., the source gamma intensity is attenuated to 1/e of its original value after passing through 6 cm of aluminum. This corresponds to a penetration of 40-50 fuel plates and hence shows that a significant portion of a given level of fuel element decay heat will be deposited at the outside of the fuel element*".**

Realistic calculations can be done for LOFA and LOCA analysis by using the half of the values of decay heat generation curve of ANS.