ABSTRACT

During Romanian TRIGA 14 MW steady state reactor commissioning in 1979 and beginning of 1980 a series of tests were performed to check the functioning of reactor systems and the response of the reactor core to transient initiators. Two of these tests consisted in inducing total forced cooling loss and also loss of primary pumps cooling with emergency pump in operation for 15 minutes after the event. Maximum fuel temperatures in hot pins were recorded. A RELAP5 reactor model was created to simulate the Loss of Coolant Transients and to compare with experiment. The scope of the paper is to describe the model and the most significant parameters which affect the results and to compare RELAP5 fuel temperature with that observed during the experiment in the hottest TRIGA fuel element.

1 INTRODUCTION

There are two independent cores sharing the same pool: a high-flux 14 MW steady state reactor (SSR), research and materials testing reactor and an independent (from operational point of view) annular core pulsing reactor (ACPR). Figure 1 shows a view from above with the pool, the two reactor cores thermal column and beam tubes.

The steady state reactor is a forced convection reactor cooled via a primary circuit containing 4 pumps and 3 heat exchangers. Secondary circuit has 3 pumps and cooling towers. At nominal power level 2 primary pumps are in operation. The reactor has an emergency (shutdown) pump located in the pool and operating through a pipe connected to the lower fitting of the reactor core. It is designed to remove decay heat and operates both simultaneously with primary pump(s) at selected power level and two hours after reactor shutdown. Emergency pump has redundant DC power supply. The ACPR is natural convection cooled by the pool water.
2 LOSS OF FORCED COOLING TESTS AND RELAP5 CALCULATIONS

2.1 Description of the tests

Loss of Cooling Tests (LOFCT) were performed in February 1980 on the reactor start-up core configuration called “Standard Configuration”. It consisted of 29 High Enriched Fuel bundles, 8 control rods, 44 beryllium blocks (20 with experimental hole) and experiments locations as follows:

- 3 small locations XC (1 fuel bundle dimensions each);
- 3 large locations XL (4 fuel bundle dimensions each).

Standard Configuration is represented in figure 2. Experiment locations contained a total of 120 CANDU fuel rods at the time of tests. Tests were performed as follows:

1) after a steady state period of operation at 14 MW power level the primary pumps and emergency pump were turned off;
2) After a steady state period of operation at 14 MW power level only the primary pumps were turned off and the emergency pump continued to operate for 15 minutes.

The recorded parameter during the transients was maximum fuel temperature. The values of maximum fuel centreline temperature are the limiting design basis parameter, because of the temperature dependence of the out-gassing of hydrogen from Uranium-Zirconium-Hydride and subsequent stress on clad material.
2.2 RELAP5 model of Loss of Forced Cooling Tests

RELAP5 computer code was selected to model the hydrodynamic and the heat transfer of the problem. RELAP allows for a detailed modelling of the phenomena involved in such cases and is a wide spread choice in light water reactor transients simulation. For the case under consideration the stress is on the capabilities or RELAP to represent different core hydrodynamic channels and the crossflow between volumes above and below the core, in order to allow for natural circulation and pump flow.

RELAP 5 mod 3.2 is the version of the code that Institute for Nuclear Research is in possession of and that is used for the purpose of this paper.

A model of the reactor open pool, core and emergency pump loop was created to calculate the transients. Nodalization of the problem is presented in figure 3. Nevertheless, this is part of a more complete TRIGA model containing primary lines and components of primary circuit (pumps, valves, delay tank, heat exchangers) which was created before to simulate steady state operation of the reactor at nominal power level. The differences between the simplified model in fig.3 and the complete model (the later is not detailed in this paper) lie in the removal of primary lines for the purpose of the LOFC tests simulation and the use of different power peaking for the hot channel, axial power distribution and different power percent produced in the moderator, as representing basically the spatial distribution after the reactor shutdown.

The nodalization includes:
- the open pool (components 906 and 101) above the core and atmosphere (component 551),
- the pool just above the core (components 677 to 673 and 777 to 773) with crossflow,
- the core (components 402 = channel associated with experiments, 302 = non-heated channel, 202=medium channel, 102=hot channel),
- the pool at core level (components 890 and 902),
- fitting below core (components 877 to 874 and 977 to 974 and 103) with crossflow,
- emergency cooling system (components 104, 105 and 135)
Figure 3: Nodalization used for LOFC transients
Cross-flow connections were realized as to allow mixing between all core channels above and below the core. No cross-flow connections were made between core channels along the axial dimension of the core. This took into account that cross-flow can be realized at this level only inside a shroud of one bundle and that hot channels are in the neighbourhood of other high rated pin channels in that same central bundle.

Axially the core was divided in 15 segments, 13 summing up to the active length and the remaining 2 representing the lower and the upper plenum of the bundle.

According to ref.[1] and [5] hot channel was estimated to have a value of actual power / medium pin power = 1.59 in the steady state calculation at 14 MW (where a more complete model including two primary pumps was used) at a flow-rate of 660 l/s through the primary line. Axial peaking factor was 1.35.

For LOFC transients simulation, the hot pin factor was 1.29 and the axial peaking factor was 1.27 according to ref.[2]. Delayed hot pin factor was calculated as $R_d=0.5[1+ R_p]$ where $R_p$ is prompt pin factor = 1.59, taking into account that gamma ray energy may be deposited a considerable distance from the location of generation and gamma energy is about half the total delayed energy. The result is the 1.29 value mentioned above.

Estimation of the energy deposited directly in the moderator after reactor shutdown was based on previous calculations of gamma flux and gamma heating in TRIGA reactor [6] and was found to be 6.65 % of the total energy.

Heat structures simulating the fuel hot, medium and experiment pins were attached to components 102, 202 and 402. Low level of power was estimated to be released by natural UO2 experiment fuel (around 2% of total thermal power) according to ref.[1].

RELAP5 gap conductance model and a linear temperature dependence of conductivity for the fuel as an average between minimum and maximum best data found in literature for UZrH and reproduced in ref.[3] pp 8 fig.4. The interval for fuel conductivity is defined with ±17% around average curve.

The emergency pump flow rate was 22 l/sec in case of partial loss of flow. The core channels have the following areas:
- hot TRIGA channel (volume 102 in the nodalization) 1.3596 cm²
- medium channel (202) 984 cm²
- non-heated channel (302) 380 cm². This includes water surface in clearance between reactor elements (80 cm²) and water inside control rods square cross section (300 cm²). This channel has large entrance and exit pressure loss coefficients because of the very restricted inlet and outlet sections in case of control rods water. For this reason the Safety Analysis Report [1] simply states that flow through this path is insignificant.
- experiments water channel (402) 340.5 cm². The surface includes the cooling area for 3 large experiments (37 natural UO2 rods each), 119 cm², and for 3 small experiments (3 rods natural UO2 each), 221.5 cm².

Entrance and exit pressure loss coefficient in core channels was taken as 2.0 and 0.3 respectively, as in ref.[1]. From the same Safety Analysis Report ref.[1] the three spacer grids have each 0.6 as their value for the loss coefficient.

The experiment channel (402) is composed of two types of flow area, one similar to CANDU in large experiments (circular design CANDU pressure tube blocking the flow outside the tube) and square shrouds in small experiments (providing larger flow area). The entrance exit and grid coefficients were approximately taken as for TRIGA channel because the complete design data were not available. Calculation was found to be not sensitive to modifications (in a reasonable range) around the above values.
For the non-heated channel the entrance and exit coefficients are calculated using dedicated handbook ref.[6] resulting the values \(\approx 2000\) at entrance and \(\approx 380\) at exit.

In the complete TRIGA model power was supplied by reactor kinetic, and the reactor was scrammed in 1.5 sec by introducing total control rod bank worth (19$). Coast-down curve for the flow rate was determined experimentally prior to the test. Complete model was used for obtaining steady state temperatures and decay heat vs. time that was used further in generating power tables for hot fuel, medium fuel and experiment fuel in transient simulation. Results for transients after shutdown, starting with minute 1 after reactor scram are presented below.

2.3 Steady State and Transient maximum fuel temperature vs. experiment

Though not exactly reproducible, steady state axially maximum fuel centreline temperature was recorded to be 530 °C at the time of the test (ref.[4]) at 14MW. RELAP5 with linear conductivity described gives 541.8 °C.

Figure 4 and 5 present RELAP5 results together with experiment temperature measurements for complete Loss of Flow and partial followed by complete Loss of Flow:

![RELAP5 vs. experimental data (emergency pump off)](image)

Figure 4: RELAP5 vs. experiment for total Loss of Forced Cooling

It can be seen from the graphical results in fig.4 that a good agreement between thermocouple data for maximum fuel temperature and RELAP5 is obtained for complete loss of flow. The irregularity (small increase in temperature after 6 minutes from scram) does not appear either in RELAP or in evolution provided by thermocouples located in other fuel pins during the test.

In case of the test with emergency pump stopped after 15 minutes of operation (fig.5) it can be observed a good agreement too between RELAP5 simulation and test data.
Additional calculations were performed modifying the experiment water surface. Dotted curve in Fig. 5 presents the calculated curve for actual experiments water surface multiplied by 1.7. In this case dissimilarities appear and a different flow pattern through hot channel is triggered. In these conditions the emergency pump flow rate is not sufficient to oppose natural circulation and temperatures are significantly higher for the same reactor power. The pump stop in this hypothetical case leads to better cooling and decrease of the fuel temperature as opposed to actual case.

2.4 Conclusions

1) The simulation of the tests with RELAP5 model presented matches very well the thermocouple data during test. Also, the steady state value for the temperature is well reproduced (541 °C calculated vs. 530 °C recorded).

2) The temperatures during the test with emergency pump in operation are lower than without the pump by roughly 20 degrees Celsius. But taking into account the calculations presented it is conceivable that conditions might arise (depending on the non-heated or almost non-heated water surface in the core) in which emergency pump opposing to natural circulation will lead to higher temperatures in the hot channels than without it in course of loss of main pumps. In this case the emergency pump existence would have no justification.
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REFERENCES


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