DEVELOPMENT OF COMPUTER CODES FOR LOSS OF COOLANT ACCIDENT ANALYSIS OF IEA-R1 REACTOR

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ABSTRACT

Two computer programs, LOSS and TEMPLOCA, were developed to analyze postulated Loss of Coolant Accidents (LOCA) in the IEA-R1 reactor, at IPEN/CNEN-SP. The LOSS program determines the time to drain the reactor pool down to the level of the bottom of the core. The TEMPLOCA program calculates the maximum temperature reached in the fuel, due to the decay heat of fission products and when there is complete loss of coolant in the core. These programs were used to assess the safety of a Miniplate Irradiation Device (MID), placed in the IEA-R1 reactor core, during the occurrence of a postulated loss of coolant accident. The MID are being used to receive miniplates of U_3O_8-Al and U_3Si_2-Al dispersion fuels, LEU type (19,9% of ^{235}U) with uranium densities of, respectively, 3.0 gU/cm^3 and 4.8 gU/cm^3. The fuel miniplates will be irradiated to nominal ^{235}U burnup levels of 50% and 80%, in order to qualify the above high-density dispersion fuels to be used in the Brazilian Multipurpose Reactor (RMB), now in the conception phase

1. Introduction

In 2007, there was the replacement of the IEA-R1 reactor heat exchanger, what allowed the reactor to run safely to a power of 5 MW. In this power, it is possible to reach high fuel burnups, in relatively shorter timeframes. Therefore, it was proposed to place in a peripheral position of the IEA-R1 reactor core, ten fuel miniplates, fabricated at IPEN/CNEN-SP, dispersion type, of U_3O_8-Al and U_3Si_2-Al, in the maximum densities qualified worldwide and to follow their performance under irradiation. To accommodate the fuel miniplates, a special Miniplate Irradiation Device (MID) [1] was designed and fabricated. The MID has the external dimensions of the IEA-R1 fuel element. The miniplates was allocated in a box with indented bars placed inside the external part of the MID.

2. LOCA analysis in the IEA-R1 reactor

In order to carry out any experiment in the IEA-R1 reactor, it is necessary to analyze its behavior in case of postulated accidents. The safety analysis demonstrated the necessity to evaluate how safe MID is, in case of postulated Loss of Coolant Accidents (LOCA). In case of LOCA, the IEA-R1 reactor has safety systems which interrupt the chain reaction, inserting the control bars. Nevertheless, even with the reactor shut down, heat continues to be produced by decay of fission products. To assure the physical integrity of fuel elements during a postulated LOCA, a Emergency Core Coolant System (ECCS) was developed, which is started in the case of partial or total core draining. The EECS covers only the reactor core, not cooling the reactor irradiation positions [1]. DIM is placed in one of these positions and the fuel miniplates would go without refrigeration. Studies were performed to evaluate possible postulated loss of coolant events, which could lead to the reactor pool emptying [2], [3]. Five of them were considered to be mostly critical: a) Tube rupture of the Irradiation Pneumatic System (IPS); b) Pool drainage failure – rupture of the access tubes for the Water

Retreatment System (WRS); c) Primary system boundary rupture; d) Undue opening of the WRS drains; e) Failure in the collimator tubes of Beam Hole-3 (BH-3).

3. Methodology for LOCA analysis

For the analyses of LOCA, the programs LOSS [2], which calculates the time for the reactor pool emptying and TEMPLOCA, which calculates the maximum temperatures reached in the fuel, during the reactor pool emptying, were developed. LOSS solves the energy equation for a control volume comprised between the pool and the rupture in the tubes, providing the results as to water level in the reactor pool, the volumetric flow for pool emptying and water velocity in the rupture, for each interval in the specified time. For the pressure loss calculation in the singularities, experimental values of the IEA-R1 were used, as well as those available in the literature [3]. For the calculation of the fuel element temperatures (or the minplate temperatures), it was used, in the TEMPLOCA, the same model applied to the safety analysis of the Omega West Reactor [4], which is based on the experimental results from the Oak Ridge Reactor (ORR) [5] and the Low Intensity Testing Reactor (LITR) [6]. The reactor core (or a fuel element) was considered without water and a energy balance in the fuel plates, where the terms of energy storage, decay power, fuel plates heat transfer by natural convection to the air, heat conduction in the structures and radiant heat transfer from the core to the environment, was taken into account. This energy balance is given by the following equations:

$$M.C_p \frac{d\theta}{dt} = Q(t) - H.\theta \qquad (1)$$

where, Q(t) is the decay heat in MW, M is the fuel mass in Kg, C_p is the fuel specific heat in MJ/Kg°F, H is the heat transfer coefficient in MW/°F, θ is the difference between the fuel maximum temperature and the mean air temperature in °F. The heat transfer coefficient H was calculated through experimental data of the ORR reactor [5], given by the following expression:

$$H = 1{,}3 \times 10^{-6} \times \left(6{,}4 \times 10^{-3} \times \theta^{0{,}72} + 0{,}5\right) \qquad (2)$$

TEMPLOCA validation was carried out using experimental data for loss of coolant tests in the LITR [6] and ASTR [7] reactors, as described in the reference [3].

4. Description of LOCA events in the IEA-R1 reactor

4.1 Tube rupture of the Irradiation Pneumatic System (IPS)

The IPS comprises eight stainless steel tubes which enter the pool by its bottom and extend to half height of the reactor core lateral part. Four tubes remain deployed in one side of the core and the other four ones stay at the opposite side. Outside the pool, the IPS has leaky plastic tubes. The system is used for samples (“rabbits”) irradiation and the postulated LOCA would be due to the rupture of the stainless steel tubes inside the reactor pool. The rupture would occur by guillotine type rupture in the IPS tubes, as a consequence of a strong external impact. Failures in the tubes or welded material may be identified through routine reactor maintenance or wetting the irradiated samples, thus diminishing the probability of this type of accident occurrence [1]. The analysis with LOSS program for the accident, due to the stainless steel tubes rupture inside the pool, resulted in a time of 1hr16 min, for the complete emptying of the pool.

4.2 Pool drainage failure – rupture of the access tubes for the Water Retreatment System (WRS)

The two drain exits of the pool, located, respectively, in the reactor compartment and in the spent fuel storage compartment, are connected to the WRS by stainless steel pipes. The drain exit of the reactor compartment remains isolated by a gate valve localized in the reactor basement. The drain exit of the spent fuel storage remains open and coupled to a hose with water collector and a buoy, having their course limited by a 8.7 meter steel cable [2], [3].Two cases of accidents may be considered:

1. Drain pipes rupture downstream of the two drain pipes junction that would cause the pool water leakage, up to the aspiration buoy blockage and later closure of the isolating valve of the fuel storage compartment drain;
2. The most critical case would occur with the release of the pump flywheel and consequent shock with the pipes leading to the pipe rupture upstream of the isolating valve from the reactor compartment and an uncontrollable emptying of the pool water would occur.

The accident analysis with the LOSS program, with the release of the pump flywheel and consequent rupture of the access pipes to the WRS, resulted in the time of 3h54min, for the complete emptying of the reactor pool.

4.3 Primary system boundary rupture

The primary system comprises the return pipes of the primary coolant that is near the pool. It operates at low pressures and temperatures so that a guillotine type rupture of its pipe would occur only by a strong external impact. As the probability of earthquakes or falling airplanes occurrences are very low, the most probable cause of rupture of the primary system would be the release of the primary pumps flywheel. The primary circuit has two manual isolation valves, located in the basement of the building, at the pool entrance and at the exit, respectively. Fig 1 shows the variation of water level in the pool as a function of time, calculated with the LOSS program, for an accident with the release of the pump flywheel and consequent primary system rupture. From Fig 1 it is noted that the time for the complete pool emptying is 7.5 minutes. The valves are observed to be ineffective in this case, due to the rapid pool emptying and their location near the rupture region.

4.4 Undue opening of the WRS drains

The WRS consists of two circuits of treatment, each of them with six drain valves. The undue opening of the WRS drains could take place by sabotage, which would be not very probable, since the safety facility devices and procedures restrict access of external personnel and even of operators to the basement, where the drains are located. The Division of Safety Analysis has requested two safety measures to prevent accidental leakage of the pool through the drains, which were adopted by IPEN-CNEN/SP: 1) the WRS isolating valve blockage with a padlock; 2) blockage of the hose buoy for the pool water collection, through an 8.7meter steel cable. The analysis with the LOSS program for the accident of undue opening of the WRS drains resulted in the time of 13 hours for the pool complete emptying.

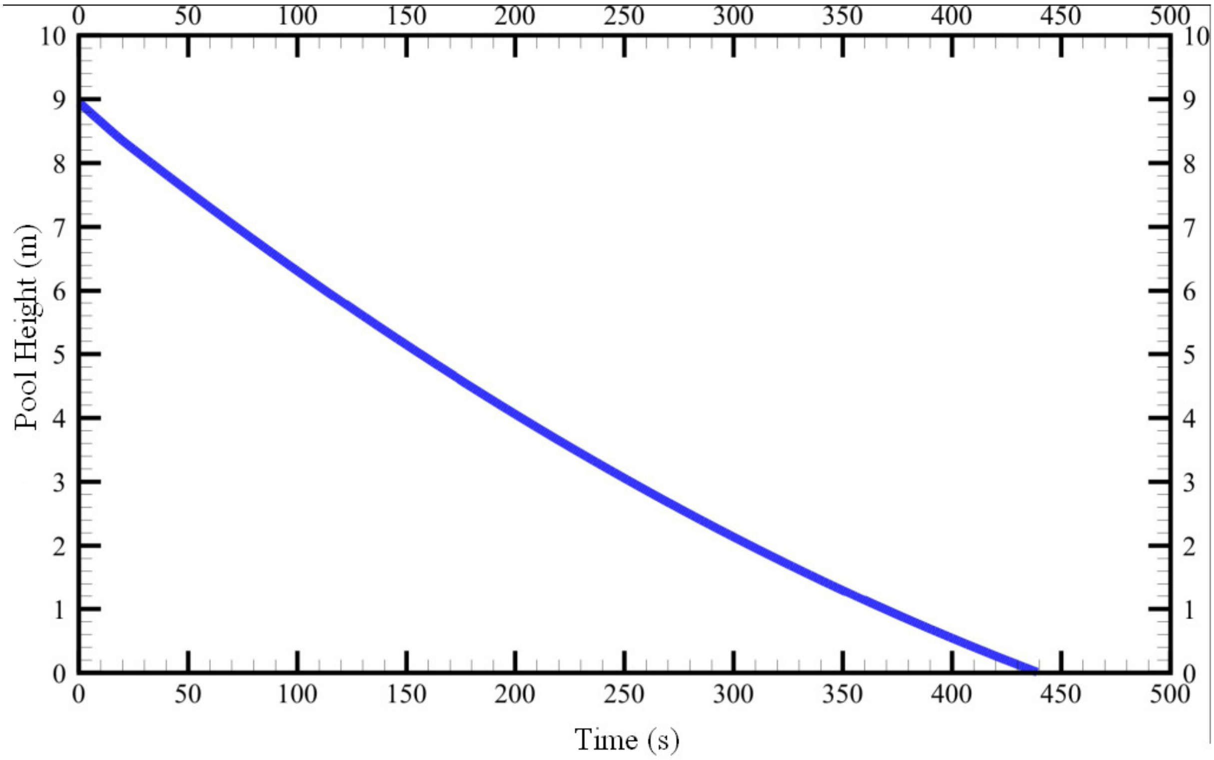

Fig 1. Time of the reactor pool emptying, due to primary system boundary rupture of the IEA-R1 reactor.

4.5 Failure in the collimator tubes of Beam Hole-3 (BH-3)

The IEA-R1 has eleven aluminum tubes (“beam holes") to conduct experiments. The tubes cross the concrete walls of the pool horizontally, in twelve points. The tubes are positioned at different heights in relation to the reactor core so that only the BH-3 rupture would uncover the fuel miniplates active part. In the case of other beam holes rupture, the water level in the pool would remain above the active part of the fuel miniplates, what would ensure the forced cooling by the main pump.

The analysis with the LOSS program for the BH-3 tube rupture resulted in the time of 23 minutes, for the pool emptying up to the BH-3 level (height minus the tube inner diameter).

5. Miniplate temperatures calculations after a primary system boundary rupture

Out of the five accidents analyzed with the LOSS program, the primary system boundary rupture was found to be the most critical. The calculations showed that about 7.5 min are necessary to drain the reactor pool during a postulated primary system boundary rupture. After the pool draining, the maximal fuel miniplate temperatures calculated with the TEMPLOCA was 125 ºC, below the blistering temperature, which is the fuel temperature design limit. At the blistering temperature the fuel miniplate will swell due to the fission gases released in the fuel and can close the fuel miniplate cooling channels with the fuel temperature increasing up to miniplate melting. The value of blistering temperature for dispersion fuels can vary between 350℃ e 600 ℃, depending on dispersion fuel type, fuel enrichment and burnup achieved.

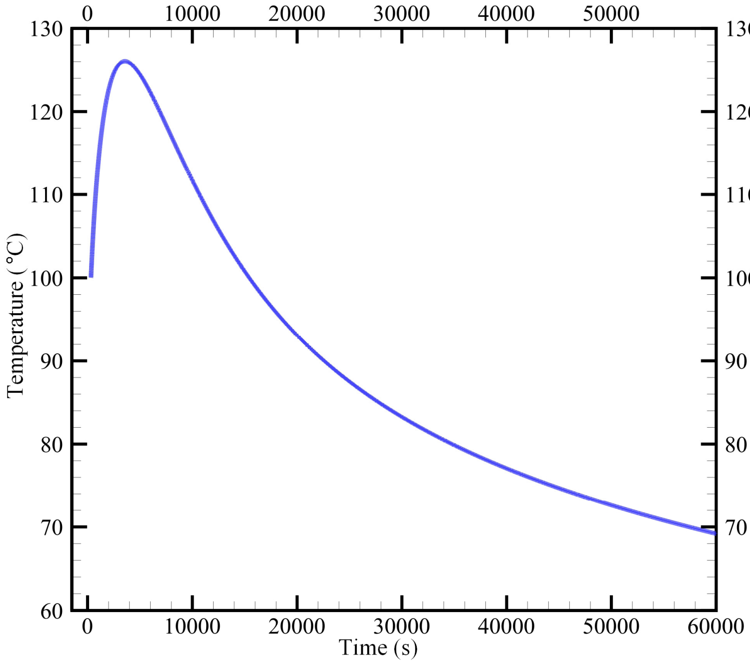

Fig 2. Miniplate Temperatures after IEA-R1 reactor core uncovering.

4. Conclusion

The safety analysis performed with the LOSS program shows that in case of a primary system boundary rupture, approximately, 7.5 min will be necessary for the reactor pool emptying. After this emptying time, the maximum temperature calculated with TEMPLOCA in the fuel miniplates located in the DIM, due to decay heat, was 126°C, which is below the fuel design limits.

5. References

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