ANALYSIS OF LOSS OF COOLANT ACCIDENT IN MTR POOL TYPE RESEARCH REACTOR

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ABSTRACT

The present work focuses on a simulation of total purge of the pool of an hypothetic research reactor operating at a fixed power before accident start-up. The objective is to verify if this may induce any fuel damage in the fuel elements due to inadequate heat removal or any loss of pressure.

1 INTRODUCTION

During normal operation of the research reactor, the residual heat is adequately removed either by forced convection or by natural convection depending on the design of the considered core and special safety engineering features are normally incorporated into the design to ensure a safeguard of the integrity during severe postulated accident.

In case of MTR pool type research reactor, a safety feature of concern is a loss of coolant accident due to a rupture of a pipe of the primary cooling system or a break of an experimental beam tube [1] and the probability for a total or partial loss of coolant accident (LOCA) is normally reduced to very small values.

Even though during the design of research reactors, the probability for a total or partial loss of coolant accident (LOCA) is normally reduced to very small values by incorporating some engineering safety features into the design depending on the type of the research reactor, however, the demonstration of the safe behaviour of research reactors under total pool purge must be shown.

The objective of the present work is to investigate the behaviour of the first barrier i.e. the clad during a total loss of coolant accident. Under these conditions, the internal natural circulation of the fluid is suppressed and an excessive core heat up is expected to occur. The decay heat is essentially diffused by thermal radiation and heat diffusion until the settlement of a natural boiling regime that will enhance the heat diffusion by the establishment of a natural convection by air of the containment.
This hypothetical scenario has been simulated by applying the RELAP5/Mod 3.2 code system to the IAEA 10MW benchmark Reactor operating at a nominal power of 1MW and 4MW.

2 NODALIZATION

The IAEA generic research reactors [1] are representative of medium MTR power research reactors with high fissile loading and more demanding thermal-hydraulic requirements (see Figure 1). The nominal power for these reactors was fixed to 10 MW. This reactor is used here to understand the behaviour of any pool type research reactor operating at 1MW and 4MW but using a same fuel loading as of a 10 MW configuration. The detailed Benchmark specifications data related to the kinetics and thermal-hydraulic parameters are given in reference (IAEA-TECDOC-643, 1990).

![Figure 1. Typical representation of a research reactor coolant loop](image)

In order to represent this reactor by RELAP5/mod 3.2, a typical research reactor nodalization [2] was modified to represent the primary cooling system and adjusted to the benchmark boundary conditions in order to permit the simulation of the LOCA accident. This new nodalization is shown in figure 2.

The core is represented by 25 Heat structures representing 25 fuels elements positioned within the core. These fuels elements are connected to the same upper and lower plenum to specify the inner and upper boundary conditions. Each of the fuel elements was divided into 21 axial nodes and 10 radial nodes. The fluid region is also divided into 21 axial nodes. At steady state, the water level in the pool was set to 8 m. The accident was analysed for a nominal power level of 1MW. The air present above the pool and in the containment is model in a Time Dependant Volume which expands as the water level is reducing.

The nodalization was satisfactory used to recalculate the steady state of the HEU core operating at nominal power of 10MW and in the simulation of typical transients [3].
3 ACCIDENT SCENARIO

When the reactor is operating at his full power, a hypothetical event inducing pump coast down and a break in one of the primary pipes occurs. The reactor is allowed to scram when the flow rate in the core is reduced to 85% with a 200 ms delay time. In the mean time, the break induces loss of coolant and the coolant level in the core begins to reduce until a complete purge of the pool.

4 RESULTS AND DISCUSSION

In figure 3, the coolant level in the pool is represented the evolution of the pool water level during the accident (1MW case). This level falls down to zero in approximately 17 minutes after the pipe break. Also, is represented the mass flow rate at the inlet of the core (volume 210) and at the outlet plenum (105). We can observe some instabilities of the mass flow which later disappear. To the author, this instability may be induced by the great production of void at the inlet of the fuel elements as flow is reduced.

In figure 4, one can see the clad surface temperature evolution during the LOCA in the hottest channel (N° 25). During the first instants of the accident, the trend is similar to that observed in case of Loss of flow transient [3] because the coolant continue to extract the heat either by forced or by natural convection when the mass flow is reduced to 15% (the Natural Convection Valve is opened at this time). Hence, a first peak is observed induced by the buoyancy forces, due to the coolant heat up by the decay heat generation that becomes important in comparison with the decaying pump active forces. This is accompanied by a mixed convection flow followed by flow reversal and natural circulation regime establishment. The natural circulation is stopped soon after that due to the beginning of core uncovering; as the water level is drained, the clad temperature continues to rise until the complete uncovering of the core. During this period, a vapour formation is expected in the wetted part of the core; on figure 4, one can see that the vapour introduced first at the end of the fuel channel that contribute to fill in the upper plenum and soon after the lower plenum also indicating a good circulation of the air between the plates. Soon after that, the circulating
air between the plates ensures (see figure 4) natural convection which successfully stops the temperature rise before melting point. This can be noticed on figure 5 for the reactor operating at 1MW and 4MW cases just before the beginning of the accident; the clad temperature stabilises and at any time during the course of the accident, the temperature does not exceed 500 °C. This result confirms the statement of TECDOC-643 that says: only Research reactors with power over 5MW are subject to core melting after a LOCA accident.

![Figure 3: Mass flux and water level during the LOCA](image)

![Figure 4: Evolution of void in channel 25 and in the plenums](image)
5 CONCLUSION

The Loss of Coolant Accident is considered as the most severe accident in pool type research reactors. In the present framework, the HEU IAEA benchmark core configuration was used to investigate the possibility of the occurrence of clad melting during a LOCA for two different power levels. The results showed that up to 4MW, the maximum temperature of the clad in the hottest channel didn’t exceed 500°C which is far from the threshold of the melting point of aluminium. However, even though no melting was detected for a 4 MW operating research reactor, other considerations should be investigated on the damage that could be induced on the cladding thermal properties. Furthermore, a deeper investigation must be taken into account for the real operating history of a reactor in order to perform a Best Estimate simulation of the real phenomena including realistic estimation of the generated residual heat and also on the evolution of material properties after irradiation.

REFERENCES

