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Comparison between RELAP5 Mod3.2 and Mod3.3 in Simulating a Large Break LOCA in Low Power Rrs

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ABSTRACT
The assessment of reactor safety during the beyond design basis accident is oneof the lessons learned from Fukushima Daiichiaccident. In this article a total break ofone of the Neutron Beam Tubes (NBT) in a low powerpool type Research Reactor(RR) is analyzed. Two modes of RELAP5 thermal hydraulic system code, Mod3.2 and Mod3.3, are used in the analysis.In addition to the assessment of RR safety, the purpose from using two modes of RELAP5 is to assess Mod3.3 comparing with Mod 3.2 which is widely used in the analysis of RR's safety. The RR is typicallynodalized and a uniqueinput dick is prepared for the two modes.The results showed that, the reactor coolant was rapidly drained and the core was partially uncovered.On the long term transient, theclad temperature in the un-covered region wascontinuously increasedwith time. Therefore, additional measures should be implemented to overcome the consequences of the accident. On the other hand, the comparison shows,in general, good agreement between the predictions of the two modes anddemonstratesthe capability of MOD3.3 in the analysis of RR safety.

KEYWORDS: *Thermal Hydraulics; Research Reactors; Accident Analysis, RELAP5 simulation*

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I INTRODUCTION

The comparison between the results of thermal hydraulic calculation codes with experimental measurements on test facilities is a vital step in the validation processes for such codes.Therefore, many qualified test facilities all over the world were built and used for this purpose [1]. Generally, there are two types of these test facilities; Integral Test Facilities (ITFs) and Separate Effect Test facilities (SETFs). While both ITFs data and SETFs date are appropriate for code validation and assessment, the SETFs are strongly preferred for model development and improvements [2, 3]. In addition to this qualified facilities there are many other experimental work and measurements on real facilities which have been used from researchers in the validation process [4,5].

Due to the highly cost of these test facilities, benchmark problems wereused forvalidation purposes through comparing the results of different codes having the same scope and domain [6-8]. Also, comparing the results of successive versions of computational codes is another wayto assess the modifications implemented on their models in the new versions of these codes. Therefore, in the development of RELAP5/Mod3.3, anassessment matrix consisting of a thirty fourproblems (10 phenomenological problems, 19 separate effects problems, and 5 integral test problems) were usedin its assessment. In this assessment,RELAP5/Mod3.2 is used to simulate the matrix problemsand the results show that the modifications overcome a certain old problemsandindicate the other parts of the code are not adversely affected [9].

The RELAP5 best estimate Thermal Hydraulic (TH) system code is one of the codes which extensively assessed and validated for transient simulation of PWR TH during postulated accidents. The scope of the new versions of the code such as Mod 3.2 and Mod3.3 has been extended to cover the transient simulation of

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Research Reactor (RR) behavior. On the contrary to Mod 3.3, Mod 3.2 took a good opportunity ofthe assessment and validation for transient simulation of RRs [10-15].In the present work the safety of a low power RR during a Large Break Loss of Coolant Accident (LB LOCA) is assessed using the two versions of RELAP5; Mod3.2 and Mod3.3. The purpose from using the two versions is to compare Mod3.3 against Mod3.2 during a categorized beyond design basis accident.The accident isa full break of one of the NBTs attached to the reactor.Unique reactor nodalization and input deck are prepared with using the two code versions.

II REFERENCE REACTOR

A pool type research reactor with 100 KW nominal powers cooled and moderated by light water. The core is consisting of 900 LEU cylindrical fuel pins cooled by natural circulation. The pool has a diameter of 2.5 m and 7.5 m height. To promote the research and development, the reactor is provided with a number of horizontal Neutron Beam Tubes (NBTs) with diameter 0.3 m installed at a level comparable to the core horizontal centerline. The main data of the reference reactor are demonstrated in Table 1 and a schematic diagram is shown in Figure 1 [10].

Table 1 Main core data [10]

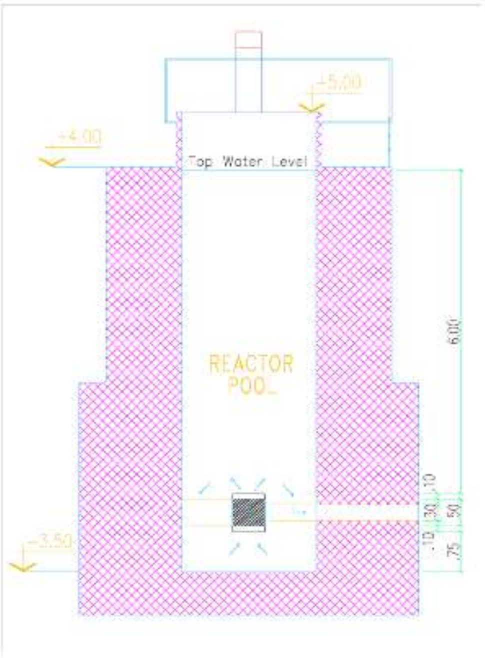
Parameter	Value
Fuel meat/Cladding material	UO_2/Zircalloy-4
Enrichment	2.1%
Reactor power	100 KW
Fuel elements shape	Cylindrical fuel rods
Pitch/Active fuel length (cm)	1.7/50
Fuel meat diameter (mm)	8.19
Cladding inner/outer diameter (mm)	8.36/9.5
Prompt neutron lifetime (Λ)	47 µs
Effective delayed neutron fraction (β_{eff})	0.00785
Axial power distribution	cosine shape
RPS scram signal	1 m decrease in pool water level
Delay time to actuate the RSS	2 s
RSS reactivity margin with single failure	-8.05 $
RSS effective insertion time	1.0 s

III ACCIDENT DESCRIPTION

The accident considered here is a full break in one of the Neutron Beam Tubes (NBT) attached to the reactor due to failing of a heavy object or due to natural disaster. This accident is a Large Break Loss of Coolant Accident (LBLOCA) and is classified as a beyond design basis accident [10]. Therefore, there are no measures considered in the design to deal with its consequences. Due to thelow elevation and large diameter of the NBT it is expected that the reactor pool will be discharged in a short time and the reactor core will be partially or totally uncovered.The selection of this accident for analysis comes in the direction of applying the lessons learned from Fukushima accident regarding the necessity of reassessment of reactorsafety during the beyond design bases accidents [15].

IV INPUT DECK AND REACTOR NODALIZATION

The reactor nodalizationIS shown in Figure (2).The core is represented by two channels;the first one is hot channel (HCH) and contains one fuel rod and the other isaverage channel (ACH) andcontains the remaining fuel rods (899 fuel rods). The two channels are divided into five axial sub-volumes and the associated fuel rods are divided into five axial heat structures. These two channels are simulated by two vertical pipes (115, 116) extending between two branches (110 and 120) which representing the core upper and lower plenums. The pool is represented by three components, a lower branch (100) corresponding to the pool lower part, pipe (130) corresponding to intermediate part, and branch (140) corresponding to the remaining part of the pool. It is assumed that there is a shroud around the core. Therefore there aren't any cross junctions between the core channels and the pool intermediate part.



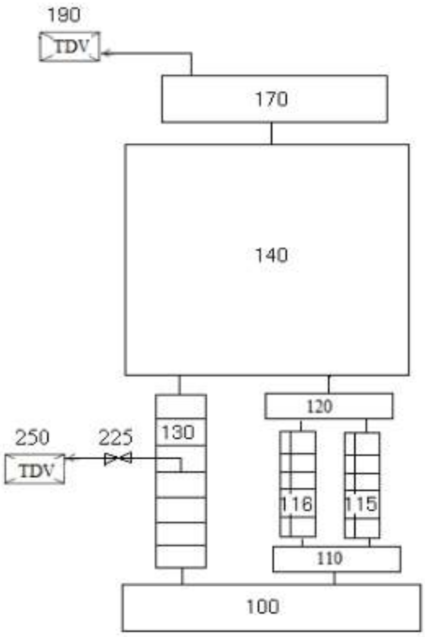


Figure (1) Schematic diagram of the reference reactor [10] Figure (2) Reactor nodalization

The heights of the lower, intermediate and upper parts of the pool are .75 m, 0.7 m and 6.05 m, respectively. The pool's upper environment is simulated by branch (170) connected to a time-dependent volume (190). The break is simulated by a valve (225) with a cross sectional area equal to that of the NBT (0,070686 m^2) and time-dependent volume (250) at atmospheric conditions identical to that considered at TDV 190. The axial power distribution in the heat structure is considered cosine shape with extrapolated length 8 cm at each core end and a total power peaking factor 3.0. The thermal properties of meat, gap, cladding, are taken from Ref. [17]. Based on the preliminary tests performed on the prepared input deck, the abrupt area change option in the control flags of the break valve 225 shall is activated. This is an important item in the accident simulation in order to avoid the unrealistic prediction in the break discharged flow rates.

V. ACCIDENT SCENARIO

After 100 seconds of steady state operation, a break occurs in one of the NBT's. This break is simulated by abrupt and full opining of valve 225. At pool water level of 1 m below the nominal value, a scram signal initiated and an actual reactor scramoccurs after 2 seconds delay.Two runs are performed with each code version, one on the short term and the other on the long term. The time sequences of the proposed eventsare summarized in Table 2.

Table 2 Time sequences of accident scenario

Event description	**Time**
Steady state operation at 100 KW nominal power	0-100 sec
Break valve opening	100 sec
Actual scram	2 sec after scram signal
Short term run	Extended for 300 sec
Long term run	Extended for 10000 sec
Maximum time step used	0.001 sec

VI. RESULT ANALYSIS

6.1- Steady State Results

Comparison of steady state calculationsforthe two modes of RELAP5 at reactor nominal power of 100 KW is presented in Table 3.As shown there is a good agreement between the results of the two modes. Thisagreement means that the used nodalization is partially qualified and the initial conditions for the transient simulation using the two modes are the same.

Table 3 Comparison between Mod 3.3 and Mod 3.2 after 200 s of steady state operation

Parameter	RELAP5 mode 3.3	RELAP5 mode 3.2
Core inlet coolant temperature (^oC)	38	38
Average heat flux of the core (W/cm^2)	0.7438	0.745
Maximum axial heat flux of the average/hot channel (W/cm^2).	0.9441/2.21	0.944/2.2
Outlet coolant temperature of the average/hot channel (^oC).	41.35/43.8	41.17/43.66
Saturation temperature at core outlet	113.6	113.6
Clad temp. at maximum heat flux in average/hot channel (^oC)	50.9/62.4	50.75/62.2
Meat temp. at maximum heat flux in average/hot channel (^oC)	60.6/85.3	60.4/85.1
Coolant mass flow rate of the average/hot channel (Kg/s)	7.45/0.011	7.48/0.011

6.2- Results of the short-term transient

On a short-term transient extends for 300 s the results of the two modes are shown below. According to the sequence of events, the transient starts with the opening of valve 225 after 100 s of steady state operation. Generally there is a good agreement in the results of the two modes. The coolant mass flow out of the breakisshown in Figure3.At the beginning of accidentthe break's mass flow rate isvery high and attains 477.9 Kg/s, where the pool is full of water and the static headwhich representingthe driving force is high. With time, the driving force and consequently the discharge flowdecreases. Finally the tank becomes nearly empty and the mass flow rategoes to zero at nearly 255 sec on the time scale, 155secafter accident initiation.

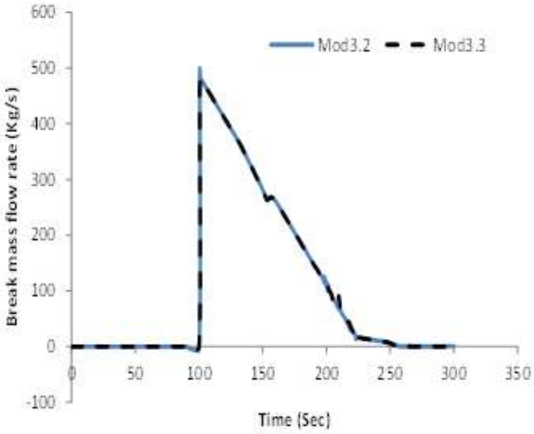


Figure 3 Break's mass flow rate

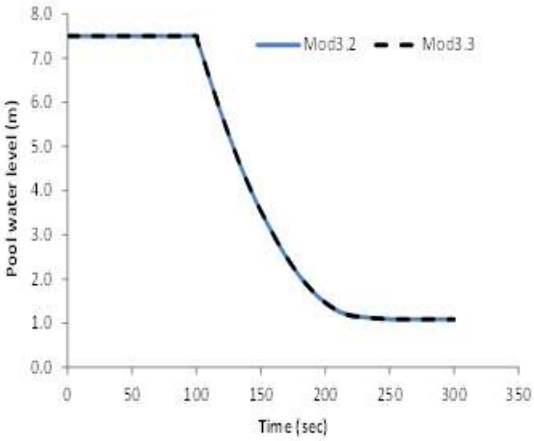


Figure 4 Pool water level (m)

The pool water level is shown in Figure 4, where good agreement appears between the results of the two
modes. After transient initiation, the water level decreases rapidly with time due to the huge break flow rate then decreases slowly until reaches the NBT level, nearly 1 m above the pool bottom. As shown, the change in the water level with time is not linear as the change in the discharge flow. This, according to the Bernoulli's equation,returns to the quadratic relation between them.

Figure 5 shows the void fractions at volumes 3 and 5 of the hot channel. As shown there is also a good agreement in the results of the two modes. After 100 s from the break's initiation, 200 s on the time axis, the core starts uncovering.Finally, the upper two volumes of the core became totally uncovered, volume No. 3 partially uncovered, and the lower two volumes remain covered with water.

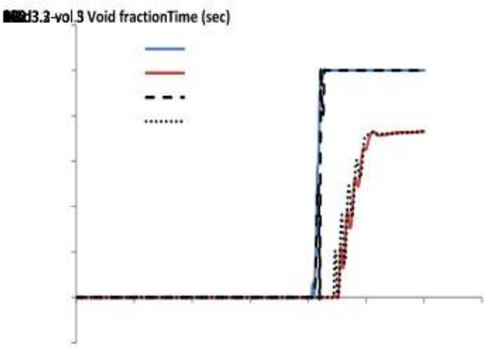


Figure 5 Void fractionsin the HCH, volumes 3, and 5

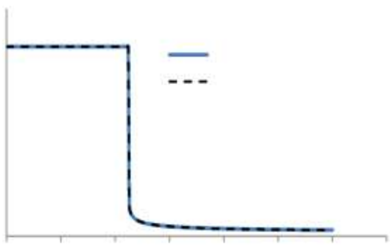


Figure 6 Reactor power

The reactor power is shown in Figure 6. The two modes show that the reactor power remains constant at 100 KW until the water level in the pool reaches the scram setting at 1m below the nominal level.At nearly 113 sa scram signal is intiated and the reactor is shuting down. Immediately after scram, the reactor power decreases to the decay power level.

The mass flow rates in the average and hot channels are shown in Figures 7, 8. Generally the figures show qualitative agreement between the two modes. Also, the coolant discharged through the core channels is very small comparing with break's flow. As shown there arethree inversions in the core flow direction before its oscillationsatnearly 200 son time axis. The timing of these inversions in the average and hot channelsare different due to the difference in the channel's coolant temperatures. The first inversion occuresimmediately after the NBT break where the downward forces causedby thebreak's flow exceed the bouncy force in the channels,therefore an upward to downward flow inversion occurs. At this moment the reactor isworking at nominal powercausing rapid increase in the channel's coolant temperatures and therefore in the bouncy forcesuntil asecond inversion from downward to upward flowoccurs in the hot and average channels at 107, 129 seconds in Mod 3.3 and 114, 138 seconds in Mod 3.2respectively.

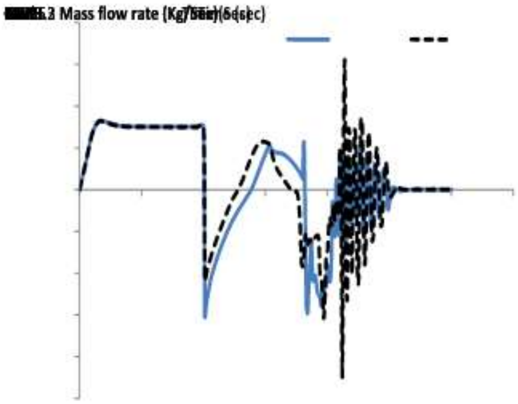


Figure 7 Coolant mass flow rate in the ACH

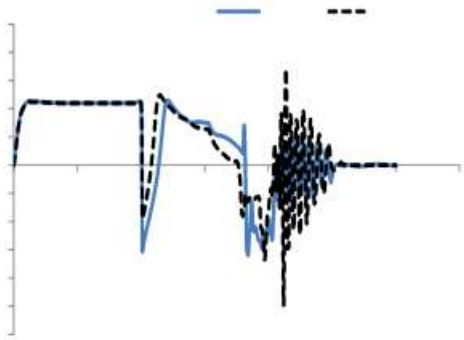


Figure 8 Coolant mass flow rate in the HCH

While the reactor is scrammed at nearly 113 s, the HCH flow reaches its maximum at nearly 115 s, and the ACH flow continually increases due to the gradual decreasing in the pool water level and consequently the downward driving force.TheACH upward flow rate reachesits maximum at nearly 154s. Increasing the upward flow rate will decrease the coolant duration time and consequently the average temperature in the channels. Therefore the bouncy forces decreases comparing with the downward force resulting from the break's flow. At 170 s a third inversion from upward to downward flow occurs in the core channels.The channel's flow remains downward until nearly 210 sec on the time scale where oscillates and finally goes to zero.

The coolant temperature at volume 5 and the clad surface temperature at volumes 3, 4, and 5 of the hot channel are shown in Figures9, 10 respectively. As shown in Figure 9, after the initiation of transient at 100 s, the coolant temperature at volume 5 drops sharply to nearly the pool temperature due to the flow inversion (first inversion), which associated with the entrance of cold water into the channels,then rapidly increases due to the heat generated in the corewhich still operates at nominal power. Amaximum coolant temperature of 323 K and 320 K is predicted by Mod 3.3 and Mod 3.2 at nearly 114and 120s respectively. The bouncy force becomes highly enough, therefore the 2nd inversion occurs andthe coolant temperature decreases due to the entrance of the cold water again into the channel.At nearly 153 s and up to the 3rd inversion at nearly 170 s the temperature changes slowly due to the comparable decreasing in the decay heat and the channel flow rate. After the 3rd inversion, the core flow becomes downward and cold water enters the channel, thereforethe volumetemperature decreases to the pool temperature until the core starts uncovering and the volume temperature progressively increases due to the core decay heat.

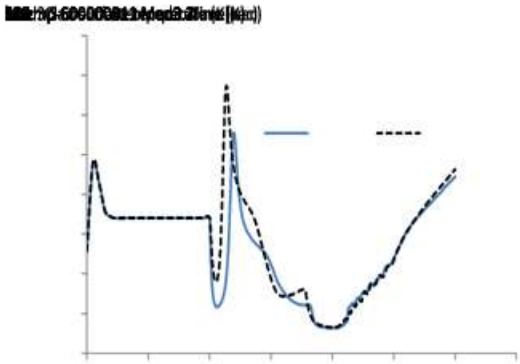


Figure 9 Coolant temperatures at HCH volume (5)

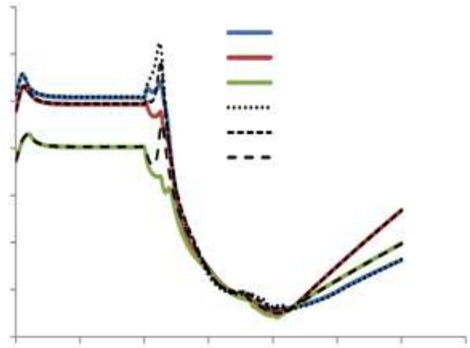


Figure 10 Clad surface temperature at the upper nodes of HCH

As shown in Figure 10,on contrary to Mod 3.2, the clad surface temperature predicted by Mod 3.3 reflects the rapid changes in the coolant temperature during the first inversion. The maximum clad temperatures predicted by Mod 3.3 and Mod 3.2 are 341 K and 337 K respectively.After reactor scram the clad surface temperature decreases in approximately similar way to the coolant temperature until reaches its minimum value at nearly 200 sec on the time axes then increases due to the decay heat and the uncovering of the core upper part.

6.3-Results of Long-term transient

On the long term, up to 10000 sec, the thermal hydraulic results of Mod3.2 in comparison with those of Mod3.3 are shown on Figures 11 to 14. The results show that the core'send state as follows; the two lower volumes are coveredwith water,the third volume partially covered, and the two upper volumesare totally uncovered. The coolant and clad surface temperatures in the three lower volumes of the hot channel (Vol.1, 2, and 3) are shown in Figures 11 and 12 respectively.Generally there is a reasonable agreement between the two versions and there is a similarity in the behavior of coolant and clad surface temperatures. As shown,the coolant temperatureinitially increases in a relatively high rate due to the core decay heat until the coolant at volume 3 reaches the saturation condition at nearly 2750 sec. After that, while Mode 3.2 predicts stabilization of the lower volume's temperatures at the sub-cooled conditions, Mode 3.3 predicts slowly increasing temperatures towards the local saturation conditions.In addition,Figure 12 shows thatthe maximum clad surface temperature in the covered part of the core is 375.5 K which is slightly higher than the local saturation temperature (~ 373 K).

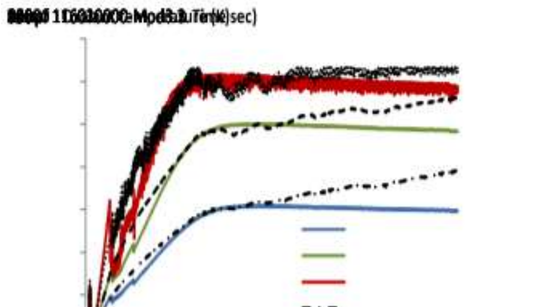


Figure 11 Coolant temperatures at vol. 1, 2 and 3 of the HCH

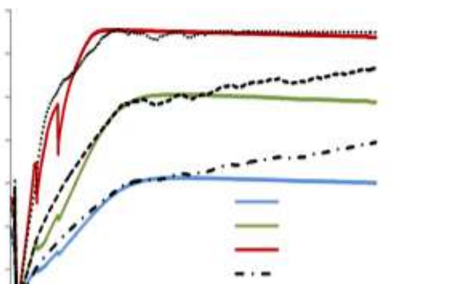
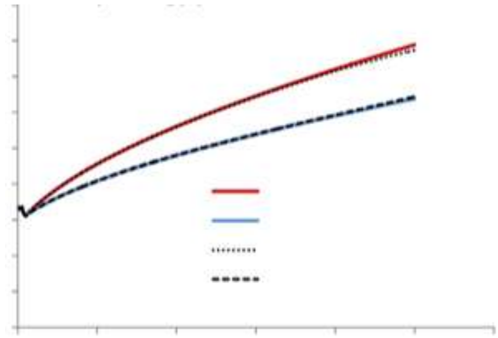


Figure 12 Clad surface temperature at vol. 1, 2, and 3 of the HCH

Figure 13 shows the temperatureevolution in the uncovered part of the core. In addition to the good agreement between the results of the two modes, thetemperatures areprogressivelyincreasesdue to the decay heat generated in the upper part of the core in addition to the loss of heat sink.Also, the heating process continues until the end time of transient. According to the initial assumption of the axial cosine distribution of power, the heat generated at heat structure 4 and consequently its temperature are greater than those at 5. The large temperature difference between heat structure 4 and 5 returns to RELAP5 code which uses one dimensional heat conduction model to calculate the temperature distribution within the heat structures.



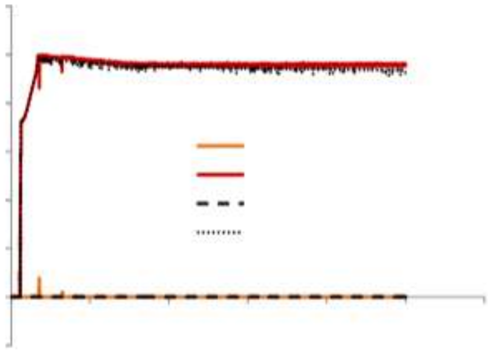


Figure 13 Clad surface temperature at vol. 4 and 5 of the HCH Figure 14 Void fractions at vol. 2 and 3 of HCH

Figure 14 shows the void fraction at volumes 2 and 3 of the hot channel. Generally, there is agood agreement between the two modes.The water level on the long term remains nearly constant up to the end of the transient. This means that, even ifthe clad temperature at volume 3 remains below the onset of nucleate boiling temperature, the two modes neglect the evaporation loss from the coolant free surface.

VII. CONCLUSIONS

In this work, an analysis for a research reactor safety during a beyond design basis accident using two modes of RELAP5, MOD3.3 and 3.2, has been presented. The reactor is a low power, pool type, and natural circulation core cooling. The accident is a large LOCA due to break of one the reactor neutron beam tubes.On the short term, the results showed that the reactor coolant was rapidly discharged in a short time and most of the core wasuncovered. On the long term, the fuel clad temperature in the un-covered part was progressively increased and attainedvery high values comparing with the temperatures in the covered part which was limited by the local saturation temperature.Therefore, it is preferred to consider additional measures to enhance the reactor safety.With respect to the comparison between the two modes of the code, the results generally show good agreement between the two modes. Also, the stabilization of the water level at core volume 3 on the long term transient needs additional in depth investigations.

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