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ESTIMATIVE OF CORE DAMAGE FREQUENCY IN IPEN`S IEA-R1 RESEARCH REACTOR DUE TO THE INITIATING EVENTS OF LOSS OF FLOW CAUSED BY CHANNEL BLOCKAGE AND LOSS OF COOLANT CAUSED BY LARGE RUPTURE IN THE PIPE OF THE PRIMARY CIRCUIT - PSA LEVEL 1

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ABSTRACT

This work applies the methodology of Probabilistic Safety Assessment Level 1 to the research reactor IEA-R1 IPEN-CNEN/SP. Two categories of identified initiating events of accidents in the reactor are studied: loss of flow and loss of primary coolant. Among the initiating events, blockage of flow channel and loss of cooling fluid caused by large pipe rupture in the primary circuit are chosen for a detailed analysis. The event tree technique is used to analyze the evolution of the accident, including the actuation or the fail of actuation of the safety systems and the reactor damages. Using the fault tree the reliability of the following reactor safety systems is evaluated: reactor shutdown system, isolation of the reactor pool, Emergency Core Cooling System (ECCS) and the electric system. Estimative for the frequency of damage to the reactor core and the probability of failure of the analyzed systems are calculated. The estimated values for the frequencies of core damage are within the expected margins and are of the same order of magnitude as those found for similar reactors. The reliability of the reactor shutdown system, isolation of the reactor pool and ECCS are satisfactory for the conditions in which these systems are required. However, for the electric system it is suggested an upgrade to increase its reliability.

1. INTRODUCTION

The PSA is an important tool to quantify the risk of operating a nuclear reactor or other plants with potential hazard, being used either in the design phase as in the operational phase of the plant. Through the PSA analysis the probability of occurrence of an accident is obtained and its consequences are evaluated, providing a numerical estimative which will indicate how safe the installation is. The PSA analysis can also be used to estimate the risk reduction that could be achieved with the adoption of changes in the plant design or in the operation and maintenance practices.

In the PSA Level 1 the sequences of events that can lead to the loss of integrity of the fuel or core damages and their probabilities of occurrence are identified and quantified.

The objective of this paper is to present the main results obtained in [1], where part of the Probabilistic Safety Assessment (PSA) Level 1 was conducted for the research reactor

IEA-R1. In the study in reference [1], it was qualitatively evaluated the initiating events of accidents identified and described in the Safety Analysis Report (SAR) of the IEA-R1 [2] for two categories of initiating events: the loss of flow and the loss of primary coolant. The list of initiating events for both categories is listed in Table 1.

Among all events with potential to cause the meltdown of fuel elements, the event envolving the blockage of one channel only is the one with the highest probability of occurrence [3].

From the initiating events in the loss of primary coolant category listed in Table 1, the event of large LOCA is the one which causes the largest consequences due to the possibility of uncovering the core in less time than all other events in this category (approximately 6 minutes) [2].

It was analyzed the evolution of the accident for two initiating events selected from those listed in Table 1. Taking as selection criteria the event with the highest probability of occurrence and that result in more severe consequences for the reactor core, it was analyzed the following events:

(a) Loss of flow: due to blocked channel;

(b) Loss of coolant accident (LOCA) due to large rupture .

In [1] was calculated the estimative of core damage frequency in the IEA-R1 due to the occurrence of the initiating events (a) and (b). The evolution of the accident and performance of the systems which should mitigate each event were analyzed by means of the event tree (ET). Furthermore, the reliability of these systems was quantified using the fault tree (FT).

Table 1. Initiating events for LOCA and LOFA categories

Category	Initiating Event
Loss of Coolant	Rupture of primary circuit
	Pool damage
	Loss of water pool 1. Loss of water through retreatment system; 2. Loss of water through drain.
	Failure in irradiation pipes
	Failure of the primary circuit drain
	Failure in pneumatic pipes of irradiation material
	Failure in thermal column
Loss of Flow	Primary circuit pump failure: in the power supply and locking the pump shaft Inadvertent closure of reactor pool isolation valves Fuel cooling channel blockage Flow reduction due to core flow bypass Loss of heat sink Primary coolant flow reduction Improper power distribution due to, for example, unbalanced rod position, in-core experiments or fuel loading Malfunctioning of reactor power control

2. ACCIDENT ANALYSIS

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The following considerations were taken for the analysis of the two initiating events:

- before the occurrence of the initiating event the reactor is in normal operation at full power;
- before the occurrence of the initiating event the safety system, all electrical power supply and support systems are available;
- after the initiating event of rupture of the primary pipe, it is considered that the reactor protection system acts successfully in order to perform the shutdown of the reactor;
- the occurrence of initiating events and failures are independent;
- the failures are an exponential distribution, i.e., the failure rates are constant in time;
- components are considered non-repairable during the adopted observation period;
- Initiating event occurs during one of the reactor's operation period. Each operation period lasts 63 hours which will be the value of mission time adopted for the analysis.

2.1 Channel blockage.

The blockage of the entrance of primary coolant through a channel or channels of a fuel element can occur when objects inadvertently falls on the core. The reduction of coolant flow can cause a local overheating of the fuel element plate followed by failure of the cladding.

The detection of this kind of event can be done:

- by the operator, by visual inspection during the operation;
- by a significant increase in pressure loss in the core, measured by pressure transducer located at the top of the pool (corresponding to a value above 10% of nominal flow);
- by significant increase in coolant temperature at the out of the core (above 48ºC);
- in the worst case, by the radiation detectors positioned below the movable platform supporting the core. In this case, some kind of damage already occurred.

There are not resources available in the reactor which allows the automatic detection of the initiating event when few channels are blocked, because in this situation the detectors of differential pressure and temperature increase at the out of the coolant system cannot detect small variations. If there is no detection by the operators (visually), the reactor will not be shut down and may cause local damage on the fuel element plates.

In the event of deterioration in the cooling of the fuel element plates, where channels were blocked, it can lead them to their melt. In this case, there will be the release of fission products to the pool water and to the atmosphere of the containment, with its detection by radiation monitors and automatic shutdown of the reactor through the protection system. The regular ventilation system of the Hot Area will be turned off and the emergency ventilation system will be activated and the isolation of this area. Therefore, the emergency ventilation system forwards the air to the filters decreasing the release of radioactive material to the environment.

After its detection (by the operator), to mitigate a blocked channel are necessary the following safety functions so that there is no damage to the core and the release of radioactivity to the environment do not be above the permissible limits:

- shutdown of the reactor by the operator on the reactor protection system;

- maintenance of the containment with the turn off the regular ventilation system and activation of the emergency and isolation ventilation system.

The expected sequences of events in this case are:

- blockage of few cooling channels of a fuel element caused by some object, without possibility of automatic detection;
- visual detection of the event by the operators and manual shutdown of the reactor;
- turn off the regular exhaustion and insufflation of the Hot Area, and start of the emergency exhaustion of this area;
- isolation of the Hot Area.

The ET, Figure 1, shows the evolution of accidental sequences. Only one sequence leads to a state without damage to the core (SEQ. 1), where the shutdown of the reactor is initiated by the action of the operator if the blockage is detected by visual inspection. The other sequences (SEQ. 2, SEQ. 3 and SEQ. 4) lead to a state with local damage to the core. In these sequences, the actions of the emergency exhaustion and isolation of Hot Area do not avoid core damage, but only act to minimize the consequences of the accident.

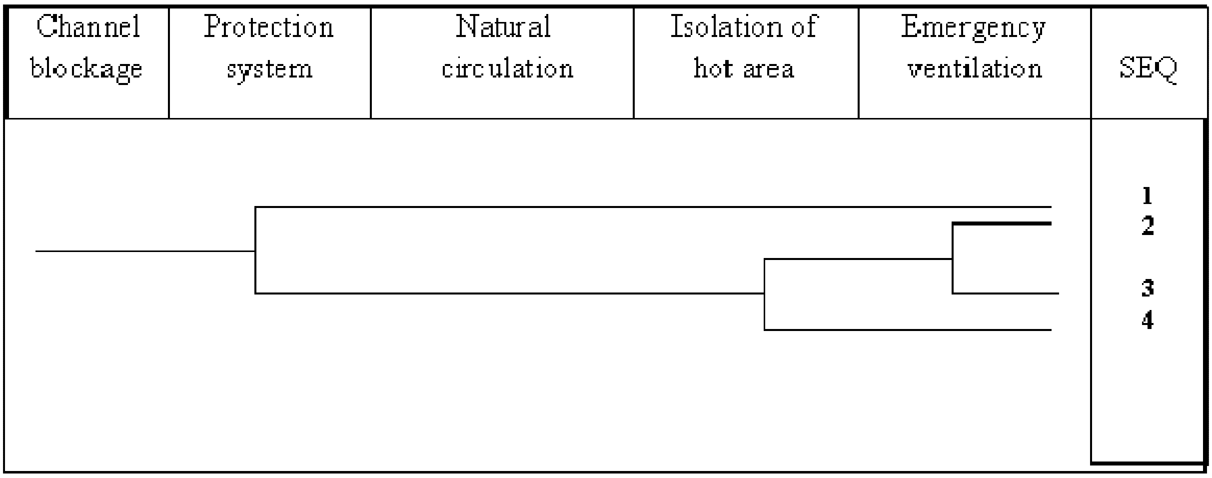

Figure 1: ET for channel blockage.

The occurrence frequency of the initiating event of channel blockage was obtained from studies for reactors similar to the IEA-R1. Greek reactor frequency is equal to 10^{-2} per year [3], and Australian reactor frequency is equal to 1.3×10^{-5} / year [4]. In this study the largest value was adopted. Both Greek and Australian reactors are open-pool type research reactors and they use plate-type fuel assemblies, like the IEA-R1. However, Australian reactor has a nominal fission power of 20 Mega-watts.

Using the SAPHIRE program [5] and failure data obtained from [3] [4], [6], [7] it was calculated the probability of failure in the shutdown of the reactor in case of channel blockage and was obtained the value of 2.97×10^{-2}.

This value is strongly influenced by the probability of error of the operator. The main human errors which contribute with more than 99% of the estimated value are:

- the operator doesn’t detect the channel blockage during visual inspection;
- the operator doesn’t proceed the visual inspection;
- the operator doesn’t initiate the reactor shutdown process, after detecting the channel blockage.

From the probability value of failure in the shutdown of the reactor (2.9×10^{-2}) and from the frequency of the initiating event of blockage of the cooling channel of the IEA-R1 core (10^{-2} per year), the estimated value for the core damage frequency due to blockage of the channel, is about 2.9×10^{-4}/year.

Although the estimated value can be considered high, it is acceptable, because this type of accident would cause only minor damage to the core which wouldn't cause large releases of radionuclides to the environment. It is only a local damage to few fuel plates and IEA-R1 reactor has containment and systems that mitigate potential releases of radiation above the limits permissible for the population. Furthermore, when this result is compared with other research reactors it is of the same order of magnitude of the Greek reactor [3] and approximately 10 times greater than that obtained for the Australian reactor, Ansto [4].

For other initiating events in the category of loss of primary coolant flow, it was concluded through the analysis performed in reference [2] that only the case of channel blockage could lead to damage to the reactor core.

This conclusion is based on the consideration that the establishment of natural circulation of coolant through the core would be enough to mitigate the occurrence of the other initiating events, removing the residual heat of it. However, this depends on the decoupling of the convection valve that can fail. If there is no establishment of natural circulation there might be damage to the reactor core for some of the initiating events described.

Considering the failure in establishing the natural circulation, the most critical situation would be the initiating event of locking the pump shaft, because the flywheel would not act and the forced circulation would be interrupted, consequently there would be a greater amount of residual heat to be removed.

2.2 Large rupture in primary circuit pipe

The postulated event would be a complete rupture, guillotine type, of the primary coolant return pipe, next to the pool, that could lead to pool emptiness in about 6 minutes [2]. Once the primary circuit operates in low pressure and temperature, the guillotine rupture of the pipe would happen only by means of missile. However, the circuit is well protected against external events, and high magnitude earthquakes or aircraft falls occurrences are very unlikely according to the references [2].

The expected sequence of events for the case of rupture of the primary circuit would be as follows [2]:

- rupture of the 10” pipe of the primary circuit (next to the return to the pool);
- alarm signal of low water level in 200 mm below normal level;
- automatic reactor shutdown when water level reaches 350 mm below normal level;

- automatic shutdown of the primary pump and closing of the isolation valves of primary circuit when water level reaches 400 mm below normal level.
- the time of closure of isolation valves is expected to be around 30 to 60 seconds, ensuring a minimum final level of water in the pool between 6.0 and 7.5 meters above the bottom;
- with the pool isolated and core covered, there will be decoupling of the convection valve, starting the cooling by natural circulation, which is sufficient to remove the decay heat and maintain the core at low temperatures;
- in case of no decoupling of the convection valve, the natural circulation will not be established and can cause local damage in the fuel plates [3];
- in case of failure in closing the isolation valves after the rupture of the pipe, the total emptying of the pool will occur in about 6 minutes. When the water level in the pool reaches 4,500 mm below the normal level, the Emergency Core Coolant System (ECCS), of passive action, is activated; which will ensure the cooling of the core.

The development of the accident after the initiating event is show in the event tree of the Figure 2. The four resulting accidental sequences are:

- **SEQ1**: rupture of the pipe and isolation system of the pool operating successfully, decoupling of the convection valve and consequent establishment of natural circulation. This sequence leads to a final state without core damage;
- **SEQ2**: rupture of the pipe and isolation system of the pool operating successfully and failure in the decoupling of the convection valve without the establishment of natural circulation. This sequence may lead to a final state with local damage in the fuel [3];
- **SEQ3**: rupture of the pipe with failure in the isolation system of the pool and emergency coolant core system (ECCS) successfully working. This sequence leads to a final state without core damage, because the ECCS was designed to cool the core and remove the decay heat in this situation. It should be emphasized that this sequence leads to loss of radiation shielding provided by the pool water, resulting in direct exposure of the reactor core and, consequently, in high doses in the pool lobby and possibly inside the reactor building;
- **SEQ4**: rupture of the pipe with failure in the isolation system of the pool and failure in the performance of the emergency coolant core system (ECCS). This sequence leads to a final state with core damage, because the core is uncovered. This scenario is the most severe with melting of fuel, and there may be release of radioactivity. It must be emphasized that the reactor has a ventilation system that should act to control potential releases of radioactivity in this accidental condition.

The frequency of occurrence of the four accidental sequences described above depends on the following values:

- frequency of rupture of the pipe of the primary circuit;
- probability of failure in isolating the pool;
- probability of failure in the performance of the natural circulation;
- probability of failure of the ECCS.

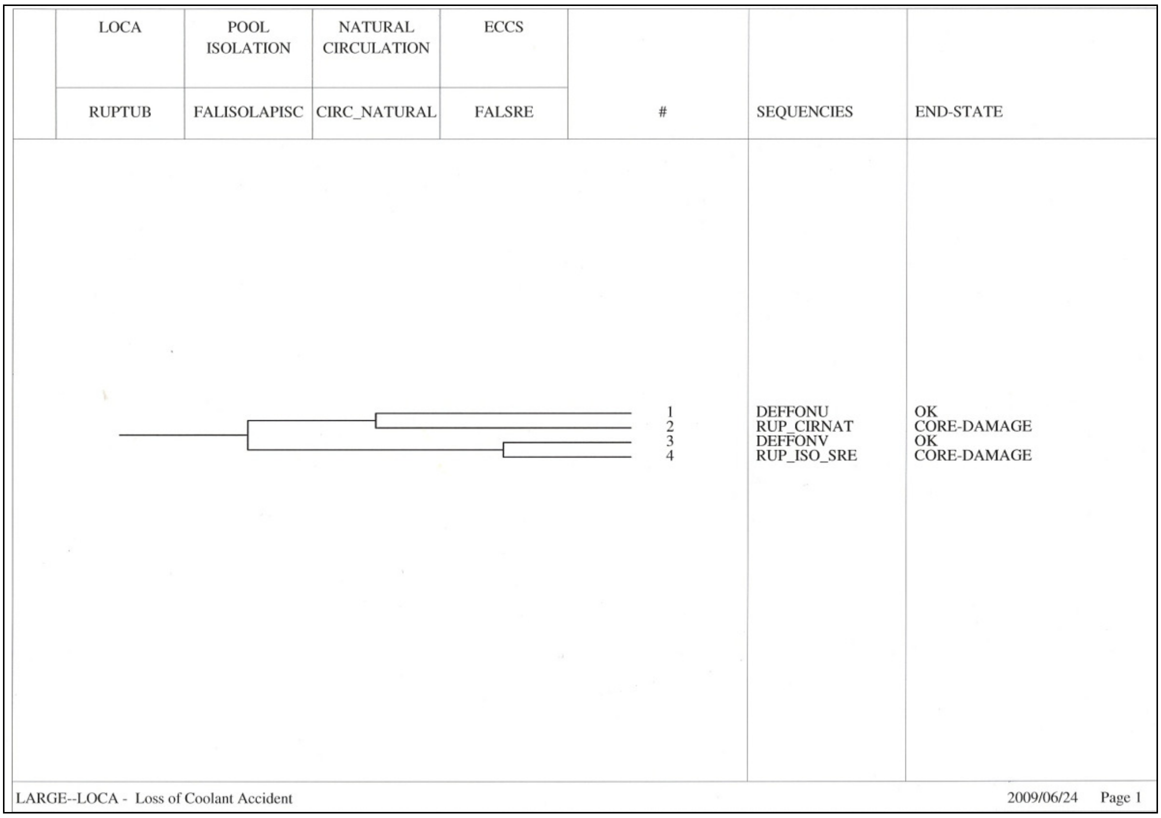

Figure 2. Event tree for the initiating event LOCA

In order to obtain the probability of failure of both the isolation system of the pool and the emergency cooling system (ECCS) was necessary to obtain the probability of failure in the supply of electric energy in the following electric panels:

- motor control center of 440 V– vital bus;
- motor control center of 440 V– essential bus;
- electric distribution panel of 220 V – vital.

The probabilities of failure were obtained using the fault tree with program SAPHIRE [5] and data of failures obtained from generic databases [6, 8], data from similar plant [3,4] and specific data from IEA-R1 [8].

The results obtained are:

- probability of failure in the pool isolation = 1.53E-03;
- probability of failure of the ECCS = 1.97E-04.

The probability of failure in natural circulation was obtained from reference [3] equal 1.008E-02.

The rupture frequency of the primary circuit from reference [3] = 1.2E-04/year.

Using the program SAPHIRE and values from above, it was obtained the frequencies of occurrence of the sequences that lead to core damage (SEQ2 and SEQ4), which are equal to 1.21E-06 and 1.35E-10 per year, respectively.

3. CONCLUSIONS

In the case of channel blockage, the reactor shutdown depends on the actions of the operator; therefore the value obtained for the probability of failure in the reactor shutdown is strongly influenced by the probability of human error.

The core damage frequency obtained for the event of channel blockage is about 2.9E-04/year. This value, which can be considered high, would cause only minor local damage to the core, and consequently without large releases of radionuclides to the environment. Comparing the estimated frequency with some research reactors, the result obtained in the reference [1] can be considered satisfactory, as it is of the same order of magnitude as the Greek reactor [3] and approximately 10 times higher than Australian reactor Ansto [4].

In the analysis of the initiating event of loss of coolant by larger rupture of the primary circuit, it was found two accidental sequences which lead to damage to the core: SEQ. 2 and SEQ. 4. The following values were obtained for the occurrence frequency of these accidental sequences:

- frequency of occurrence of SEQ. 2 equal to 1.21E-06 per year;
- frequency of occurrence of SEQ. 4 equal to 1.177E-12 per year.

For the conditions analyzed, the pool isolation system and the ECCS showed good reliability. This is due to the adoption of redundancies in the most sensitive points of these systems.

For the electrical system, the CCM-E/V-11 buses presented relatively high probabilities of failure in electric power supply. This is due to the fact that: a single failure of various components leads to loss of power in these buses. Although these values are high, the redundant valves of the same side of the primary circuit are supplied in the crossover mode. Therefore, in order to happen a failure in the isolation of the pool is necessary that both buses of the CCM-E/V-11 remain simultaneously without power (i.e. at least two components of the electrical system should fail).

As expected, SEQ. 4 showed a very low frequency of occurrence and its occurrence can be considered not credible.

The value obtained for frequency of occurrence for SEQ. 2 is comparable to those obtained in studies conducted in two research reactors cited in the references [3, 4].

The SEQ. 3 is not a totally safe scenario, because it leads to the uncovering of the core, showing a frequency of about 1E-07/ year.

From the results obtained in the reference [1], it is concluded that the safety systems and operational processes of the reactor IEA-R1 present a satisfactory performance, when compared with other research reactors.

However, some recommendations are suggested in order to increase the safety of the reactor:

- installation of an automatic system for detection of channel blockage;
- upgrade of the electrical system of the IEA-R1 in order to improve its reliability and availability;
- verification of the probability of failure of the convection valve decoupling, and consequently of the establishment of natural circulation when needed.

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