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PRELIMINARY STUDY OF PROBABILISTIC SAFETY ASSESSMENT LEVEL 1 FOR THE IEA-R1 RESEARCH REACTOR OF THE IPEN/CNEN

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ABSTRACT

Due to the fact that nuclear energy is today one of the most important supply source of "clean" energy, increases the possibility of investments in new nuclear power plants in Brazil. The National Commission of Nuclear Energy (CNEN), that is the Brazilian nuclear regulatory commission, imposes safety and licensing standards in order to ensure that the nuclear power plants operate in a safe way. For licensing a nuclear reactor one of the demands of CNEN is the simulation of some accidents and thermal-hydraulic transients considered as design base to verify the integrity of the plant when submitted to adverse conditions. The accidents that must be simulated are those that present large probability to occur or those that can cause more serious consequences. These accidents are previously identified through a Probabilistic Safety Assessment (PSA). The objective of this paper is to present a preliminary study which information will be used in a future Probabilistic Safety Assessment Level 1 for the IEA-R1 Research Reactor of IPEN/CNEN-SP.

1. INTRODUCTION

Due to the fact that nuclear energy is today one of the most important supply source of "clean" energy, increases the possibility of investments in new nuclear power plants in Brazil. The National Commission of Nuclear Energy (CNEN), the Brazilian Nuclear Regulatory Commission, imposes safety and licensing standards in order to ensure that the nuclear power plants operate in a safe way.

For licensing a nuclear reactor one of the demands of CNEN is the simulation of some accidents and thermal-hydraulic transients considered as design basis to verify the integrity of the plant when submitted to adverse conditions [1].

The accidents that must be simulated are those that present large probability to occur or those that can cause more serious consequences. These accidents are previously identified through a Probabilistic Safety Assessment (PSA).

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 a preliminary analysis of the PSA Level 1 [2, 3] for the Research Reactor IEA-R1 of IPEN/CNEN-SP. In this analysis the following steps will be performed:

- Identification of the main release source of radioactivity to be considered;
- Identification of accidents initiating events;
- Identification of the accidental sequence with the highest probability of occurrence; and
- Analysis of this accidental sequence.

2. PRELIMINARY STUDY OF THE PSA LEVEL 1 OF IEA-R1

The phases considered in the execution of the preliminary study of the PSA Level 1 of IEA-R1 are described below.

2.1. Plant Description

The IEA-R1 is a 5MW pool type research reactor. Its core is basically constituted by a set of fuel elements of Material Test Reactor (MTR) type submersed in the pool. The reactor fixed in a metallic structure and cooled with light water and its cooling is obtained by the flow of the water of the pool through the fuel elements.

This reactor was designed and built by Babcock & Wilcox Co., in cooperation with CNEN and it was financed by the American program - Atoms for Peace. The IEA-R1 operated during its first three years at a maximum power of 1MW and from 1960 to 1995 at a power of 2 MW. With the growing demand and the advent of new applications for radioisotopes, for medical use, the IEA-R1 was modified to increase the operating power to 5 MW. The increase in power from 1 MW to 5 MW demanded a complete revision in the installation which resulted in modifications in many systems, replacement of some structures and the incorporation of new systems in order to guarantee the safety of the installation. These modifications are the motivation of the work.

2.2. Identification of the Radioactive Main Source Release

This study considers as radioactive release source the fuel present in the core of the reactor, It does not consider possible radioactive materials stored in the plant due to their lower potential of liberation when compared to the fuel in the core.

2.3. Operation Mode

The analysis will be performed considering the normal operation at full power (5 MW). At the safety point of view the normal operation at full power covers all others stages of operation (startup, shutdown, etc.).

2.4. Identification of the Initiating Events.

The identification and selection of the initiating events, i.e., the occurrences that can lead to (depending on the actuation or not of the mitigating systems of the plant) accidental scenarios with core damage, was carried out in Chapter 16 of the Final Safety Analysis Report (FSAR) of IEA-R1 [4]. The method used to identify the initiating events is based on the Safety Series 35 [1] and consists of several stages that include:

- Identification of a preliminary set of initiating events;
- Elimination of the inadequate events;
- Gathering of the events in categories; and
- Identification of the restraining events (events with the worst consequence in a category).

The preliminary list of accidents initiating events was obtained from a generic list, for research reactors, presented in Table I of Safety Series 35 [2]. It was eliminated from this list the inconsistent or inappropriate initiating events, non credible events (impossible to occur in IEA-R1), very rare events (initiating events of which occurrence frequency is so small that can they can be disregarded in the probabilistic field) and the events resulting from the combination of events mutually independents with low occurrence frequency.

New events were incorporated in the reduced list based on the operational experience of 39 years of the IEA-R1 operation and on verification of other safety reports from similar installations. After the list was completed, the initiating events were gathered in categories of events that can cause similar influences over the reactor behavior. The categories adopted are:

- Loss of electric power supply;
- Insertion of excess of reactivity;
- Loss of flow
- Loss of coolant;
- Erroneous handling or failure of equipment components;
- Special internal initiating events;
- External event; and
- Human error.

The initiating events identified for the IEA-R1 are presented in Table 1.

2.5. Analysis of the Accidental Sequences

The accidental sequences were described in Chapter 16 of the FSAR of IEA-R1. These sequences were evaluated starting with the occurrence of the initiating event until the final stage of damages in the reactor, following the entire development of the accident. It was considered in this evaluation the human interactions as well as the important systems for the elimination or mitigation of the accident consequences, including the reactor protection system and the safety systems. Moreover, not only the primary barrier, constituted by the fuel cladding, was considered, but also the reactor pool and the reactor containment.

Table 1 – Initiating events for the IEA-R1 reactor

Loss of Electric Power Supplies
Normal Power Supply System failure Essential Power Supply System failure Vital Power Supply System failure
Insertion of Excess Reactivity
Criticality during fuel handling (fuel insertion error) Start-up accident Control rod failure Control rod drive mechanism failure Unbalanced rod positions Failure or collapse of structural components Cold water insertion Moderator changes Influence from experiments and experimental facilities Insufficient shutdown reactivity Inadvertent rod withdrawal Dropping of a fuel element
Loss of Flow
Primary circuit pump failure 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
Loss of Coolant
Primary circuit boundary rupture Reactor pool damage Total loss of pool water Damage to beam tubes or other penetrations
Erroneous handling or failure of equipment components
Fuel element cladding failure Mechanical damage to core or fuel element Criticality in fuel storage Containment or ventilation system failure Loss of coolant to fuel during transfer or storage Loss or reduction of proper shielding Failure of experimental apparatus or material Exceeding fuel ratings
Special Internal Initiating Events
Internal fires or explosions Internal flooding Loss of support systems Problems with physical security Improper access to restrict
Human Errors
Pre-operational human error Human error during reactor operation management Post-operational human error
External Events
Earthquake Flood Tornadoes, tornadoes missiles and hurricanes Storming and lightning Aircraft crashes Fires and explosions Release of toxic products Transportation vehicles accident Influence from adjacent facilities

During the study of the sequence of events, the following issues were identified:

- The significant occurrences along the time, of the same events, for instance: reactor shutdown, startup and the end of the safety rods insertion, etc.
- The correct or incorrect functioning of the reactor instruments and controls in normal conditions and
- The required actions from the operator.

The three main safety functions were also evaluated: reactor shutdown, fuel cooling and radionuclide confinement, including an indicator of the simultaneous correct functioning of the reactor protection systems and safety systems and their failures.

Among all events with potential to cause the meltdown of fuel elements, the accident of one channel blockage is the one with the highest probability of occurrence [5].There are several causes for the event of channel blockage. Among the possible causes, the following ones distinguish from the others:

- Fall of an object in the reactor pool and its posterior deposit on the top of the core;
- Failure of some experimental device, present in the interior or around the core, resulting in detachment of parts or objects, which later on come to deposit in the core or cause a deformation in a cooling channel; and
- Failure in some mechanism or equipment, for instance, the control and safety bars movement mechanism in resulting either in the detachment of parts or objects, which later on come to deposit in the core, or in deformation of a cooling channel.

From the three causes presented above, the worst and also the most probable is the fall of an object in the pool and its posterior deposition in the core. This is caused because there is not an upper limit for the size of the object generator of the accident.

In case of occurrence of this event, there might be the obstruction of one or more cooling channels of the fuel elements. It is more likely that such events involve only small objects and few fuel channels, however, as mentioned, unfortunately there isn't an upper limit for the size of the object generator of the accident. Therefore, for the IEA-R1 reactor is postulated the obstruction of one and the maximum of five fuel elements. It is observed that, in the IEA-R1 reactor, the core configuration with four control elements with bigger length than the other elements positioned doesn’t allow the obstruction of more than five elements, according to the FSAR.

It is observed that the event of a blockage, which involves five fuel elements, is not expected to occur in the useful life of the installation. However, this event consists in an upper limit, in terms of damages to the core as well as in terms of radiological consequences, which no event with higher probability of occurrence has conditions to exceed [4].

The sequence of events expected in this accident is as follows:

- Blockage of one to five channel(s) of fuel elements caused by some object;
- Deterioration in the cooling of the obstructed fuel element(s) and consequent fusion of part of them;
- Release of products of the fission to the pool water and to the atmosphere in the containment;
- Indication of high level of radiation by the radiation monitors;
- Automatic shutdown of the reactor due to the high level of radiation;
- Containment isolation
- Automatic turning off of the normal exhaustion and of the hot area swelling and the start of operation of the emergency exhaustion of the hot area.

The event tree that shows the evolution of this accident with the dependency on the actuation or not of the mitigation systems of the installation is shown in Figure 1.

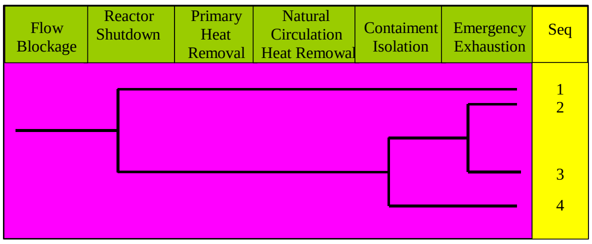

Figure 1.Event Tree for Flow Blockage

In the subsequent stages of this work, it must be performed on analysis for each accidental sequences deriving from the accidents initiating events which are presented in Table 1 will be performed. Fore each accident sequence will be quantified the estimative for the extension of damages to the core, the reliability of the plant safety systems and the frequency of occurrence. This last set of evaluations is part of a master under development at IPEN-CNEN/SP.

3. CONCLUSIONS

This work shows the results of the preliminary studies for the elaboration of a Probabilistic Safety Assessment Level 1 for the IEA-R1 Research Reactor of the IPEN. In this phase some considerations to be adopted in the PSA were raised and the following results were obtained:

- Identification of the fuel of the core as the radiation source to be considered;
- Identification of the accidents initiating events;
- Identification of the accidental sequence resulting from a channel blockage as the one with the highest probability of occurrence (to be confirmed in quantitative evaluations);
- Qualitative analysis of this accidental sequence; and
- Event tree for this accidental sequence.

The next stages of this work, 6, will be performed in the future in the master of the Daniel Massami Hirata.

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