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Introduction to Safety Analysis Approach for Research Reactors

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1. Introduction

According to the IAEA research reactor database [1], 244 research reactors are operating and 18 research reactors are under construction or planned. The research reactors have a wide variety in terms of thermal powers, coolants, moderators, reflectors, fuels, reactor tanks and pools, flow direction in the core, and the operating pressure and temperature of the cooling system. Around 110 research reactors have a thermal power greater than 1 MW.

This paper introduces a general approach to safety analysis for research reactors and deals with the experience of safety analysis on a 10 MW research reactor with an open-pool and open-tank reactor and a downward flow in the reactor core during normal operation.

2. General Approach to Safety Analysis

2.1 Identification of Initiating Events

A complete set of initiating events has to be identified and set by considering operating experience of research reactors such as Incident Reporting System for Research Reactor (IRSRR) database and design features of a specific research reactor that safety analysis is carried out. Incredible initiating events and inherently excluded events by design features can be eliminated from the set of initiating events. The safety requirements and specific safety guide for research reactors [2, 3] provide selected initiating events, which are a starting point to establish the complete set of initiating events for safety analysis. The documents categorize the initiating events as follows:

- Loss of electrical power supplies,
- Insertion of excess reactivity,
- Loss of flow,
- Loss of coolant,
- Erroneous handling or failure of equipment or components,
- Special internal events,
- External events,
- Human errors.

2.2 Classification of Initiating Events

The document of safety requirements [2] classifies the reactor states into operational states and accident conditions, which are additionally divided into normal operation, anticipated operational occurrences (AOOs), design basis accidents (DBAs) and beyond design basis accidents (BDBAs). The classification of reactor states is newly modified as normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation and with core melting in the draft specific safety requirements [4].

The initiating events should be classified into the reactor states by considering the expected occurrence frequency in each initiating event. The classification of initiating events can be determined by probabilistic analysis and engineering judgement based on the experience of similar research reactors.

2.3 Acceptance Criteria

Acceptance criteria should be established to judge the acceptability of safety analysis results for both the operational states and the accident conditions. Radiological criteria such as dose limits to the public, the environment, and occupied personnel including experimenters and workers at the reactor site should be set by the regulatory framework. The dose limits depend on the regulations of countries. Nuclear fuel performance criteria to judge the integrity of first barrier such as fuel cladding should be defined with additional margins to fuel failure as follows: (a) blistering temperature, (b) critical heat flux (CHF), (c) onset of significant voiding (OSV), (d) onset of flow instability (OFI). These specific acceptance criteria may be defined by the designer or the operating organization satisfactory to the regulatory body [5].

Blister may cause that fission gases are diffused and released from the fuel meat and that the flow area between fuel plates can be reduced. Thus, fuel cladding temperature is limited to avoid blister during AOOs. It is reported that the blister threshold temperature of U_3Si_2 fuel plates ranges from 515℃ to 575℃ [6]

Critical heat flux ratio (CHFR) is used as a design limit parameter to prevent fuel damage during AOOs. The correlations of Kaminaga [7] and Mirshak [8] have been widely used for plate-type fuels to predict critical heat flux. The occurrence of Ledinegg-type instability in plate-type fuels can lead to flow redistribution in cooling channels and then to reduce flow rate in the hot channel causing burnout or critical heat flux. Accordingly, onset of flow instability ratio (OFIR) is considered conservatively as a design limit parameter to avoid fuel damage. The Whittle and Forgan correlation [9] has been usually used to predict the heat flux at the onset of flow instability. The significant void generation can lead to onset of flow instability and burnout or CHF. Thus, onset of significant void ratio (OSVR) can be used a design limit parameters. The design limits of the parameters should be developed by statistical analysis on the uncertainties of CHF, OFI and OSV correlations.

Void generation in the reactor core is unfavorable in research reactors during normal operation due to a disturbance of reactivity control. Hence, onset of nucleate boiling ratio is a design limit parameter for the core thermal hydraulic design in normal operation. The Bergles-Rohsenow correlation is applied to predict the onset of nucleate boiling heat flux.

2.4 Methods for Transient Analysis

Combined approach is widely used for safety analysis of research reactors [5]. The combined approach is to use best-estimate codes with conservative assumptions on availability of systems in modeling the sequences of event and with conservative input data on initial and boundary conditions [10]. The best-estimate codes such as RELAP5, CATHARE and ATHELET have been widely used for thermal hydraulic system analysis. PARET code [11] has been used to assess reactivity-induced events and core thermal hydraulic transients without modeling of cooling systems.

Limiting values covering the normal operation ranges of flow, temperature, pressure and pool level including their measurement uncertainties and the nuclear design data including calculation uncertainties are selected as the initial conditions for safety analysis. Uncertainties of fuel manufacturing and thermal hydraulic correlations are considered as hot channel factors in calculating fuel temperature and CHFR. Evaluation methods for the hot channel factors have been studied [12, 13].

The rules or conventions regarding the extent to which reactor systems are assumed to function should be established in accordance with nuclear regulations. There are no specific rules on research reactors in most countries. Generally, the rules on nuclear power plants are applied taking into consideration the characteristics of research reactors. Systems and components classified nuclear grade are assumed to function. Single failure criteria are considered. The operator action time allowed in safety analysis should be justified if the operator action is obviously available within the time. The first reactor trip parameter is neglected and then the second trip parameter is credited for design basis event analysis in some research reactors. A loss of off-site power subsequent to design basis events is not usually considered but it is assumed in loss of coolant accidents sometimes. Independent multiple failures are considered as beyond design basis events but dependent failure is considered as sequences of design basis events.

3. Safety Analysis of a Research Reactor

3.1 Design Features of a Typical Research Reactor

Most research reactors have open-pool and open-tank design features operating at low pressure and low temperature. The reactor core is submerged in a deep pool with big water inventory, which plays a role of effective radiation shield and ultimate heat sink in loss of normal cooling capability. The effective radiation shield leads to that the workers can access the pool top area and experimental facilities surrounding the reactor pool wall during power operation. The hot water layer to prevent the movement of radioactive material to the pool surface diminishes radiation level at the pool top area. In recent the hot water layer is usually implemented in open-pool and open-tank reactors.

A siphon break device is provided in pool-connected systems to maintain sufficient water inventory even in loss of coolant accidents. This design feature ensures a long-term cooling capability of the reactor core without any operator actions.

In general the core flow in normal operation is downward for research reactors with a thermal power less than 10 MW whereas it is upward for those over 10 MW. The downward core flow takes advantages for handling of in-core irradiation rigs and fuel assemblies while the upward core flow has disadvantages. This is very important in research reactors for loading and unloading the irradiation rigs during power operation.

Meanwhile, the downward core flow takes many disadvantages in design of normal and emergency core cooling systems. The applicability of a passive emergency core cooling system to reactors with a downward core flow is limited to a lower thermal power than that of reactors with an upward core flow, the normal cooling pump with a larger flywheel is required, and the normal cooling system should be arranged at a lower elevation to meet the net pump suction head. By contrast the upward core flow has advantages in design of normal and emergency core cooling systems. The criterion if passive emergency core cooling is applicable to reactors with a downward core flow is determined from the cooling performance during the flow reversal that the core flow is switched.

3.2 Description of a Reactor used in Safety Analysis

The research reactor referred to this safety analysis is an open-pool and open-tank type and operating at a full power of 10 MW. The core flow is downward in normal operation. The passive emergency core cooling is accomplished by flap valves after the primary flow coasts down to a certain level. The flap valves are contained on the core outlet pipe. The core flow driven by the suction of two primary cooling pumps is cooled by two heat exchangers, discharged to the pool bottom, and sucked into the reactor core through the reactor top.

The reactor has box-type fuel assemblies consisting of 21 fuel plates. The design basis of power peaking factor is 3.0 and the axial power profile is like a chopped cosine function. The feedback coefficients of void, fuel temperature and moderator are negative. The reactivity feedbacks are not taken account in this study. The effective delayed neutron fraction and prompt neutron generation time are around 0.0068 and 1.3E-4 seconds, respectively. The reactor core is modelled as a hot fuel assembly, an average fuel assembly and fuel assemblies bypass taking account the reflector and fuel gaps.

There are four reactivity control rods, which regulate the reactor power in normal operation and shutdown the reactor by the trip signal of reactor protection system (RPS) in emergency. It is modelled that the most reactive rod is stuck at the full-out position and that the remained rods are dropped from the full-out position.

3.3 Insertion of Excess Reactivity

The representative events in the insertion of excess reactivity are a ramp insertion of reactivity due to an inadvertent withdrawal of a reactivity regulation rod and a step insertion of reactivity due to mishandling of an irradiation target. The ramp insertion of reactivity is investigated in both startup operation and full power operation. These events are classified into anticipated operational occurrences. The ejection of a reactivity control rod is not usually concerned in open-pool and open-tank type reactors due to low pressure operation. The cold water injection is also not concerned due to low temperature operation.

Figure 1 shows the relative power in the step insertion of reactivity by 1.8 mk. The reactor is tripped by the power level Hi set-point of RPS. The trip signal of the rate of log power is neglected in this analysis. The power abruptly increases up to the peak due to the delay time of RPS and drop of reactivity control rods. As the reactivity control rods drop into the reactor core by gravity force, the power decreases sharply.

Figure 2 shows the CHFR and fuel temperature (FT) at the hot fuel assembly. The CHFR decreases to the minimum and then increases. By contrast the fuel temperature increases up to the maximum and then decreases in this event. The minimum CHFR is greater than the design limit CHFR of 1.5 and the maximum fuel temperature is less than the design limit temperature of 400℃. Accordingly, the fuel integrity is ensured.

The trends of the power, CHFR and fuel temperature in the ramp insertion of reactivity are similar to those in the step insertion of reactivity.

3.4 Loss of Flow

The representative events in the loss of flow are failure of all primary pumps and shaft seizure of a pump. The failure of all primary pumps may be expected since the electrical powers for the pumps are not separated completely in research reactors. In general the failure of all primary pumps is classified into an anticipated operational occurrence while the shaft seizure of a pump a design basis accident.

Figure 3 shows the relative power in the failure of all primary pumps. The reactor is tripped by the primary flow Low set-point signal of RPS. The primary flow Low set-point is 85% of thermal design flow. The trip signal of the core differential pressure is neglected. The power decreases steeply as the reactivity control rods drop into the core after the reactor trip signal occurs.

Figure 4 shows the flow rates at the reactor core and the flap valve. The core flow decreases gradually as the primary pumps coast down. The flap valve becomes to open around 52 seconds later since the event occurs. The coolant in the reactor pool flows into the core outlet pipe right after the flap valve opens. The flow fluctuates a little and reaches a steady state. The flow at the reactor core is reversed from the downward direction. In the figure the negative flow at the reactor core means the upward direction. This flow reversal is caused by buoyancy force driven at the reactor core. The negative flow at the flap valve means that the coolant flows from the reactor pool to the core outlet pipe.

Figure 5 shows the CHFR and fuel temperature at the hot fuel assembly. The CHFR decreases with the decrease of core flow and increases with the decrease of reactor power by the reactor trip. The CHFR decreases due to the low core flow during the flow reversal. After that the CHFR increases as the core flow is developed by the buoyance force. The fuel temperature increases until the reactor power starts to decrease and decreases after that. The fuel temperature increases due to the low flow during the flow reversal. After that the fuel temperature decreases as the core flow is developed by the buoyance force. The minimum CHFR and the maximum fuel temperature do not exceed the design limits. Consequently, the fuel is not damaged.

In the shaft seizure of a pump the core flow deceases faster than that in the failure of all primary pumps. The fast decrease of the core flow results in a lower CHFR right before the reactor trip as compared with that in the failure of all primary pumps. However, the flow finally reaches a certain level slightly higher than half of the initial core flow rate.

4. Summary

The general approach to safety analysis for research reactors is described and the design features of a typical open-pool and open-tank type reactor are discussed. The representative events expected in research reactors are investigated. The reactor responses and the thermal hydraulic behavior to the events are presented and discussed. From the minimum CHFR and the maximum fuel temperature calculated, it is ensured that the fuel is not damaged in the step insertion of reactivity by 1.8 mk and the failure of all primary pumps for the reactor with a 10 MW thermal power and downward core flow.

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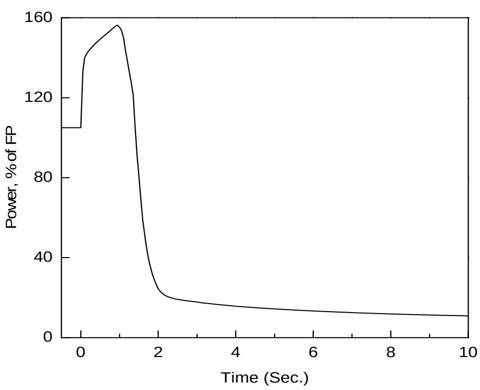

Fig. 1. Power in the ramp insertion of reactivity.

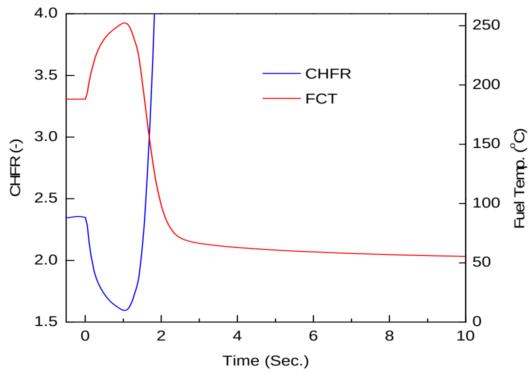

Fig. 2. CHFR and FT in the ramp insertion of reactivity.

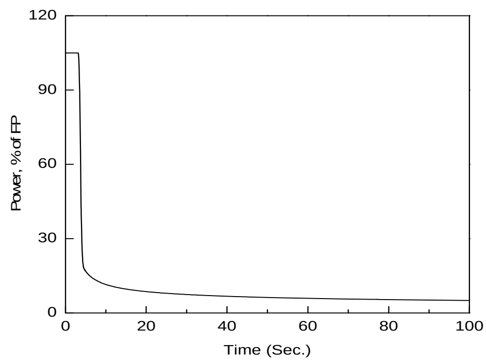

Fig. 3. Power in the failure of all primary pumps.

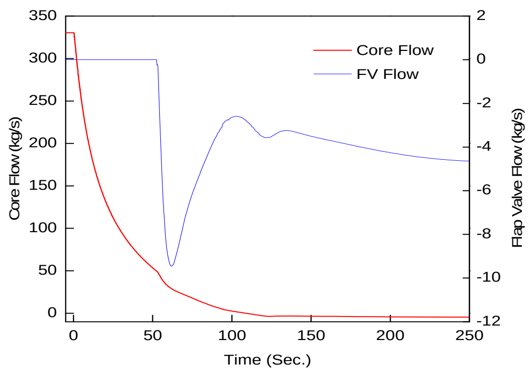

Fig. 4. Flow rates in the failure of all primary pumps.

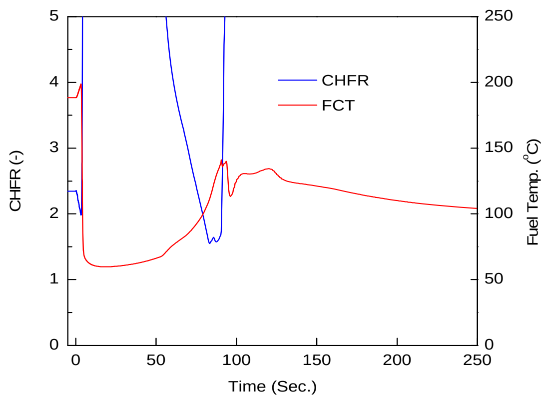

Fig. 5. CHFR and FT in the failure of all primary pumps.