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Framework analysis of fluoride salt-cooled high temperature reactor probabilistic safety assessment*

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Probabilistic safety assessment (PSA) is important in nuclear safety review and analysis. Because the design and physics of the fluoride salt-cooled high temperature reactor (FHR) differ greatly from the pressurized water reactor (PWR), the methods and steps of PSA in FHR should be studied. The high-temperature gas-cooled reactor (HTR-PM) and sodium-cooled fast reactors have built the PSA framework, and the framework to finish the PSA analysis. The FHR is compared with the PWR, HTR-PM and sodium-cooled fast reactors from the physics, design and safety. The PSA framework of FHR is discussed. In the FHR, the fuel and coolant combination provides large thermal margins to fuel damage (hundreds of degrees centigrade). The tristructural-isotropic (TRISO) as the fuel is independent in FHR core and its failure is limited for the core. The core damage in Level 1 PSA is of lower frequency. Levels 1 and 2 PSA are combined in the FHR PSA analysis. The initiating events analysis is the beginning, and the source term analysis and the release types are the target. Finally, Level 3 PSA is done.

Keywords: Fluoride salt-cooled high temperature reactor (FHR), Probabilistic safety assessment (PSA), Framework of PSA

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I. INTRODUCTION

The WASH-1400 report of the 1970s is a milestone of probabilistic safety assessment (PSA) in safety assessment of nuclear power plant. After the Three Mile Island accident, the PSA was pushed to development and application. It has become increasingly important in the safety and operation assessment for nuclear power plants (NPPs), and NPPs in increasing numbers have been benefited from the PSA in their design and operation.

Today, PSA is a necessary prerequisite for licensing of NPPs and is widely used in NPP design, safety analysis and operation. For the NPP types already in common use, either light water reactor (LWR) or heavy water reactor, the PSA technologies and methods are almost perfect. However, for reactors of advanced types, especially the non-LWR type of experiment reactor which is developing by us, in-depth study and discussions are required for the PSA methods.

II. PSA METHODS AND FRAMEWORK IN LWR

PSA of LWR NPPs provides a comprehensively structured approach to identify failure scenarios of NPPs. And the risk to staff and the public caused by NPPs are calculated. PSA describes quantitatively the system risk of an NPP by the event tree and fault tree methods. It finds the weaknesses and provides suggestion for improvements. The scope of data and understanding of the system decide the reliability and uncertainties of the PSA.

PSA is always divided into three levels. Level 1 PSA analyzes the sequence of events which lead to core damage, and the core damage frequency (CDF) is estimated [1]. Level 1 PSA provides insights into strengths and weaknesses of the safety-related systems and procedures in place or envisaged as preventing core damage. Level 2 PSA analyzes the responses which lead to containment failure. These kinds of responses are caused by the results of Level 1 PSA. Level 2 PSA identifies ways in which associated releases of radioactive materials from fuel can result in releases to the environment. It also estimates the frequency, magnitude and other relevant characteristics of the release of radioactive materials to the environment [2]. In Level 3 PSA, public health and other societal consequences are estimated, such as the influence of the radioactive material release to the environment [3]. Figure 1 shows the Levels 1, 2 and 3 PSA relationships and framework in the light-water reactors [4].

Table 1 shows the beginning and analysis target of PSA analysis in different levels, with which each level ensures the correctness and completeness of its PSA analysis.

The initiating event analysis, a basic requirement for risk analysis, stars also the whole PSA analysis. Accuracy and completeness of the initiating events and grouping affect quality and results of the entire analysis. In order to ensure that the analysis can be implemented, the initiating events must be screened and grouped into one group with similar process or the same response [1, 5]. Level 1 PSA has two final states. One is accident mitigation success, described as "OK", i.e. it will not lead to loss of the core integrity. The other leads to core damage (CD) and the loss of core integrity. It will be represented by "CD".

The severe accidents process and the source term release are studied and analyzed in Level 2 PSA. The risk of the severe accidents and uncertainties are quantitative and determined. Level 2 PSA focuses on the CD sequences of the

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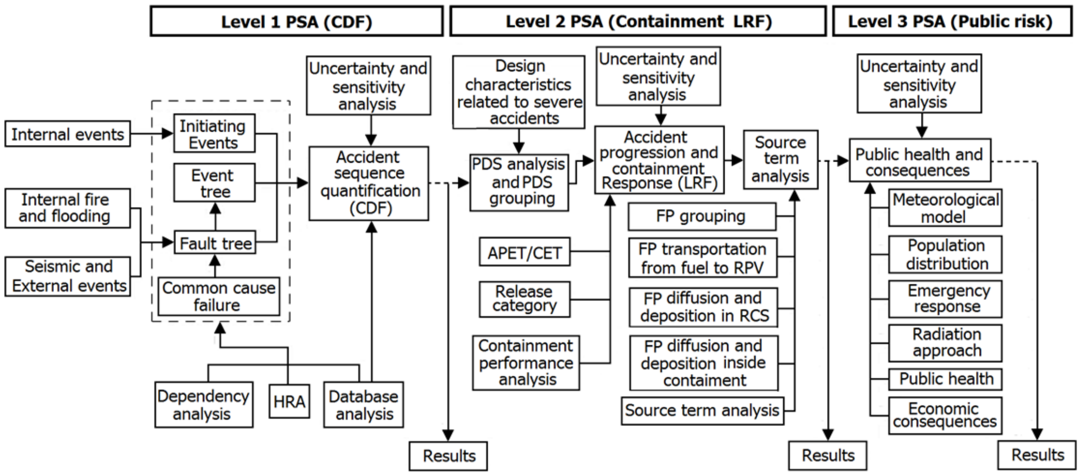

Fig. 1. PSA framework of light water reactor.

TABLE 1. The beginning of different level PSAs and analysis in PWR

PSA Level	Analysis target	The beginning of analysis
Level 1	The sequence of events leading to core damage, the estimated core damage frequency (CDF)	Initiating event analysis and grouping
Level 2	The responses which lead to containment failure from Level 1 PSA results. The path of radionuclide release into the environment, and estimated the LERF/LRF	Plant damage state analysis and grouping
Level 3	Estimate public health and other societal consequences.	Radiation release type analysis and grouping

Level 1. The sequences are grouped by the similar follow-up process. The groups are called the plant damage state (PDS) [6]. Usually, a PDS should have the key information in the CD sequence. And the kinds of information should be reflected and applied in Level 2 PSA analysis. To implement Level 2 PSA, the number of PDS should be controlled. PDS are mainly analyzed by the containment event tree (CET) to get the release category.

Before Level 3 PSA analyses, the severe accident sequences are grouped by similarity of the estimated release radiation. These kinds of group become the release category (RCs). Then the source term analysis for the release categories is carried out. Firstly, the release categories and their properties are determined, and grouped. Then the source term analysis is done for the release categories. Finally, based on the results, the release categories are classified into source term groups. After this, the analysis goes into Level 3 PSA analysis.

III. PSA FRAMEWORK OF HIGH TEMPERATURE GAS-COOLED REACTOR

High temperature gas-cooled reactor (HTR) and sodium-cooled fast reactor are among the six reactor types in the generation IV (Gen-IV) nuclear energy systems in the world [7, 8]. For safety of a Gen-IV reactor, it is important to improve its inherent safety, as its design and concept differ greatly from an LWR [9–11].

A. PSA framework of HTRs in USA, Germany and South Africa

In the probabilistic safety analysis report of the modular high temperature gas-cooled reactor (MHTGR), designed by GA, USA, Levels 1 and 2 PSA are integrated [12]. The release frequency and consequences are as the target of event tree analysis. Frequency of the release categories are analyzed quantitatively by the events sequence [12].

Germany proposed the 200 MW pebble bed modular HTR (HTR-MODULE) design, and its probabilistic risk assessment is done. The initiating event, event tree and release categories are the framework of PSA. The release categories frequency is quantified by the events sequence [13].

The pebble bed modular reactor (PBMR) is developed by South African. The PSA of PBMR used the event tree analysis to model the initiating events, plant response, and the reactor building response. And the development of accidents is modeled by PSA. The initiating events are divided into four categories, with 16 groups, and six release categories are defined as the final states [14].

B. PSA framework of HTR-PM in China

The pebble-bed modular high-temperature gas-cooled reactor (HTR-PM) is developed in China. The safety goal is that the cumulative frequency should be less than 10^{-6} reactor^{-1}year^{-1}, for all beyond design basis accident sequences which lead to off-site (including the site boundary) individual effective dose beyond 50 mSv [15].

Figure 2 shows the framework of HTR-PM's PSA analysis [9, 10]. The initiating event, event tree and the release categories are as the framework of HTR-PM's PSA. It begins with the initiating event analysis, and ends with the release categories analysis. The event tree considers release scenarios, rather than the plant response and operator intervention. Figure 2 shows the framework in the dotted line and the support information outside the dotted line.

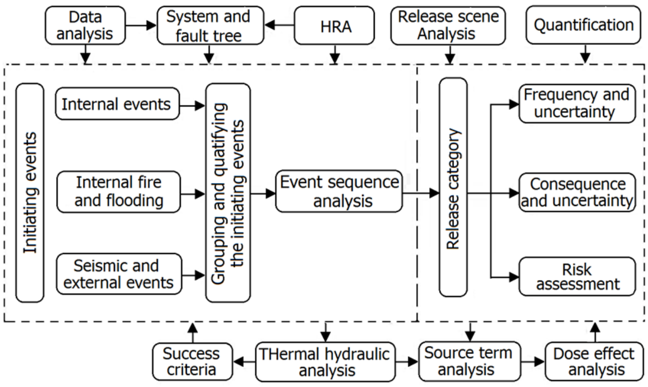

Fig. 2. PSA framework of HTR-PM [9, 10].

Compared with the framework of the pressurized water reactor (PWR, Fig. 1), the boundaries between Levels 1 and 2 PSA is cancelled, without CDF analysis, and the release categories are directly used as the final state of events sequence. This is because of the special characteristics of HTR-PM, which uses spherical fuel element, termed as tristructural-isotropic (TRISO) fuel, and low-pressure containment.

TRISO is UO_2 particles coated with the silicon carbide layer, the buffer layer, and the inner and outer pyrolytic carbon layer. The coated particles ensure that there is a gap between the maximum core temperature and the temperature limit of the outermost layer in a reactor accident. So the core damage frequency is reduced greatly. Even it happens, the damage is inside the coated particles, which are independent of each other, without affecting other fuel particles. So, the radioactive release is limited greatly, hence no need of CDF calculation in PSA [10].

The modular design of HTR-PM is of low power TRISO fuel. In an reactor accident, the decay heat is transferred outside the pressure vessel by radiation and conduction. Even if the coolant is lost, the core damage is of low frequency and the early radiation release is of low probability. The system of negative pressure ventilation and filtration in the containment reduces the radiation emission to the environment. The helium coolant has excellent neutron properties and thermal stability, so the accident phenomena and processes are clearer.

C. PSA framework of sodium cooled fast reactor

From the safety review, the probabilistic safety goals for a fast reactor should be the same as the statement in the HAD 102/17 (The safety evaluation and verification of nuclear power plant). This means the CDF of less than 10^{-5} reactor^{-1}year^{-1}, and the frequency of large radioactive release of less than 10^{-6} reactor^{-1}year^{-1} [16]. For the sodium-cooled fast reactor, the PSA method and framework are similar with PWR, but different greatly from HTR-PM.

Based on PSA analysis in China Experimental Fast Reactor (CEFR), framework of the sodium-cooled fast reactor will be discussed. In the CEFR, Level 1 PSA was done, but Levels 2 and 3 PSA are in the stage of planning. The framework in CEFR is almost the same with PWR, with main differences in detailed system analysis [17–19]. As shown in Fig. 3, for the sodium-cooled fast reactor, the PSA framework is divided into three levels, too. The analysis goals of each level PSA are the same as those of PWR, and analyzed the same as those in Table 1 [17, 18]. The core damage modes, which are closely related to designs of the reactor types, safety protection and accident mitigation system, can be divided into not meltdown (OK) and core damage (CD).

But there are some differences between the sodium-cooled fast reactor and PWR, such as the neutron energy and coolant, good thermal capacity of sodium and active chemical properties, etc. These make the details of PSA in sodium-cooled fast reactor different from PWR. For example, the initiating events, success criteria, event tree analysis, and fault tree analysis are quite different. Also, the sodium-cooled fast reactor is designed as non-pressure, with low pressure in its primary and secondary loops. Therefore, the pressure-bearing containment is replaced by the airtightness confinement. This is the difference in concrete analysis of PSA [17, 18].

IV. PSA FRAMEWORK OF FHR

Molten salt reactor concept originated in the 1950s and is mainly liquid reactor. Solid fuel molten salt coolant reactor was developed in 2001 [20, 21]. It uses coated particles as fuel and fluoride salt as coolant, with passive cooling safety systems. Base on the general design of FHR and the thorium-based molten salt reactor (TMSR-SF1) design in research at Shanghai Institute of Applied Physics, Chinese Academy of Sciences, the PSA framework of FHR is studied [22].

A. Characteristics of FHR

The general characteristics of FHR are as follows. TRISO fuel pellets are used as fuel and the coolant is mixture of molten fluorine salts. It uses passive cooling safety systems are designed and supercritical water circulation system. Its maximum fuel temperature is about 1250 °C under normal operation conditions. In the primary loop, the coolant is 2LiF-BeF_2 molten salt mixture; while in the secondary loop, the coolant is FLiNaK molten salt mixture. The maximum pressure in the reactor vessel is less than 0.5 MPa. The TRISO in the core is randomly arranged and the average temperature of core is around 600 °C. Other systems include the protective

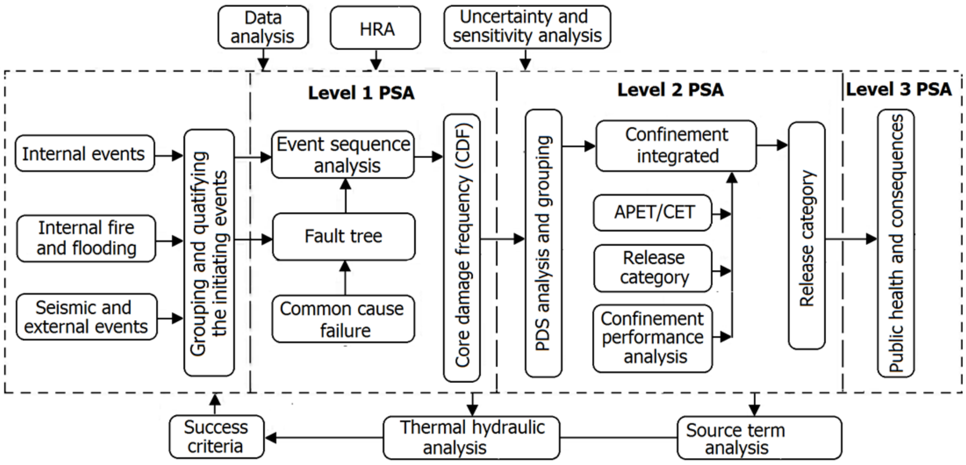

Fig. 3. PSA framework of Sodium-Cooled Fast Reactor.

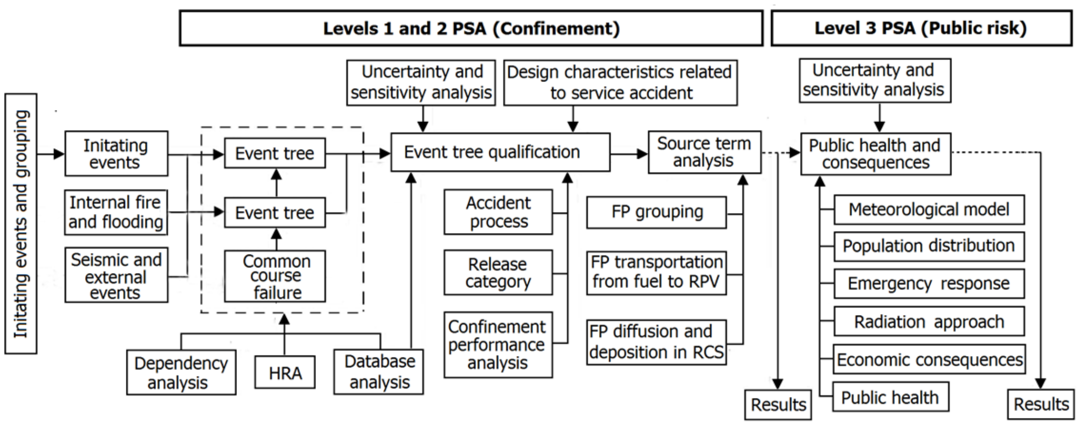

Fig. 4. PSA framework of FHR.

gas system, molten salt cleanup system and monitor system, etc. [23–27].

FHR has many outstanding features. Its particular coolant and fuel will have an impact on its PSA framework. In FHR the operating temperature of molten salt is far lower than the boiling point of 1430 °C for 2LiF-BeF2, which means a large gap as safety margin. In normal pressure, the fluoride salt is molten in high temperature, and the environment does not need to maintain a high pressure state. The TRISO fuel particles fail at about 1600 °C, which is much higher than the boiling point of the coolant temperature and average temperature of core. So the probability of core meltdown due to a temperature rise is greatly reduced [25].

B. PSA framework of FHR

In PWR and sodium-cooled fast reactor, the fuel structural is fuel pellets and cladding composition. The damaged fuel cladding will result in radioactive leaks and fissile material. This will cause core damage. Therefore, Level 1 PSA of the two reactor types is to quantify the core damage and obtain the core damage frequency.

In FHR, the CDF caused by temperature rising up is reduced. The TRISO failure temperature of 1600 °C is much higher than the peak coolant temperature of 700 °C, and higher than the boiling point of coolant ($\sim$ 1400 °C). So there is a large heat margins to extend the artificial non-intervention time and extend the accident processing time. From the physics, the TRISO damage is of such a lower frequency that it can be hardly happen.

The TRISO fuel particles are independent of each other. So damage of one TRISO fuel particle does not affect other fuel particles. Because of its limited volume, the release of radioactive fission products is limited. Therefore, similar to the HTR-PM, the large core damage and large radiation release of FHR is of lower frequency.

As an experimental reactor, the FHR's power density is lower than a power plant reactor. The containment in a power plant reactor can be replaced by the confinement. The PSA analysis method will be revised with this.

Because of the low CDF, the PDS analysis in Level 2 PSA of PWR can be combined into the event tree quantification step in PSA of FHR, and Levels 1 and 2 PSA be combined in

TABLE 2. The Different Level PSAs in FHR

PSA level	Analysis target	The beginning of analysis
Combined Level 1 and Level 2	Analyze the event tree and failure tree to quantification, and obtain source term to analyze the release category.	Initiating event analysis and grouping
Level 3	Estimatiing public health and other societal consequences	Radiation release type analysis and grouping

FHR for the source term analysis.

The PSA framework of FHR is similar with HTR. The boundary of Levels 1 and 2 PSA is not clearly in FHR PSA analysis. They can be combined. The beginning is the initiating events analysis and grouping, followed by the event tree analysis and quantification. The release category and source term are the end of this part PSA analysis. Also the results of source term analysis are input of Level 3 PSA. The framework and detail process is shown in Fig. 4. The analysis target and beginning of different level PSAs in FHR are shown in Table 2.

The analysis and grouping of initiating events must follow the general standard, and this must be evaluated according to the reactor type and design. In the FHR's PSA analysis, the event tree will not only consider a range of usually level 1 PSA, but also should expand to the serious accident and relief. The performance and response of the confinement should be also considered. The corresponding fault tree will also expand its model range which the event tree covered. Because there is similar physical state of coolant between the FHR and sodium cooled fast reactor, the initiating events and fault tree development could be referenced.

V. CONCLUSION

The PSA is more and more important in the nuclear safety analysis and review. The method of PWR is perfection, but for the FHR and HTR the process of PSA is different with the PWR. From comparing the PSA framework of PWR, HTR-PM and sodium-cooled fast reactor, the PSA method and process of FHR have been studied. The different parts of FHR are similar with the HTR-PM or sodium-cooled fast reactor, so the PSA experience of them will be referenced. When we consider the FHR's design characteristics, based on the existing experience, the framework of FHR are established. The FHR's PSA analysis will be developed by the combination of the level 1 and level 2 PSA. The beginning is the initiating events analysis and grouping. The end is the release category and source term analysis. The source term analysis will be the input of level 3 PSA. The PSA framework of FHR will be a guide of its PSA analysis. But some details of the analysis should be discussed. The PSA framework will also guide the safety design and analysis of FHR.

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