# Analysis and Frequency Study on Initiating Events of Break LOCA in High Temperature Gas-Cooled Reactor

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**Abstract:** The analysis of initiating events serves as the preliminary step in performing the probabilistic safety assessment (PSA) and plays a vital role in the process of implementing the PSA of nuclear power plants (NPP). Among the various types of initiating events in nuclear power plants, the loss of coolant accident (LOCA) initiating events specifically refer to accidents where the primary loop system pressure boundary of the reactor ruptures, resulting in the loss of coolant and the leakage of radioactive substances, posing a significant threat to the safety of the reactor.

However, high temperature gas cooled-reactor (HTGR), as a typical fourth-generation advanced reactor type, exhibits unique characteristics in its primary loop design. The conventional analysis methods for LOCA initiating events are not applicable to HTGR, and there is a lack of referenceable experiences and data from similar reactors.

This paper aims to propose an innovative analysis method for the LOCA initiating events based on the design characteristics of HTGR in order to fill the research gap. The research commences by exploring and examining the analysis methods of LOCA initiating events both domestically and internationally. It then proceeds to analyze the primary loop of the HTGR and its associated systems. Based on the design characteristics of the HTGR, a novel classification for LOCA initiating events is defined, categorizing them into isolatable small LOCA, isolatable large LOCA, non-isolatable small LOCA, and non-isolatable large LOCA. Subsequently, by incorporating the system design of the HTGR, a precise calculation based on HTGR NPP is carried out on the frequency of LOCA initiating events, deriving the frequencies of various LOCA initiating events. Finally, these results are compared with the frequencies of initiating events using general data, validating the rationality of the analysis.

This analysis method and its results form a solid foundation for the subsequent conduct of PSA analysis for HTGR. The outcomes of this study offer valuable insights for the safe operation of HTGR and present novel perspectives for future research and enhancements.

**Keywords:** Initiating Event analysis, Break LOCA, Probabilistic Safety Assessment, High Temperature Gas-cooled reactor.

## 1. INTRODUCTION

Probabilistic Safety Assessment (PSA), also known as Probabilistic Risk Assessment (PRA), is a systematic technical approach to engineering safety evaluation developed since the 1970s [1]. It uses system reliability (i.e., fault tree and event tree analysis) and probabilistic risk analysis methods to conduct a comprehensive analysis of the occurrence and development process of various possible accidents in complex systems [2]. This technology has been widely recognized and applied in the nuclear energy field.

The initiating event analysis is the first step in implementing an internal event PSA. The break loss-of-coolant accident (LOCA) is defined as a loss of coolant caused by a rupture in the pressure boundary of the reactor's primary loop system and the leakage of radioactive material, which endangers the reactor's safety [3]. The safety analysis of the break LOCA as an initiating event is an essential part of the final safety analysis document during the reactor design process. This paper analyzes the initiating events of break LOCA of high temperature gas-cooled reactors and performs detailed analysis and calculation of their initiating event frequencies to support the initiating event analysis of the internal event PSA of high temperature gas-cooled reactors. This analytical method and its results will serve as a foundation for future work on PSA analysis.

## 2. Definition of the LOCA initiating event

According to the high temperature gas-cooled reactor's design specifications, in the event of a small diameter pipe rupture (DN≤10mm), the released air containing radioactive helium will be released into the reactor plant through small holes in the reactor compartment walls; in the event of a large diameter pipe rupture (DN>10mm), the released helium will be burst through the insulation wall surrounding the hot gas duct connecting the reactor pressure vessel to the storage tank. Based on the above description, the high temperature gas-cooled reactor pipeline rupture-type initiating events are classified as follows:

(1) Isolatable small LOCAs (DN≤10mm);

(2) Isolatable large LOCAs (DN>10mm);

(3) Non-isolatable small LOCAs (DN≤10mm);

(4) Non-isolatable large LOCAs (DN>10mm).

Such a consideration is reasonable in light of the design characteristics of high-temperature gas-cooled reactors, especially their primary loop design and the design of systems connected to the primary loop, even though it differs significantly from the definition of pressurized water reactors, which classifies primary loop ruptures with the rupture size of less than 10 mm as very small LOCA in PSA analysis.

## 3. Calculation of the frequency of initiating events of break LOCA

### 3.1. Frequency of initiating events of the same type of unit

The MHTGR project in Georgia, the HTR-Module in Germany, and the PBMR project in South Africa are among the current known international high temperature gas-cooled reactor initiatives. The accident analysis report from HTR-Module in Germany, the probabilistic safety analysis report from MHTGR in Georgia, and the safety analysis report from PBMR in South Africa all include thorough evaluations for primary loop break LOCAs.

Primary loop break LOCAs are divided into four groups according to size in the MHTGR probabilistic safety analysis report [4], as illustrated in Table 1. The rupture classification is based on its distinct design features and the data is more precise. For example, one of the reasons for the rupture classification is whether it will result in damage to the cooling system fans during the shutdown. Since the high temperature gas-cooled reactor studied in this paper does not have such a system, the U.S. MHTGR findings are thought to be less applicable.

Table 1. Frequency of MHTGR primary loop rupture

| **Rupture size** | | **Frequency (/reactor year)** |
|---|---|---|
| **cm$^2$** | **DN(mm)** | |
| 1.94E-4~1.29E-2 | 0.16~1.28 | 8.4E-1 |
| 1.29E-2~1.94E-1 | 1.28~4.97 | 1.9E-2 |
| 1.94E-1~6.45 | 4.97~28.7 | 1.4E-2 |
| 6.45~83.85 | 28.7~103 | 2.4E-3 |

As indicated in Table II, the first-loop break LOCAs are split into two groups based on the magnitude in the German HTR-Module incident analysis report [4]. The German data omits information on the frequency of isolatable ruptures and only provides information on the frequency of non-isolated ruptures.

Table 2. Frequency of HTR-Module First-Loop Rupture

| **Rupture size** | | **Frequency (/reactor year)** |
|---|---|---|
| **cm$^2$** | **DN(mm)** | |
| <2 | 16 | 5E-4 |
| 2~33 | 16~64.8 | 1E-5 |

Primary loop ruptures are divided into six groups according to size in the South African PBMR safety study report [4], as illustrated in Table 3. The South African PBMR classifies ruptures on the basis of similarities with the pile types studied in this paper, both of which have a breach type with a 10 mm cut-off. However, the South African PBMR data itself has inapplicable features, such as the fact that isolatable and non-isolatable ruptures occur with similar frequency, and even isolated ruptures occur more frequently than non-isolated ruptures. In contrast, the high-temperature gas-cooled reactor studied in this paper has more valves and longer piping downstream of the primary loop isolation valve than it does upstream.

Table 3 Frequency of PBMR primary loop rupture

| **Rupture type** | **Rupture size** | **Frequency（/reactor year）** | |
|---|---|---|---|
| | | **Isolatable** | **Non-isolatable** |
| SLOCA | <10mm | 2.12E-2 | 4.10E-2 |
| MLOCA | 10mm~65mm | 2.15E-4 | 6.92E-3 |
| LLOCA | >65mm | 8.62E-5 | 4.44E-6 |

When the frequency of initiating events of the same type of units is used as the reference data for the frequency of various type of break LOCA initiating events in domestic HTRs, it may be inferred that PBMR in South Africa is the preferred recommendation. Based on the data above, the frequency of the primary loop initiating events can be taken as follows: isolatable small ruptures occur at a frequency of 2.12E-2/reactor year; isolatable large ruptures occur at a frequency of 2.15E-4/reactor year; non-isolatable small ruptures occur at a frequency of 4.10E-2/reactor year; non-isolatable large ruptures occur at a frequency of 6.92E-3/reactor year.

### 3.2. Specific calculation method for the frequency of initiating events of break LOCA

The calculation of the initiating event of break LOCA at a given frequency, based on the fundamental information about rupture occurring in pipelines, valves, and other components, combined with the design data of pipelines, valves, and other components available in engineering design or construction, LOCA primarily considers the failure of pipelines and their associated valves and other components for detailed assessment.

There are two primary methods for calculating the frequency of initiating events of break LOCAs at home and abroad: using common international data or referring to data from comparable power plants, such as the M310 power plant, which is typically used for common data. The other is to carry out plant-specific calculations based on the pipeline design information of the power plant and the failure rate data of the pipeline, such as AP600, AP1000, Hualong 1, and other newly designed units, all of which use specific calculations.

The second method mentioned above is used for the newly designed power plant because it allows for detailed analysis and calculation based on pipeline design data and pipeline failure rate data from power plants, which is especially important when the design of primary loop has changed significantly and there are no common data or similar power plant data to use as a reference.

According to the research, there are two recommended methods for calculating the frequency of initiating events of break LOCAs. One is the calculation based on statistical pipeline segment data, and the other is the calculation based on statistical pipeline unit length.

a) Based on the number of pipe sections

According to the previous paragraph, the division of large and small ruptures of the high-temperature gas-cooled reactor pipeline is defined by the equivalent diameter of 10 mm.

Based on the research, the EPRI report considers pipe ruptures larger than 1/2 inch (12.7 mm) and recommends that the number of pipe segments are used to calculate the frequencies of break LOCA initiating events [5]. Pipe ruptures larger than 3/8 inch (9.5 mm) are considered in the URD document, and the same number of pipe sections is recommended to calculate the frequency of break LOCA initiating events [6].

The URD report defines a rupture in a way that can include a significant rupture in a pipeline within a high temperature gas-cooled reactor.

As a result, the URD document can be used to calculate the frequency of initiating events of large LOCA that occur in HTR pipelines.

b) Based on pipe unit length

According to the research, the EPRI report considers leakage in the primary loop and defines leakage flow as less than 50gpm, although it is unclear how to calculate leakage frequency. Based on the analysis of the NUREG report, it is clear that the pipeline external leakage is defined and classified as small and large external leakage, with the leakage flow rate of 50gpm being used as the dividing line between the two. It is also advised to calculate the external leakage frequency based on the length of the pipeline [7].

It is clear that the leakage flow definitions shown in the EPRI and NUREG reports are in general agreement with the pipe rupture size definitions used in high temperature gas-cooled reactors. Therefore, the pipeline breach frequency can be calculated using the pipeline external leakage data recommended by the NUREG report.

## 4. Analysis of piping and valves of primary loop connected systems

### 4.1. Definition of the boundary of system analysis

According to the definition of initiating events of break LOCA and the design characteristics of high temperature gas-cooled reactor, the system connected with the primary loop and downstream system are considered in calculating the frequency of various types of break LOCA event.

These system components are not included in the analysis scope of this paper since the fracture of the reactor pressure vessel, steam generator shell, and hot gas duct shell is categorized as a pressure vessel rupture initiating event.

According to the plant design of the high temperature gas-cooled reactor, the systems connected to the first circuit mainly include thermal process measurement systems (CFA), fuel handling system (FC), primary loop pressure relief system (JEG), absorption sphere system (JDE). The pressure relief of primary loop system lines will lead to a line to the helium purification and helium auxiliary system(KBE). In addition, the helium purification and helium assist systems are linked to the gas sampling and analysis system (KUK) and the process radiation monitoring system (CFR). All of the above systems fall under the purview of study in this paper since a rupture in any one of them could result in an initiating event of break LOCA.

### 4.2. Determination of data collection principle

The following guidelines are followed for collecting data for initiating events of break LOCA:

(1) The pipe's section is divided into discontinuities, which include valves, pumps, instrumentation pipes, main pipeline connections, pipe nozzles with the restricted flow, tees, pipe branch points, and other non-energy equipment.

(2) Assuming that a break or leak after the first normally closed valve does not result in a break in the primary loop, only the pipeline data up to the first normally closed valve at the interface with the system connected to the primary loop is counted.

## 5. Calculation of the frequency of initiating events of break LOCA

The frequency of each system under different rupture initiating events can be calculated using the results of information statistics and data on the failure rate of pipelines and valves. To further validate the rationality of the calculated results, it is compared with the common data used in current projects as the frequency of initiating events of break LOCA, as shown in Table 4.

As can be seen from the results that the non-isolatable large and small breaks are 1-2 orders of magnitude smaller relative to the original data. This is due to the fact that non-isolatable breaks occur in the primary loop isolation valve and pressure vessels, as well as steam generators connected to the pipeline; these pipes are generally very short, with fewer pipe sections, so the possibility of a break is low. The isolatable large and small breaks are 1 order of magnitude larger compared to the original data. Because isolatable ruptures are located downstream of the primary loop isolation valve, a relatively large portion of the piping and equipment such as valves (especially the helium purge system and fuel handling system) is at higher risk of rupture.

Table 4 Comparison of the calculated frequency of initiating events of break LOCA for the primary loop-connected system with generic data

| **Rupture type** | **Results（/reactor year）** | **Generic data（/ reactor year）** |
|---|---|---|
| Non-isolatable Large LOCAs | 7.07E-05 | 6.92E-03 |
| Isolatable Large LOCAs | 1.92E-03 | 2.15E-04 |
| Non-isolatable Small LOCAs | 6.96E-03 | 4.10E-02 |
| Isolatable Small LOCAs | 1.30E-01 | 2.12E-02 |

## 6. CONCLUSION

In this paper, we first define initiating events of break LOCA in accordance with high-temperature gas-cooled reactor design features, and then we look into rupture accident analysis of comparable units abroad, from which we choose the data from the South African PBMR as common data. Then, after clarifying the pipeline rupture analysis method and its frequency calculation, the primary loop system and its connected systems in the high temperature gas-cooled reactor are thoroughly analyzed. The frequency of each type of initiating events of break LOCA is determined using the pipeline and valve statistics and the method for calculating the frequency of break LOCA initiating events. The results are compared with those of the general data to verify the rationality of the analysis.

In summary, the initiating events of break LOCA analysis and frequency calculation method used in this paper is appropriate and practical for high temperature gas-cooled reactors, and it establishes the framework for future projects to carry out the initiating events of break LOCA analysis and frequency calculation. This is a good start to gradually adopt plant-specific calculation data for high temperature gas-cooled reactor initiating event frequencies.

## References

[1] Xi S. Probabilistic Risk Assessment of Nuclear Power Plant. Tsinghua University Press, Beijing, 2003.
[2] Feng C R, Li L Z, & Zhao X. Frequency Analysis for Steam Generator Tube Rupture Initiating Event. Research & Design of Nuclear Engineering. 111, 2017.
[3] Ma S C, Yin H Q, He X D, Li J, Meng Y C, Yang X T, & Jiang S Y. Safety Analysis of Large Break Loss of Coolant Accident of PWR Nuclear Power Plant. Atomic Energy Science and Technology. 53, 1036-1043, 2019.
[4] Probabilistic safety analysis report of internal initiating events in power conditions of the high temperature gas-cooled reactor nuclear power plant demonstration project of Huaneng Shandong Shidaowan nuclear power plant, Chapter 4. Tsinghua University, Beijing, 2019.
[5] EPRI. EPRI TR-100380, Pipe Failure in U.S. Commercial Nuclear Power Plant, 1992.
[6] EPRI. Advanced Light Water Reactor Requirements Document, Volume III, Chapter 1, Appendix A, PRA Key Assumptions and Ground Rules, Rev.8, 1999.

[7] Eide S A, et al., Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928 (INL/EXT-06-11119), U.S. Nuclear Regulatory Commission, Washington, DC, 2007.