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Research on Regulatory Safety Performance Indicators for High-temperature Gas-cooled Reactor

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Abstract: Safety performance indicator (SPI) is an important tool for evaluating and monitoring the operational performance of nuclear power plants. U.S. Nuclear Regulatory Committee (NRC) has developed a set of safety performance indicators for reactor oversight, and these indicators have greatly influenced the development of regulatory process in other countries across the world. This paper analyzes the applicability of NRC indicators for high-temperature gas-cooled reactor pebble-bed module (HTR-PM) in China. The study shows that 9 out of 17 indicators need to be modified, as to facilitate their use for HTR-PM. Preliminary monitoring suggestions are given for IE04, MS05, BI01 and BI02. For mitigating systems, this paper analyzes the risk importance of different systems by using probabilistic safety assessment methods. The birnbaum value analysis shows that system performance is highly correlated with only a few component failures. The safety system unavailability analysis shows that train unavailability thresholds for some systems are too strict to achieve practically.

Keywords: Safety performance indicator, High-temperature gas-cooled reactor, Risk-informed regulation, Initiating event.

1. INTRODUCTION

U.S. Nuclear Regulatory Commission (NRC) introduces risk-informed concept into its reactor oversight process [1] (ROP) to focus its resources on matters that are safety significant. Under this concept, NRC has developed safety performance indicator [2] (SPI) for objectively assessing the operating performance of nuclear power plants. By integrating risk-informed information, nuclear safety regulation requirements and the operating experience, the SPI determines acceptable thresholds of operation within considerable safety margins, which helps to give a clear tie between regulatory actions and nuclear power plant (NPP) operating performance, and simultaneously improve regulatory efficiency. For more than two decades, SPI has been proved to be a useful tool, not only for implementing risk-informed and performance-based regulation, but also for helping public communication. The good practice of NRC has provided much reference to many other nuclear safety regulatory agencies across the world for developing their own regulatory indicator strategies.

Although the technical basis of NRC SPI has been fully tested considering to the traditional light water reactors (LWR) [3], its applicability to new reactor designs still need further discussion. As one of generation IV reactors, high temperature reactor (HTR) has many design differences compared with traditional LWRs, and the currently used regulatory indicators need to be refreshed with some special treatments for HTR. This paper conducts a preliminary research on the applicability of NRC SPI for HTR by referring to the design of HTR-PM demonstration power plant in China [4], and gives some suggestions for further development on regulatory indicators for HTR.

2. INTRODUCTION OF SPI AND HTR-PM

2.1. Safety Performance Indicator Introduction

Nuclear industry and regulatory agencies have always been interested in assessing the safety status of NPPs, and many organizations have developed their own performance indicators as well, such as world association of nuclear operators (WANO), institute of nuclear power operations (INPO), international atomic energy agency (IAEA)[5], NRC, etc.. Every set of performance indicators have their unique purposes and distinct characteristics, and NRC have greatly influenced the development of performance indicators for nuclear regulation.

In the middle of 1980s, a systematic assessment of licensee performance (SALP) was developed by NRC for measuring the safety of NPPs. SALP assessed four areas of NPP, including operation, maintenance, engineering and support. NRC staff would assign a performance degree to each of the four areas, and then calculate the overall rating of the NPP performance. However, the SALP rating was too depended on the expertise of NRC staff, and could not give an objective evaluation. In 2000, SALP was replaced by a new risk-informed ROP, which included clear requirements and thresholds for evaluating NPP performance regarding to the operating data collected objectively.

SPI is an important tool for carrying out the aim of risk-informed and performance-based regulation in the framework of ROP. The performance indicators and the corresponding thresholds are determined through careful process, which combines the considerations from mandatory requirements, both deterministic and probabilistic requirements, defense-in-depth philosophy, operational experience, accident analysis and risk insights, etc.. The fulfillment of NRC mission to protect public health and safety is ensured by keeping three strategic performance areas safe, which includes reactor safety, radiation safety and safeguards. The three areas are divided into seven cornerstones to represent the focus of regulation, and performance indicators are used under each cornerstone. Figure 1 shows the framework of indicators of NRC, and more details of these indicators could be found in reference of NEI 99-02[2].

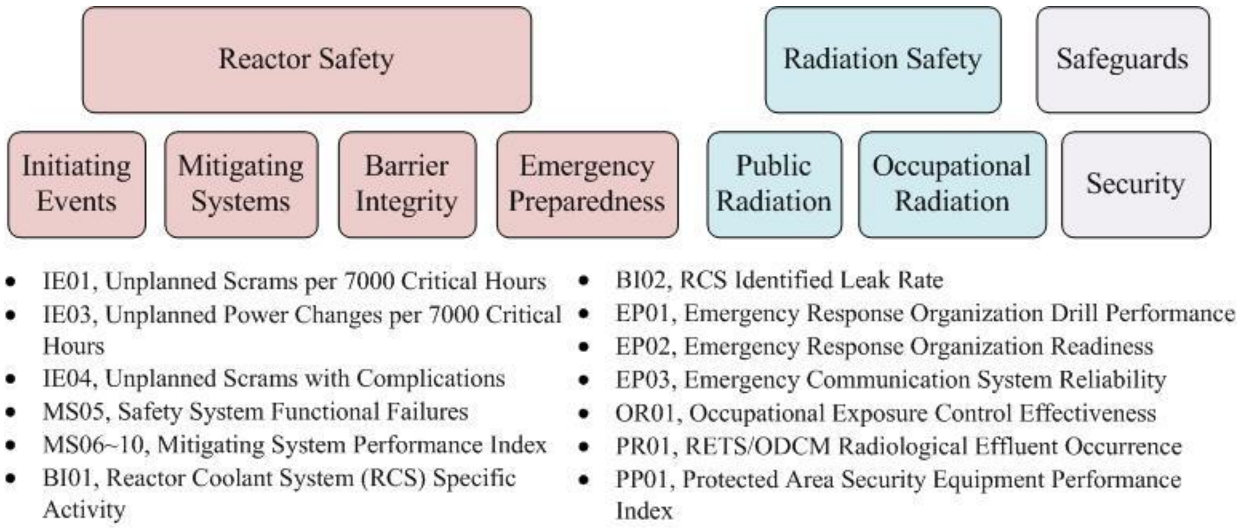

Figure 1. Framework of Safety Performance Indicators

2.2. HTR-PM Introduction

HTR-PM has reached initial full power operation. There are two pebble-bed reactor modules together with one 211 MWe steam turbine. Each reactor has more than 420,000 fuel spheres, and every sphere consists of 12,000 coated fuel particles. Helium is used as reactor coolant to transfer heat from reactor core. Graphite serves as neutron moderator.

HTR-PM is a typical generation IV reactor, and is quite different in the system design and accident prevention process comparing to conventional LWRs, which has brought some challenges to the applicability of commonly used performance indicators. In this paper, the design feature of HTR-PM concerning to the determination of indicators will be discussed, while the other details will not be iterated.

3. PRELIMINARY SPI APPLICABILITY EVALUATION FOR HTR-PM

This section gives a preliminary applicability analysis of NRC SPI for HTR-PM. For each of 17 indicators from 7 cornerstones, their applicability will be discussed.

3.1. Initiating Events Cornerstone

The initiating events cornerstone monitors events that affect the stability of NPP, and if without proper mitigation these events may eventually evolve to accidents. Such events include unplanned scrams and unplanned power changes, which are covered by IE01 and IE03 separately. In some circumstances, things may be more complicated than normal scram, and these events are deemed to be risk-significant and needs special regulatory attention. IE04 covers the scrams with complications, such as requiring additional actions or having critical system unavailable.

In terms of counting the event frequency type indicators, IE01 and IE03 could be applied to HTR-PM similar as traditional LWRs. While, HTR-PM have different scram response procedures, and IE04 should be modified to consider those differences.

3.2. Mitigating Systems Cornerstone

There are two kinds of indicators under mitigating system cornerstone. The first is MS05, which counts the safety system functional failures. The other indicators of MS06 to 10 share the same evaluation method of mitigating system performance index (MSPI).

HTR-PM does not share much similarity in the design of mitigating systems with traditional LWRs, and therefore the indicators under this cornerstone are the most challenging ones need for further discussion.
For MS05, the safety functions and related structures and systems that are considered need to be defined for HTR-PM.

For MS06 to 10, the monitored systems for HTR-PM are not determined, and monitored systems and the system performance evaluation method need to be discussed.

3.3. Barrier Integrity Cornerstone

The physical barriers of NPP are designed to protect public from radioactive material release in terms of accidents. Typically, they are fuel cladding, primary loop boundary, and containment. HTR-PM is still designed with these barriers, however with different realization styles.

BI01 and BI02 need to consider the design feature of HTR-PM, and modify the definition.

3.4. Emergency Preparedness Cornerstone

The emergency preparedness cornerstone monitors and evaluates the capability of licensee to commit proper measures under a radiological emergency. The three indicators under this cornerstone are management related indicators and are commonly used across the industry, and they are applicable to HTR-PM as same as traditional LWRs.

3.5. Public Radiation Cornerstone

The public radiation cornerstone assesses the licensee performance in managing radiological effluent release. The evaluation criterions of liquid and gaseous effluents are derived from nuclear regulatory requirements, which are kept the same for all type of reactor designs. Therefore, the PP01 indicator is also applicable for HTR-PM.

3.6. Occupational Radiation Cornerstone

The OR01 indicator evaluates the occupational exposure control effectiveness of licensee. Similar with PP01, the evaluation criterion follows mandatory requirements, and there is no modification need for HTR-PM.

3.7. Security Cornerstone

The indicator for this cornerstone monitors the availability of security equipment to perform their intended functions. The security management strategy and indicator evaluation method are commonly used across different type of NPPs. Therefore, PP01 is applicable for HTR-PM.

3.8. Summary

This section gives a preliminary analysis of NRC SPIs concerning the design of HTR-PM. Table 1 shows that 9 out of 17 indicators need to be modified so as to better facilitate their use in HTR-PM. For other indicators, they are applicable to HTR-PM as same as traditional LWRs, which are counting number type indicators, or indicators with similar management strategy or follow the same mandatory rules.

Table 1. Preliminary Evaluation Result of SPI for HTR-PM

SPI	Apply	Data Scope	SPI	Apply	Data Scope
IE01	YES	Reactor	EP01	YES	Unit
IE03	YES	Reactor	EP02	YES	Unit
IE04	Modification	Reactor	EP03	YES	Unit
MS05	Modification	Reactor	OR01	YES	Unit
MS06 to 10	Modification	Reactor	PR01	YES	Unit
BI01	Modification	Reactor	PP01	YES	Unit
BI02	Modification	Reactor			

4. DISCUSSION ON INDICATOR MODIFICATIONS FOR HTR-PM

This section will give a more detailed discussion on those indicators that need to be modified for HTR-PM.

4.1. IE04 Discussion

IE04 counts the number of unplanned scrams with complications, and the complication conditions for HTR-PM needs to be clarified.

For a pressurized water reactor (PWR), the complication conditions are defined as: a) two or more control rods fail to fully insert; b) turbine fails to trip; c) engineered safety feature buses lose power; d) safety injection signal is generated; e) main feedwater unavailable or not recoverable using approved plant procedures during scram response; f) the scram response procedure unable to be completed without entering another emergency operating procedure.

For HTR-PM, the complication conditions should be modified to better facilitate its unique accident response process. By referring to the safety functions descried in section 4.2, and the risk importance analysis results of section 4.3, the complication conditions for HTR-PM are preliminarily defined as follows: a) control rods system and absorption sphere shutdown system both failure; b) primary helium blower stop signal is generated; c) the scram response procedure unable to be completed without entering another emergency operating procedure.

More detailed requirements for the above conditions should refer to the reactor operating procedures of HTR-PM, and will be discussed in future.

4.2. MS05 Discussion

MS05 counts the safety system functional failures. For a PWR, the monitored safety functions are reactor shutdown, residual heat removal, radioactive material release control, accident consequence mitigation. While, HTR-PM has different accident response process, and the required safety functions are different too. In the following, safety functions and the corresponding structures and systems of HTR-PM will be discussed.

<u>4.2.1. Reactivity Control</u>

The design of reactor core physics and fuel element ensures HTR-PM with a large negative coefficient, and this could realize an automatic reactor shutdown. It is one of inherent safety features of HTR-PM.
Besides, the reactor is designed with two independent systems to realize the diverse control of reactivity. They are control rods system and absorption sphere shutdown system.

4.2.2. Heat Removal of Primary Loop

During normal operation, primary helium blower provides driving force for helium circulation, bringing reactor heat to steam generator.

Under non-scram scenarios, the heat will be transferred through secondary active heat removal pathway that is composed of steam generator, helium blower, reactor startup and shutdown loop. Under scram scenarios, the heat will be transferred by secondary small flow rate active heat removal pathway that is composed of steam generator, fuel handling system helium compressor, reactor startup and shutdown loop.

When the active heat removal pathways are lost, the residual heat could still be transferred to the outside of reactor vessel by natural mechanisms of heat conduction and heat radiation. The heat radiation and natural convection of air will transfer heat from reactor vessel to the water-cooled wall. The water in water-cooled will form natural convection between wall and air-cooled tower. Finally, the residual heat is transferred through heat exchanger in the air-cooled tower to the ultimate heat sump of atmosphere. The above safety function is realized by way of residual heat removal system.

The deterministic analysis shows that residual heat removal system plays a crucial part in lowering temperature of reactor vessel and concrete wall of primary loop cabin. And its main function is to ensure the safety and integrity of reactor vessel under accident scenarios.

Vessel support structure cooling system is also a passive system, and its function is similar with residual heat removal system.

4.2.3. Pressure and Integrity Control of Primary Loop

The safety function of pressure and integrity control of primary loop is realized through primary loop isolation system, primary loop pressure relief system.

Primary loop isolation system consists several pipes that are required to be isolated, which include fuel sphere loading and discharge pipes of fuel handling system, and helium purification system inlet and outlet pipes.

The primary loop pressure relief system prevents the pressure of primary loop from exceeding design limit. There are safety relief valves that perform pressure relief function at different pressure setting values.

4.2.4. Mitigate the Consequences of an accident

Primary loop cabin is a barrier of HTR-PM used for preventing the release of radioactive materials from reactor core and RCS to surroundings. There is a negative pressure ventilation system in primary loop cabin, and it keeps a negative pressure condition for the cabin.

When a small leak accident occurs, the normal negative pressure ventilation system will be transferred to emergency negative pressure ventilation system, and the iodine remover will remove radioactivity from air.
When a severe loss of pressure accident occurs, the negative pressure ventilation systems will be isolated from the cabin. When the pressure of primary loop cabin reaches to a certain limit, the rupture disc device will function to release air to the atmosphere. When the cabin pressure decreases to atmosphere pressure, the operator will close the normally opened valve on the pressure relief pipe, and the cabin will be restored to isolation state. The emergency negative pressure ventilation system will then come into function.

4.2.5. Confinement of Radioactive Material

When a steam generator water ingress accident occurs, there are two ways in limiting water ingression quantity to primary loop.

The first is secondary loop isolation, which consists the main feedwater pipe isolation and main steam-pipe isolation.

The second is by using steam generator accident discharge system, through which the stored water and steam of steam generator and related feedwater pipe and steam-pipe will be discharged to a particular tank.

4.3. MS06 to 10 Modification Discussions

In the reference of NEI 99-02[2], MSPI is used for monitoring important safety systems of LWRs. While for HTR-PM, there are two questions need to be discussed, a) which systems are needed to be monitored, b) how to evaluate the system performance. In the following parts, a probabilistic risk analysis is conducted for highlighting the systems with risk importance in HTR-PM. And also, a preliminary analysis on system performance evaluation methodology concerning MSPI and safety system unavailability (SSU) is discussed.

4.3.1 System Selection Discussion

For a traditional PWR, MSPI monitors the performance of important mitigating systems, such as emergency AC power system, high pressure safety injection system, auxiliary feedwater system, residual heat removal system, and cooling water support system. And typically, these systems are risk significant systems in terms of preventing core damage of a PWR. While, HTR-PM does not share the same design feature with PWR, and it has totally different accident response considerations.

As is mentioned in reference [6], HTR-PM uses a risk metric of LARGE, which represents “cumulative frequency of all the beyond design basis accident sequences that may cause the off-site individual effective dose at the site boundary exceeding 50mSv”. The probabilistic safety goal of LARGE is set to be less than 1.0E-06/RY.

Table 2. System Risk Importance Analysis of HTR-PM

System Risk Importance	**Risk Metric (ΔLARGE, /RY)**	**Systems**
High	≥1E-5	1)Helium Circulator System 2)Main Feedwater Circulation System 3)Primary Loop Pressure Relief System 4)Residual Heat Removal System 5)Vessel Support Structure Cooling System
Intermediate	1E-6~1E-5	1)6kV Power Bus
Low	1E-7~1E-6	1)Control Rod System 2)380V Safety Power Bus 3)Steam Generator Accident Discharge System
Very Low	＜1E-7	1)Work Buses of Electric Building, Reactor Building, Spent Fuel Building, Auxiliary Building 2)System of Active Residual Heat Removal after Shutdown 3)Balance-of-plant work bus 4)Neutron Absorption Spheres System 5)Helium Purification and Helium Auxiliary System 6)Main Steam Isolation System 7)Primary Loop Ventilation System 8)Primary Loop Isolation System 9)Service Water System 10)Component Cooling System 11)380V Emergency Buses I and II

In terms of finding the systems with risk significance, a thoroughly risk analysis for HTR-PM is committed with internal events at-power PSA model, where internal flooding, internal fire and external events are not included in the model. ΔLARGE is chosen to be the risk metric, and is calculated for each case with one system total failure. The results are shown in Table 2, and 6 out of 20 systems are considered to be with high or intermediate risk importance, which represents that a total failure of a specific system will cause a risk increase greater than the safety goal of HTR-PM. And these systems should be included into the preliminary system list that needs further attention and discussion.

4.3.2. MSPI Evaluation

In 2005, NRC published NUREG-1816[3] to confirm the technical readiness for MSPI. And later, MSPI replaced the use of SSU in reactor oversight process of NRC. As is stated in reference [2], there are three important parts in the scheme of MSPI, and they are Unavailability Index (UAI) and Unreliability Index (URI) and Performance Limit Exceeded (PLE). MSPI is sum of UAI and URI. PLE serves as a supplement for determining if the performance of a specific system is abnormal, when the practical component failure number exceeds expected number basis.

$$\mathrm{MSPI} = \mathrm{UAI} + \mathrm{URI} \tag{1}$$

$$UAI_t = CDF_p \left[\frac{FV_{UAp}}{UA_p}\right]_{MAX} (\Delta UA_t) \tag{2}$$

$$URI = \sum_{j=1}^{m} \left[CDF_p \left[\frac{FV_{URc}}{UR_{pc}}\right]_{cj} (\Delta UR_{cj})\right] \tag{3}$$

CDFp is the plant-specific Core Damage Frequency, FV is the Fussell-Vesely value for train/segment-specific unavailability or component failure mode unreliability, △UAt is the increase value of train unavailability, △URcj is the increase value of component failure mode unreliability. Generally, the quantity CDF*FV/U is called Birnbaum factor and could reflect the degree of risk impact, in which U represents UA or UR. In order to have a deeper understanding of MSPI, the above equations could be deformed as following:

$$MSPI = \sum_{j=1}^{m} \left[CDF_p \left[\frac{FV_j}{P_{0j}}\right] (\Delta P_j)\right] \tag{4}$$

By referring to a commonly used equation in PSA analysis of FV*(△P)/P= (△CDF)/CDF. Now, the true meaning of MSPI can finally be seen, that MSPI represents the overall △CDF under a condition when the performance of components and trains within a particular system changes.

For HTR-PM, a preliminary analysis of MSPI could be done with replacing CDF with LARGE. Considering the lack of operational data for HTR-PM, the analysis of this paper will be focused on the birnbaum value discussion instead of the final MSPI results. And the PLE is not an emphasis for this phase of study for the same reason.

The component failure mode birnbaum of two systems are calculated to show some characteristics of performing MSPI for HTR-PM. Table 3 shows the results of residual heat removal system, and Table 4 shows the results of primary pressure relief system, and.

Table 3. Birnbaum Results of Residual Heat Removal System

Failure Description	**Birnbaum**	**Failure Description**	**Birnbaum**
Physical Process Failure	1.00E-04	Manual Valve Fails to Remain Open (i.e 01AA06/02AA06/03AA06)	1.33E-08
Air Cooler Fails to Operation (i.e #1/2/3)	1.34E-08	Tank Operation Failure (i.e 01BB01/ 02BB01/ 03BB01)	1.32E-08
Air Inlet Door Fails to Remain Open (i.e #1/2/3)	1.33E-08	Pipe Operation Failure (i.e 01BR01/ 02BR01/03BR01/01BR03/ 02BR03 /03BR03/01BR05/02BR05/03BR05)	1.25E-08

Table 4. Birnbaum Results of Primary Loop Pressure Relief System

Failure Description	**Birnbaum**	**Failure Description**	**Birnbaum**
Manual Valve 40AA11 Fails to Open	2.59E-05	Safety Valve 10AA41 Pressure Detector Failure	<1.0E-13
Manual Valve 40AA11 Fails to Remain Open	2.59E-05	Safety Valve 10AA41 Fails to Sit Back	<1.0E-13
Isolation Valve 20AA11 Fails to Open	2.71E-07	Safety Valve 20AA41 Pressure Detector Failure	<1.0E-13
Isolation Valve 20AA11 Fails to Remain Open	2.70E-07	Safety Valve 20AA41 Fails to Close	<1.0E-13
Isolation Valve 10AA11 Fails to Remain Open	6.36E-08	Isolation Valve 10AA11 Fails to Close	<1.0E-13
Safety Valve 10AA41 Fails to Open	6.45E-08	Isolation Valve 10AA11 Fails to Remain Closed	<1.0E-13
Manual Valve 20AA17 Fails to Open	6.36E-09	Isolation Valve 10AA11 Circuit Breaker fails to Close	<1.0E-13
Manual Valve 20AA17 Fails to Remain Open	5.90E-09	Isolation Valve 20AA11 Fails to Close	<1.0E-13
Manual Valve 10AA17 Fails to Open	6.36E-09	Isolation Valve 20AA11 Circuit Breaker fails to Close	<1.0E-13
Manual Valve 10AA17 Fails to Remain Open	5.90E-09	-	-

There are components in primary loop pressure relief system are required to change state for accident mitigation, which will apply to the rules of MSPI. While, some components, such as safety valves may be a little vague concerning MSPI component selection requirements, for their function principle is similar with that of check valve, and which is excluded from MSPI calculation. And the passive residual heat removal system may be totally against the rules of MSPI. However, in this paper, the birnbaum is calculated for all the component failures of above two systems without screening, which is to show the original results for further discussion.

In Table 4, Manual Valve 40AA11 fails to open and fails to remain open are the two top failure modes. This valve is located on the manual pressure relief part of the system, and it is used for emergency pressure relief during some extreme conditions, where usually both low pressure relief safety valve and high pressure relief safety valve fail to function. The results reflect their risk significance. While for other component failures, the sum of birnbaum is less than 1.0E-06, which means that even with an unreliability change $\triangle$P of 1.0, the risk increase is still within the limit of safety goal. In conclusion, the performance of the system is highly correlated with only a few component failures.

In Table 3, the empirical mode of physical process failure of residual heat removal system takes an absolutely leading part, and if the failure probability change is more than 1.0E-02, then the risk increase will directly pass the safety goal limit of HTR-PM. And this means that the physical process failure is a highly sensitive mode, while other component failures are not.

In summary, HTR-PM has some passively operated components and systems, which may challenge the MSPI rules that are used in traditional LWRs. The preliminary birnbaum results show that the system performance is highly correlated with only a few component failures, while most of component failures do not have distinct impact on the risk increase even with extremely unacceptable failure probabilities.

4.3.3. SSU Evaluation

In 2005, NRC published NUREG-1816[3] to confirm the technical readiness for MSPI. And later, MSPI replaced the use of SSU in reactor oversight process of NRC.

As is mentioned in Section 4.3.2, SSU is replaced by MSPI in the regulatory process of NRC. Of all the shortcomings of SSU, the main reason is that SSU uses generic performance thresholds and cannot reflect the risk variation between different NPPs. However, as a method that is still commonly used in the nuclear industry, it is still meaningful to have a preliminary analysis on SSU for HTR-PM.

This paper conducts some modifications on the PSA model of HTR-PM. The train unavailability is considered for risk analysis, and a basic event is added to each train of the analyzed system. The train unavailability parameters could be determined by comparing with different △LARGE thresholds. The results are shown in Table 5.

Table 5. Averaged Train Unavailability Analysis Results

System	Basic Failure probability	Risk Metric (ΔLARGE, /RY)		
		1E-07	1E-06	1E-05
Helium Circulator System (Helium blower trip function)	6E-10	≈5E-07	≈5E-06	≈5E-05
Main Feedwater Circulation System (Main feedwater isolation function)	2E-08	≈5E-06	≈5E-05	≈5E-04
Primary Loop Pressure Relief System (Manual pressure relief function)	4E-04	≈2E-03	≈2E-02	≈2E-01
Residual Heat Removal System	5E-06	≈5E-04	≈5E-03	≈5E-02
Vessel Support Structure Cooling System	4E-06	≈2E-03	≈2E-02	≈2E-01
6kV Power Bus	2E-05	≈8E-02	≈8E-01	NA
Primary Loop Pressure Relief System (High pressure relief function)	6E-03	≈2E-01	NA	NA

As to mention, there are some systems do burden several different safety functions with different parts. For helium circulator system, the more risk significance function of helium blower to trip is considered in risk quantification. Main feedwater isolation function of main feedwater circulation system is selected. For primary loop pressure relief system, it is designed with high pressure relief branch and low pressure relief branch, and different branch will operate at different pressure limits. And there is also a manual pressure relief branch used in some extreme circumstances. These branches cannot be seen as separate and redundant trains. A preliminary analysis shows that the manual branch failure will increase LARGE more than 1.0E-05, and the high pressure relief branch failure will result in △LARGE about 5.0E-07, and the low pressure relief branch failure will result in △LARGE about 1.0E-07. In table 5, the averaged train unavailability of manual branch and high pressure branch are also calculated in terms of risk metrics.

By referring to a same set of risk metric thresholds, the results of Table 5 show that systems with high risk importance and low basic system failure probability are strictly required to have very low unavailability thresholds. The results also show that, if △LARGE of 1.0E-07 is used as an abnormal standard for HTR-PM, which similar with △CDF of 1.0E-06 is used in NRC ROP, the train unavailability thresholds for most systems in Table 5 are still too strict to achieve practically.

4.4. BI01 Modification Discussions

BI01 monitors the integrity of fuel cladding, which is one of the three barriers for radioactive materials confinement. It measures the radioactivity in the RCS. LWRs monitor the monthly RCS activity in becquerel per gram (Bq/g) dose equivalent Iodine-131 by following the requirements of technical specifications. The result of BI01 is expressed as a percentage of monthly maximum value compared to the technical specification limit.

HTR-PM does not directly collect Iodine-131 data. Most of the radioactive substances are contained in the four-layer ceramic-coated fuel particles, and the released radioactive substances will lead to the increase of γ radioactivity in the helium coolant. Therefore, the γ radioactivity results could reflect the damage rate of fuel particles.

Therefore, BI01 of HTR-PM is suggested to monitor total γ radioactivity in the primary coolant instead of Iodine-131. The BI01 result is also expressed as a percentage to the limit of technical specification.

4.5. BI02 Modification Discussions

BI02 is intended to monitor the integrity of RCS pressure boundary. For HTR-PM, the leakage of helium includes all the leakages in the primary loop boundary, the related fuel handling systems and helium auxiliary systems. The leakage calculation methods should follow the requirements of technical specifications. The maximum monthly value of helium leakage rate should be reported, and the BI02 result could be expressed as a percentage of the technical specification limit.

4.6. Summary

This chapter gives an analysis of NRC SPIs that are needed to be modified for HTR-PM. Preliminary modification suggestions concerning IE04, MS05, BI01 and BI02 are given. The thresholds of these indicators should be kept the same with traditional LWRs, and with the accumulation of operational experience, these thresholds should be evaluated timely. For MS06 to 10, the issues of system selection, MSPI and SSU applicability are analyzed. However, further analysis and operational experience readiness are still needed for determining a proper method to evaluate mitigating system performance, and the thresholds of these indicators cannot be decided for now.

5. CONCLUSION

This paper analyses the applicability of NRC regulatory performance indicators for HTR-PM. The study shows that the most challenging part lies in the mitigating system performance monitoring. The currently used MSPI and SSU method cannot be directly used for regulation, and further analysis and operational experience readiness are still needed for determining a proper method to evaluate mitigating system performance. As for other indicators that need modification for HTR-PM, this paper has given preliminary regulation suggestions.

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