The Safety Design Guideline Development for Generation-IV SFR Systems

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Abstract. The Generation-IV International Forum (GIF) Safety Design Criteria Task Force (SDC TF) has been developing a set of safety design guidelines (SDG) to support practical application of SDC since the completion of the "SDC Phase I Report" that clarifies safety design requirements for Generation IV Sodium-cooled Fast Reactor (Gen-IV SFR) systems. The main objective of the SDG development is to assist SFR developers and vendors to utilize the SDC in their design process for improving the safety in specific topical areas including the use of inherent/passive safety features and the design measures for prevention and mitigation of severe accidents. The first report on "Safety Approach SDGs" aims to provide guidance on safety approaches covering specific safety issues on fast reactor core reactivity and on loss of heat removal. The second report on "SDGs on key Structures, Systems and Components (SSCs)" focuses on the functional requirements for SSCs important to safety; reactor core system, reactor coolant system, and containment system.

Key Words: Safety Design Guideline, Safety Design Criteria, Generation IV SFR, severe accident.

1. Introduction

The GIF SDC TF was formed to develop SDC for the Gen-IV SFR systems in 2011. After the completion of SDC Phase-I Report that clarifies safety design requirements, the report was disseminated for external review and has been updated based on the comments/suggestions received from the international review [1]. The international feedback on the SDC Phase-I Report also provided an incentive and motivation for further technical interpretation and clarification of the SDC. Based on these needs, the Phase-II activity of the SDC TF was started for the development of SDG in 2013. The SDG is conceived as a series of detailed guideline documents at the level lower than the SDC in a hierarchy of the safety standards (see *FIG.1.*). The main objective of the SDG development is to assist SFR developers and vendors to utilize the SDC in their design process for improving the safety in specific topical areas including the use of inherent/passive safety features and the design measures for prevention and mitigation of severe accidents. The two expected outputs of the SDG development effort are the reports on "Guidelines on Safety Approach and Design Conditions of Generation IV SFR systems" (so-called "Safety Approach SDG") , and on "Safety Design Guidelines on the Key Structures, Systems and Components" (so-called "SSC-SDG").

The first report on "Safety Approach SDG" aims to provide guidance on safety approaches covering specific safety issues on fast reactor core reactivity and on loss of heat removal. It covers specific safety issues on "prevention and mitigation of severe accidents (issues related to fast reactor core reactivity)" and "situations to be practically eliminated (issues related to loss of heat removal)." Based on the progress for facilitating the common understandings on specific safety measures through the discussions not only by the TF but also with the stakeholders related to SFR development in a workshop and an international symposium, "Safety Approach SDG" was completed by the TF and was delivered to IAEA (International

Atomic Energy Agency) and GSAR (OECD/NEA CNRA/CSNI joint ad-hoc Group on the Safety of Advanced Reactors) for review. The second report on "SSC-SDG" focuses on the functional requirements for SSCs important to safety such as reactor core, reactor coolant system and containment system, and establishes design parameters and constraints for postulated accidents. The second SDG report is currently under development by SDC TF.

This paper elaborates on the contents of "Safety Approach SDG".

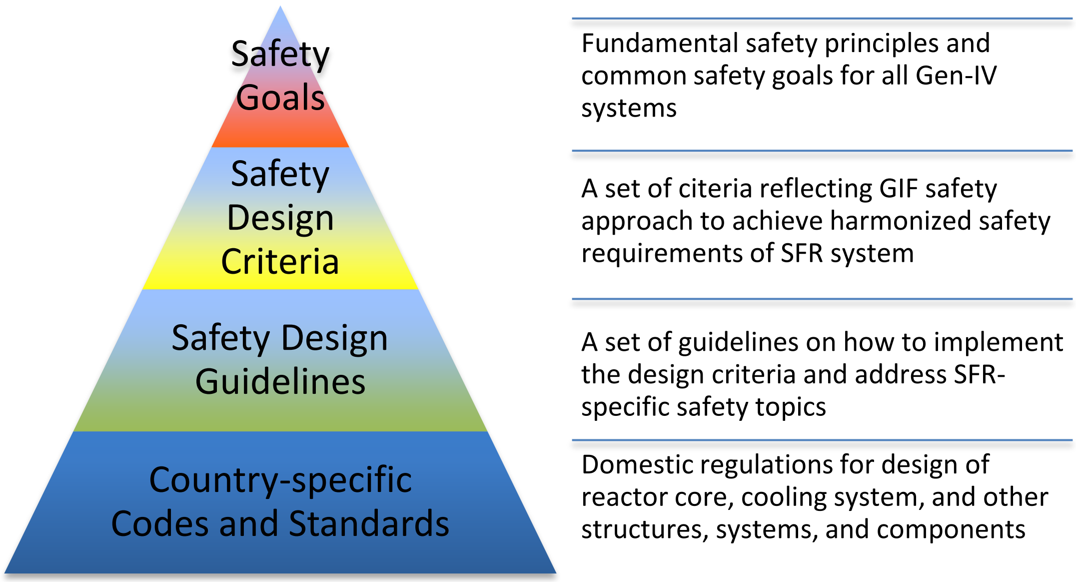


FIG. 1. Hierarchy of Safety Standards

2. Safety Approach SDG

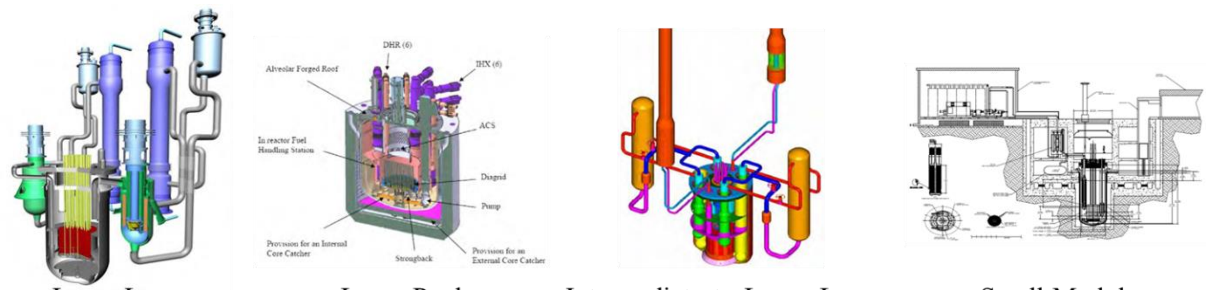
The primary focus of "Safety Approach SDG" is to provide SDGs for Gen-IV SFR systems as a technical supplement to the SDC report. The enhancement of Gen-IV SFR safety is mainly focused on improving each level of Defence-in-Depth (DiD), including the 4th level, with particular attention on the robustness of safety demonstrations (practical elimination demonstrations, independence of lines of defence). Like Generation-III Light Water Reactors (LWRs), Gen-IV SFR safety is primarily based on the use of multiple redundant safety systems to lower the probability of accidents and to limit the consequences of anticipated operational occurrences (AOOs) and design basis accidents (DBAs). These safety systems include independent and diverse scram systems, multiple coolant pumps and heat transport loops, decay heat removal systems (DHRSs), and multiple barriers against release of radioactive materials. In addition to these systems, passive/inherent features for cooling and shutdown/power reduction may also play a significant role in the safety performance of Gen-IV SFRs by improving the diversity of safety systems.

2.1. Gen-IV SFR Systems

An SFR uses liquid sodium as the reactor coolant, allowing high fuel power density with low coolant volume fraction. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. The current plant size options range from small modular reactors, 50 to 300 MWe, to larger plants up to 1,500 MWe. The coolant outlet temperature of primary system is 500 - 550°C, which allows the use of materials developed and proven in prior SFR programmes. An SFR closed fuel cycle enables generation of fissile fuel and facilitates the management of minor actinides. Important safety features of SFR systems include a large margin to coolant boiling during normal operating conditions, a primary system that operates at near atmospheric pressure, and an intermediate coolant loop between the radioactive sodium in the primary circuit and the power conversion system. Water/steam, nitrogen and carbon-dioxide are considered as working fluids for the power conversion system to achieve high efficiency, safety and reliability.

An SFR can be arranged in a pool layout or in a compact loop layout. The following three options are considered in the GIF SFR System Research Plan, with examples provided in FIG. 2[2]:

- A large size (600 to 1,500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based on advanced aqueous processing at a central location serving a number of reactors.
- An intermediate-to-large size (300 to 1,500 MWe) pool-type reactor with oxide or metal fuel.
- A small size (50 to 150 MWe) modular pool-type reactor with metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.



Large Loop | Large Pool | Intermediate-to-Large Loop | Small Modular

FIG. 2 Gen-IV SFR Concepts

2.2. Design Basis and Residual Risk

In the design of a nuclear power plant, compliance with the fundamental safety objectives[1] [3] should be demonstrated for all accident initiators. Potential initiating events or accident sequences are identified and grouped into a limited number of plant states, primarily on the basis of their frequency of occurrence. Postulated initiating events or accident sequences in the plant states [4], described below, provide the design basis[2] for the safety design of nuclear power plants, while the residual risk is not included in the plant states. The situations to be practically eliminated are considered to be part of the residual risk. An illustration of design basis and residual risk is given in FIG 3.

[1] The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits. (Criterion 5 in SDC)

[2] The range of conditions and events taken explicitly into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.

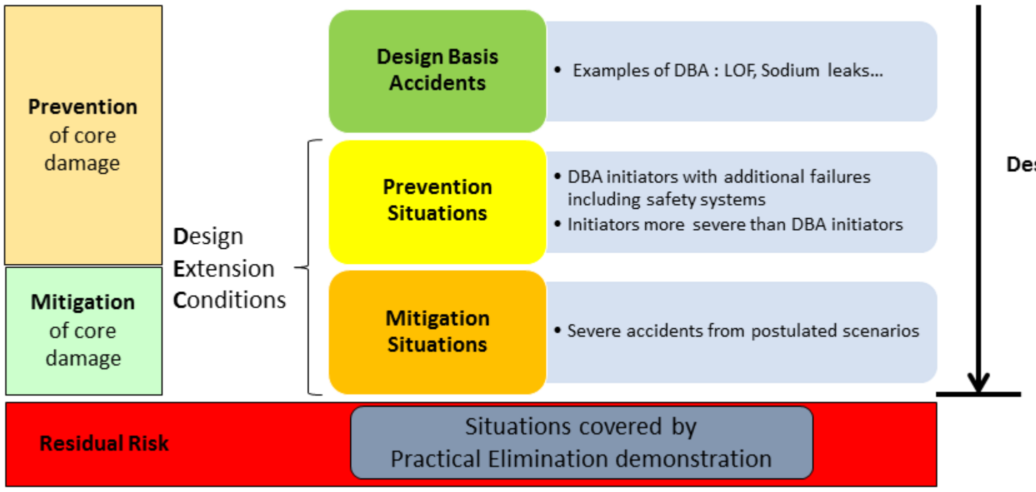


FIG.3. Illustration of design basis and residual risk

2.3. Approach to Normal Operation, AOOs and DBAs

Normal operation – The plant is operating as intended, with all plant parameters (temperature, pressure, etc.) within the design ranges for normal operation, and is considered for the development of design measures for DiD Level 1 [3].

In the new SFR designs, strong emphasis should be given to the prevention, detection and control of accident sequences. An SFR must be designed to allow for stable normal operation, which requires being able to control reactor temperatures, within a small range set by the designers, possibly also by varying the reactor coolant flow using the coolant pumps, and by keeping the reactor power in balance with the demand for power from the electric grid. Reactor power is regulated using control rods, which are moved in response to the changing demand for power production.

AOO – AOOs include events, which disrupt plant conditions from their normal state and are expected to occur during the lifetime of a plant, and which are typically caused by a failure or an inadvertent operation of a single SSC, accommodated by the safety systems. Since such events may occur during the lifetime of a plant, the frequency of occurrence of AOOs is as low as approximately 1 x 10^{-2} per reactor year. Plant conditions, expected as a result of an AOO, are considered for the development of design measures for DiD Level 2 [3]. Any significant consequences of an AOO should be prevented (no clad or fuel damage and no significant release of radioactive materials).

DBA – DBAs include events, which are typically caused by the failure of a single SSC with consequences of greater severity than those considered for AOOs and which are not expected to occur during the lifetime of a plant. The frequency of occurrence of DBAs is less than 1 x 10^{-2} per reactor year, and as low as approximately 1 x 10^{-5} per reactor year or less, consistent with global probabilistic objectives assigned to core damage frequencies. Plant conditions, expected as a result of a DBA, are considered for the development of safety systems for DiD Level 3 [3]. Reactor and plant systems should respond to prevent any significant core damage or radioactive release exceeding acceptable limits, although a limited number of fuel pin failures may occur.

AOOs and DBAs are usually managed by using safety systems to shut down the reactor and to remove decay heat. These rapidly responding systems can be actuated by using a signal from the plant (i.e. the plant protection system) that detects an off-normal condition of sufficient magnitude (set by the plant designer). The goal, in case of AOOs and DBAs, is to shut down the reactor and sustain decay heat removal to keep the reactor core and system temperatures within applicable design limits. Some AOOs might be managed by plant control system without reactor shutdown.

2.4. Approach to Design Extension Conditions

Design Extension Condition (DEC) – Accident conditions that are of lower probability than design basis accidents and involve the failure of more than one SSC important to safety or part of a safety system. Design extension conditions should include plant conditions that result in severe accident conditions and are considered for design measures to prevent or mitigate core damage.

Design provisions for DECs are assigned within the fourth level of the DiD [3]. This level includes measures for “prevention of core damage” in “prevention situations” and for “mitigation of core damage” in “mitigation situations”.

The goal of design measures in “prevention of core damage” of DiD Level 4 is to provide lines of defence to prevent conditions leading to significant core damage. Design features for “prevention of core damage” in DECs deal with accident sequences that are typically caused by failure of one or more systems related to safety, such as the reactor coolant pumps, followed by failure of other safety systems needed to prevent excessive power and/or temperatures resulting from the off normal conditions of the plant. “Prevention situations” of DECs also include postulated initiating events, more severe than those in DBAs.

Design measures in “mitigation situations” provide design features for mitigation of consequences of postulated accidents where significant core damage occurs, with the objective of maintaining the containment function to limit radioactive release.

As described in the following section, design provisions to achieve In-Vessel Retention (IVR) is crucial for addressing “mitigation situations”, since the reactor vessel (RV) can, in this case, serve as the boundary for retention and cooling of core material, limiting any threat to the containment.

2.5. Exploiting SFR Characteristics to Enhance Safety

Approaches to prevent and/or mitigate severe accidents for Gen-IV SFR should be taken into account SFR characteristics to enhance safety.

2.5.1. Inherent and Passive Safety

Inherent reactivity feedback effects are obtained by using intrinsic SFR features to reduce power as the core temperature rises in accident conditions. The large temperature margin to sodium boiling (e.g. from 500-550°C in normal operation to about 900°C until boiling) of the reactor coolant provides sufficient room to use reactivity feedback due to thermal expansion of core components, as well as neutron leakage effects due to the change in coolant density.

Passive shutdown systems, such as the Self-Actuated Shutdown System (SASS) [5] are also applicable. In SASS, a Curie-point magnetic alloy is utilised for automatic de-latching of control rods under high coolant temperature accident conditions, higher than for normal operating operation, but still below the coolant boiling point. A Hydraulically Suspended Rod (HSR) [6] system, where the control rods are automatically dropped into the core when the hydrodynamic force is reduced under accident flow reduction conditions, could also be used. In fast reactors, due to the sensitivity of the core reactivity to neutron leakage, it is also possible to consider using concepts like the Gas-Expansion Module (GEM) [7], where a decrease in core inlet pressure is exploited to increase neutron leakage under a flow reduction condition.

2.5.2. Decay Heat Removal (DHR)

Since an SFR is operated at nearly atmospheric pressure and at temperatures far below the coolant boiling point, coolant leakage or a pipe break does not lead to the same type of loss-of-coolant accident as postulated in an LWR, which has the potential for depressurisation, coolant boiling and loss of cooling capability. The requirements for core cooling of an SFR comprise keeping the sodium coolant level above the reactor core and the circulation of the liquid coolant to an appropriate heat sink for decay heat removal, using the normal heat transport system, a separate sodium-to-air heat exchanger, or any other system that would allow cooling of the sodium. As long as these two requirements are satisfied, significant core damage can be prevented. Natural circulation of a single phase sodium coolant can be effectively utilised if an adequate difference in height is available between the core and the heat exchanger, due to the fact that sodium has a relatively large density variation with temperature. Such passive DHRSs can be placed in different locations, e.g. in the RV or in the primary-/secondary-coolant circuits. Alternative emergency cooling can be made available via steam generators and guard vessels (GVs) for enhancing diversity.

2.5.3. In-Vessel Retention (IVR)

For an SFR, IVR is a safety design strategy aimed at ensuring long-term retention of core materials inside the RV for any accident situation, including those resulting in degradation or loss of core integrity, by providing coolability of the core materials under sub-critical conditions. This is typically accomplished by providing the means to keep the core submerged under the sodium coolant and the DHR paths available. Such an approach can be a key design measure to address “mitigation situations” for DECs.

2.6. Design Considerations for DEC

2.6.1. Anticipated Transient Without Scram (ATWS)

Postulated ATWS events (AOO followed by active shutdown system failure) cause an imbalance between generated power (heat) in the reactor core and its removal from the system. If adequate heat removal is not provided, substantial core degradation will occur, which may lead to severe consequences, such as large energy releases. Reliable means of maintaining the balance between heat generation and heat removal must be provided to avoid such consequences. This can be accomplished by ensuring alternate means of shutdown, including the use of inherent and/or passive reactor shutdown, as long as sufficient heat removal capability is also provided. Provisions for retention and cooling of degraded core materials are necessary to mitigate the consequences of a core damage. Provisions for ATWS can be summarised as follows.

- ➢ Prevention of core damage
  Means for maintaining an acceptable balance between reactor power and heat removal capabilities should be provided to avoid core damage, given an assumed failure of the active reactor shutdown function in AOOs. These capabilities should include inherent and/or passive means. In order to terminate the accident, means for reactor shutdown should be provided.

- Mitigation of core damage
  Provisions for prevention of a large energy release that could threaten the integrity of the containment and provisions for long-term cooling of a degraded core to avoid reactor coolant boundary failure, should be made available for achieving IVR against unprotected transients with core damage.

2.6.2. Loss of Safety Systems for DHR

In a situation of a failed DHRS after the reactor has been shut down and the heat generation has dropped to only a few percent of the nominal power shortly, the temperature of the reactor coolant system, including the core, coolant and reactor coolant boundary, increases. The rate at which the temperature increases depends on the overall heat capacity of the system, and it may take a long time before reaching temperatures that would threaten the core or system integrity. It is therefore possible to consider recovery actions for failed DHRSs and/or to implement back up cooling measures, before reaching unacceptable temperatures for SSCs. However, if no heat sink is available the coolant boundary will eventually fail due to creep damage, leading to release of radioactive fission products and sodium vapours into the containment. Such situations should be practically eliminated by design measures for enhanced core cooling capabilities. Provisions can be summarized as follows.

- Prevention of core damage
  Extension of the DHRS (normally designed for DBAs) capability should be considered, and other alternative cooling provisions should be made available to prevent core damage and reactor coolant boundary failures due to overheating, given the assumed causes of DHRS failures as DECs.

2.6.3. Reactor Coolant Level Reduction

If the core is uncovered, following events causing a reduction of the reactor coolant level, it is impossible to avoid core melt. Depending on the course of the accident and under some circumstances, significant radioactive material would be released into the containment atmosphere. Therefore, an uncovered core configuration should be practically eliminated by design measures. Provisions can be summarised as follows.

- Prevention of core damage
  RVs and GVs should be designed, manufactured, installed, maintained and inspected to have the highest level of reliability in order to prevent double leakage from RVs and GVs. For a loop-type reactor, measures for ensuring a minimal primary coolant level to prevent core damage should be provided against postulated leakage from the primary loop components and piping. If double leakage from RVs and GVs cannot be practically eliminated, the situation has to be considered for implementing design provisions.

2.7. Situations to be Practically Eliminated

Specific situations, whose consequences can lead to early or large radioactive release and which cannot be managed by the design at acceptable conditions, have to be practically eliminated by implemented design measures. These situations have to be demonstrated either as physically impossible by design, or as extremely unlikely to arise with a high level of confidence. Situations to be practically eliminated are part of the residual risk.

All potential situations which might lead to unacceptable radioactive release should be considered to identify the situations to be practically eliminated. Some examples of situations to be practically eliminated are as follows:

(1) Severe events with mechanical energy release exceeding the capability of the containment
- ➢ Power excursions for intact core situations (during power operation)[3]
  - ✓ Large gas flow through the core
  - ✓ Large-scale core compaction
  - ✓ Collapse of the core support structures

(2) Situations leading to failure of the containment with a risk of fuel damage
- ➢ Complete loss of the decay heat removal functions, leading to core damage and failure of the reactor coolant boundary
- ➢ Core uncovering due to sodium inventory loss

(3) Fuel degradation in the fuel storage or when the containment is not functional due to maintenance (e.g. opening containment for replacement of large equipment)
- ➢ Core damage without a functioning containment
- ➢ Spent fuel melting in the storage

Demonstrations of situations to be practically eliminated are made on a case-by-case basis, founded on deterministic methods, and in some cases supplemented with probabilistic studies. Deterministic demonstrations are based on the following general principles:

- ➢ Establishing a complete list of initiating events and/or accident sequences, including stress and human factors. For each event, provisions to either physically exclude the event or to make it very unlikely, should be established.
- ➢ If it is not possible to physically exclude a situation, favouring provisions that enable early detection and corrective actions to avoid degradation or to make the consequences acceptable. The accident sequences, which may have common mode failures, reducing the effectiveness of the provisions, should be identified and additional provisions taken to avoid their consequences.

In addition, probabilistic studies are performed to ensure the expected very low frequency of occurrence.

2.8. Specific Considerations on Reactivity Characteristics of SFRs

This section summarises the general reactivity characteristics of an SFR and contains concluding remarks on the sodium void reactivity in relation to SFR safety, emphasising that it is the overall reactivity feedback that is important and not any single part, such as sodium void. In addition, integrated transient analysis is required to determine if the sodium void worth of the design is acceptable or not since the entire core typically does not void during a postulated transient, but sodium voiding is usually predicted to initially occur locally with a much lower reactivity effect. Positive sodium void worth is not necessarily a problem, nor is negative sodium void worth necessarily a solution.

Normal operation, AOOs and DBAs

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[3] Note: As far as reasonably possible, severe accidents have to be managed with mitigation means, before implementing a practical elimination demonstration.

Various reactivity coefficients, such as the Doppler coefficient, sodium coolant temperature coefficient and cladding temperature coefficient, exist for the respective core constituents in relation to changes in temperature and geometry. The core characteristics are commonly represented by an integral effect from these reactivity coefficients, e.g. the total power coefficient, the isothermal temperature coefficient and the power/flow coefficient. For normal operations, these integral coefficients are required to have certain characteristics, such as a negative power coefficient, to allow stable operation and reliable control of the reactor. Within the AOO and DBA domains, a reactor shutdown by the reactor protection system determines the outcome; therefore, there are no definite additional requirements on the reactivity feedback, as long as the design limits are not exceeded.

The coolant temperature coefficient is defined as the rate of reactivity change due to the coolant density change with temperature. It is also related to the void coefficient, as the liquid phase turns to vapour phase at elevated temperatures (in excess of around 900C for a sodium coolant). In general, a core with a large coolant temperature coefficient has also a large sodium void coefficient and a large overall sodium void worth. Usually, all reactor cores have regions of positive void worth (the interior) and negative void worth (the periphery). The overall sodium void worth can be positive for large cores, and can be reduced to zero or become negative for small cores, due to the larger neutron leakage. For DBAs, sodium void worth is not particularly relevant, since coolant boiling is not expected to occur.

Core damage prevention under DECs

As long as boiling of the sodium coolant is prevented, any effect of sodium void worth is not relevant. Although the sodium coolant temperature coefficient might be positive, the importance of various reactivity coefficients comes into play during ATWS events. No additional requirements are specified for the integral reactivity coefficients, on the condition that core damage is prevented by a passive shutdown device/measure during ATWS events. When an inherent reactivity feedback approach is used, instead of a passive shutdown device/measure, the total power coefficient, isothermal temperature coefficient and power/flow coefficient should all be sufficiently negative and respond in a timely manner to quickly achieve sub-criticality during ATWS events.

In a loss-of-flow transient, with failure of the reactor protection system (i.e. without scram), the coolant temperature and/or void reactivity could pose a challenge, depending on their magnitude and timing, since they can be quite positive (can add reactivity to the core when the coolant temperature rises). Therefore, the coolant temperature coefficient should be limited within a range where other reactivity feedback effects (such as control rod expansion and/or core expansion) can compensate. In this context, the positivity of the sodium void worth plays a minor role. The reactor core damage should be prevented by applying a passive shutdown device/measure or by reliance on inherent safety with net negative reactivity feedback at elevated temperatures.

Mitigation of consequences of core damage under DECs

A coolant phase change and a material relocation of a degraded core can have significant reactivity consequences, both favourable and unfavourable, depending on design choices. When a core damage and material relocation occur, prompt criticality should be avoided in order to prevent large mechanical energy release. In particular, the maximum net reactivity during an initiating phase should be limited below 1 dollar. Positive reactivity effects, such as sodium boiling, should be limited considering e.g. spatial and temporal incoherence of the sodium voiding during the transient so that other negative reactivity components from the Doppler effect, fuel expansion and failed fuel dispersion can overcome them during the entire phase of the transient. Sodium void worth is usually not relevant during transition phases, since the original core geometry does not exist, and the larger reactivity effects of molten fuel and cladding motions dominate the overall reactivity [8].

3. Conclusions

The GIF SDC TF has developed "Safety Approach SDG" which is more detailed recommendation and guidance to safety design in application of SDC for Gen-IV SFRs emphasizing on the use of passive/inherent safety features, consideration of SFR specific safety characteristics, and the reliable design measures for prevention and/or mitigation of severe accidents. Currently the report was disseminated to international organization/group such as IAEA and GSAR for external review. The constructive feedbacks at international level will enable to contribute to the establishment of common understanding of safety design approach on Gen-IV SFRs.

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Appendix 1: Reference

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