Sodium-Cooled Fast Reactor (SFR)

System Safety Assessment

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AC:	Alternating Current
AOO:	Anticipated Operational Occurance
ASTRID:	Advanced Sodium Technological Reactor for Industrial Demonstration
ATWS:	Anticipated Transient Without Scram
CDA:	Core Disruptive Accident
CEFR:	China Experimental Fast Reactor
CDFR:	China Demonstration Fast Reactor
CP:	Corrosion Product
DBA:	Design Basis Accident
DEC:	Design Extension Condition
DHRS:	Decay Heat Removal System
DiD:	Defence in Depth
EBR-I:	Experimental Breeder Reactor-I
EBR-II:	Experimental Breeder Reactor-II
FBTR:	Fast Breeder Test Reactor
FP:	Fission Product
GEM:	Gas-Expansion Module
GEN IV:	Generation IV
GIF:	Generation IV International Forum
GV:	Guard vessel
HRS:	Hydraulically Suspended Rod
IHX:	Intermediate Heat Exchanger
LOHRS:	Loss of Heat Removal System
LOHS:	Loss of Heat Sink
LORL:	Loss of Reactor Level
ODS:	Oxide-dispersion Strengthened
PFBR:	Prototype Fast Breeder Reactor
RV:	Reactor vessel
SASS:	Self-actuated Shutdown System
SDC:	Safety Design Criteria
SDG:	Safety Desing Guidelines
SFR:	Sodium-cooled Fast Reactor
SG:	Steam Generator
ULOF:	Unprotected Loss of Flow
ULOHS:	Unprotected Loss of Heat Sink
UTOP:	Unprotected Transient Overpower

Sodium-Cooled Fast Reactor (SFR) Safety Assessment
Risk and Safety Assessment White Paper

1 General overview of the performance goals

The Sodium-cooled Fast Reactor (SFR) uses low-pressure liquid sodium as the reactor coolant, allowing high 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. Plant size options under consideration range from small and medium sized reactors (50 to 300 MWe) to larger plants up to 1 500 MWe. The outlet temperature is 500-550°C for the options, which affords the use of the materials developed and proven in prior fast reactor programmes. The SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of minor actinides. However, this requires that recycle fuels be developed and qualified for use. Important safety features of the SFR system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate coolant loop between the radioactive sodium in the primary system and the power conversion system. Water/steam, supercritical carbon-dioxide or other gases are considered as working fluids for the power conversion system to achieve safety and reliability. With innovations to reduce capital cost, the SFR is aimed to be economically competitive in future electricity markets. In addition, the fast neutron spectrum greatly extends the uranium resources compared to thermal reactors. The SFR is considered to be the nearest-term deployable GEN IV system especially for actinide management.

The SFR is typically considered within the context of a fuel cycle to facilitate actinide management,. The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle. Fast reactors hold a unique role in the actinide management mission because they operate with high-energy neutrons that are more effective at fissioning transuranic actinides. The main characteristics of the SFR for actinide management mission are:

- Enhanced utilization of uranium resources through more efficient management of natural materials.
- Consumption of transuranics in a closed fuel cycle, thus reducing the radiotoxicity and heat load, to facilitate waste disposal and geologic isolation.

The reactor unit can be arranged in a pool layout or a compact loop layout. Three options are considered in the GIF SFR:

- 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 upon 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 and potentially minor actinides, supported by a fuel cycle based upon advanced processing.
- 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.

2 Historical review of, and feedback from, past construction and operation experiences

With more than 20 reactors built and in operation around the world and combining nearly 400 reactor years of operation, SFRs benefit from extensive design and operating experience feedback. Experimental reactors, industrial prototypes and industrial-sized reactors were built and operated in a number of countries. The first nuclear power reactor of this kind was the EBR-I at the Idaho National Laboratory (INL) in the USA (1.4 MWth6, 0.2 MWe). The EBR-I operated between 1951 and 1963. A number of SFRs have been put into operation since then in France, UK, Germany, USSR, USA. Japan and India. Several SFRs are still in standby or operation: BOR-60, BN-600 and BN-800 in Russia, CEFR in China, FBTR in India, Joyo and Monju in Japan.

Several projects are under way, with varying degrees of advancement:

- In India, the 500 MWe Prototype Fast Breeder Reactor (PFBR).
- In Russia, n the BN-800 reactor (800 MWe). Russia is also considering developing a commercial version with a power output of 1 200 MWe (BN-1200) based on the BN-800 reactor.
- In Japan, restart of Joyo, improvement of Monju maintenance and design study of demonstration/commercial reactors are ongoing.
- The China Demonstration Fast Reactor (CDFR) project, which has a generating capacity of between 600 and 900 MWe, is under way.
- South Korea is considering building a 390 MWth/150 MWe prototype SFR.
- In the US, the TerraPower company is developing a 1 200 MWth/500 MWe SFR design while GE Hitachi Nuclear Energy has developed a 310 MWe reactor known as PRISM (Power Reactor Innovative Small Module).
- In France, the 600 MWe ASTRID reactor is expected to be a prototype of Generation IV SFRs designed to provide experimental qualifications, industrial validations and the maturity and operating experience feedback needed for the growth of the industry.

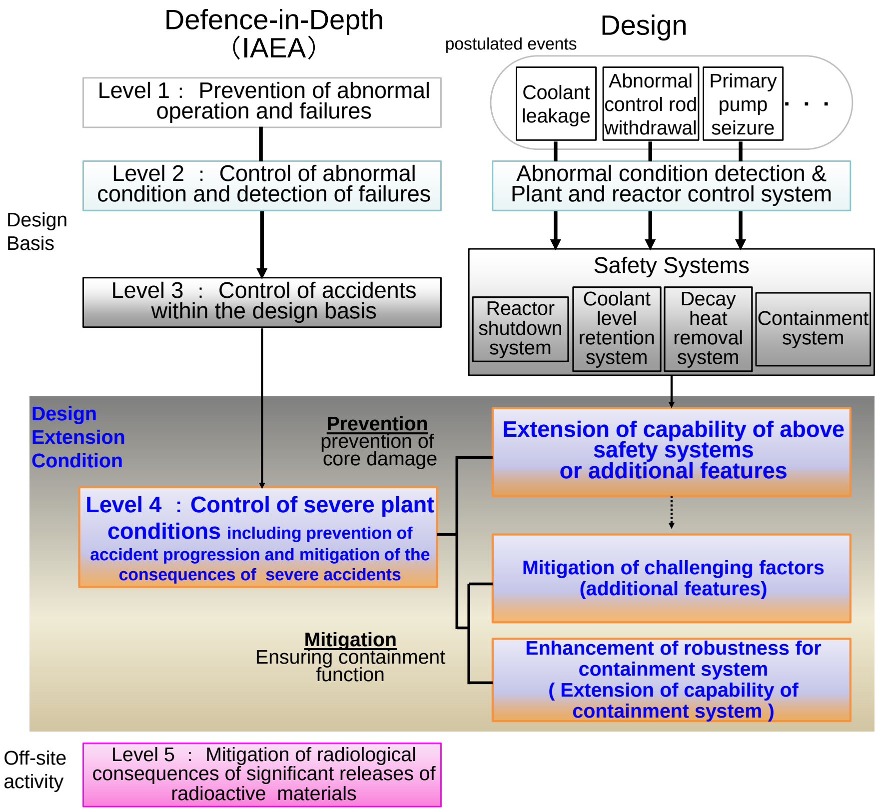
The main advantage of SFRs lies in the use of a liquid coolant that is low pressure and whose temperature under normal operating conditions offers a significant margin (300°C) over its boiling point, creating significant grace periods of several hours in the event of loss of cooling. SFR cooling systems are categorized into pool and loop-types. Both pool-type and loop-type SFRs benefit from highly favourable operating experience feedback on radiation protection. Worker exposure in pool-type SFRs is lower by a factor of 10 than that observed in PWRs.

3 Level of ongoing safety-related research and development.

When developing a new reactor system, accumulation of experiences on fuel, safety, and material behavior is extremely important and takes a significant amount of time. In the SFR case, those experiences have been successfully accumulated during the past seventy years world-wide. There were and are a lot of experimental and prototype SFRs. Fuel/subassembly specification, safety designs and materials of future SFRs are based on perspectives from those experiences. From the safety point of view, a lot of large demonstration experiments have been conducted and evaluation tools validated by those experiments have been developed. In several reactors, demonstrations of natural convection decay heat removal operation have been successfully conducted. Safety performances of future SFR will take benefit of this important feedback and have to comply with the stringent objectives of the GEN IV on the other hand[1].

Based on that wide range of experiences, SFR is now ready to discuss global standards of safety criteria. The GIF high-level safety standard is already written in “Basis for safety approach for design & assessment of Generation IV Nuclear Systems”[2]. However, there is a large gap between the high-level safety standards and actual safety designs. Safety Design Criteria (SDC) is being prepared as middle-level global safety standards[3]. The SDC provide specific safety criteria for general and each component such as fuel, core, cooling system and containment in SFR system regarding the GIF safety goals, SFR safety features, innovative technologies and lessons learned from the 1F accident. The SDC follows the DiD philosophy as the most basic safety approach. The safety design based on DiD provides design measures for every plant state, i.e. normal operation, AOO, DBA and DEC (Figure 1). The design for operational states and design basis accidents shall be conservative with due account of uncertainties of design conditions and transient phenomena. For DEC, the safety design process used to prevent significant radioactive material releases to the environment shall be based on best estimate analysis.

Figure 1: Postulated events, Safety Systems and Prevention/Mitigation measures for Design Extension Conditions (in relation to the Defence-In-Depth Levels 1-4)



In order to ensure the safety of a nuclear power plant facility, the release of radioactive materials must be limited. The appropriate management of radioactive materials and measures to accommodate abnormal events must therefore be provided for the reactor system, as well as for the fuel handling and storage systems and for the radioactive waste management facility.

4 Achievement of fundamental safety function

4.1 Reactivity control

4.1.1 Control systems

The GIF SFR SDC/SDG[3, 4] requires two diverse and independent active reactor shutdown system, passive shutdown system and inherent reactivity feedback for reactivity control.

Two diverse and independent active reactor shutdown systems, consisting of control rods, actuation mechanisms, detectors and signal processing systems. Two shutdown systems should be designed to have independence and diversity to the extent practicable to prevent common cause failures. Examples of design measures are:

- Different structures/mechanisms for rod insertion, different designs for the control rods and their guide tubes

- Physical separation of electric distribution boards and cables; isolated arrangements, divided by walls, etc.
- Electrical independence of instrumentation and control systems.

Detection parameters for the reactor shutdown systems should be diverse to the extent practicable. At least one of the two reactor shutdown systems should be designed according to a single failure criterion that applies to components of actuation mechanisms, detectors and signal processing systems. Fail safe features, such as control rod insertion by loss of the holding function in case of electric power supply failure, should be considered.

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) 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) system, where the control rods are automatically dropped into the core when the hydraulic 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), where a decrease in core inlet pressure is exploited to increase neutron leakage under a flow reduction condition.

Inherent reactivity feedback, based on the total power coefficient, isothermal temperature coefficient and power/flow coefficient, should be negative to reduce the core power at elevated temperatures in balance with available heat rejection capacities during an ATWS. Complementary reactor shutdown measures should be provided in order to make the reactor core subcritical in the long term.

4.1.2 Risk of criticality

Fast reactors including SFR have a characteristic that a core is not in the most reactive configuration, and thus they have possibility to result in positive reactivity changes when assuming coolant boiling, clad discharge and fuel concentration resulting from a core disruptive accident (CDA). The GIF SFR SDC/SDG[3, 4] requires several items to avoid sever re-criticality during CDA. The GEN IV SFRs are designed taken into account those requirements.

Core reactivity characteristics should be designed so as to prevent prompt criticality, i.e. $\rho net < 1\$$ during the initiating phase of unprotected transients. Positive reactivity effects, such as sodium boiling, should be limited so that negative reactivity effects e.g. Doppler effect, fuel expansion and failed fuel dispersion are sufficient to counteract the positive reactivity effects. Design parameters, such as sodium volume fraction, core height and other geometric parameters, should be properly chosen based on the effects of sodium void worth during transients.

Core design parameters, such as core height, should be properly chosen to obtain effective negative feedback due to failed fuel dispersion. Fuel reactivity feedback is dependent on the choice of fuel type for the reactor. The effects should be appropriately included in transient analysis of an accident.

In the course of core degradation during unprotected transients, measures should be provided to prevent prompt criticality, potentially leading to large mechanical energy release. For this purpose, design measures, such as facilitating molten fuel discharge outside the core, neutron absorber added to the core, and core cooling to prevent failure progression, i.e. early termination, should be taken. These measures should include consideration of using favorable inherent phenomena occurring in the course of a core degradation.

When developing a new reactor system, accumulation of experiences on fuel, safety, and material behavior is extremely important and takes a significant amount of time. In the SFR case, those experiences have been successfully accumulated during the past seventy years world-wide. There were and are a lot of experimental and prototype SFRs. Fuel/subassembly specification, safety designs and materials of future SFRs are based on perspectives from those experiences. From the safety point of view, a lot of large demonstration experiments have been conducted and evaluation tools validated by those experiments have been developed. Safety performances of future SFR will take benefit of this important feedback.

4.2 Decay heat removal

4.2.1 Thermal inertia and grace period

The GIF SFR SDC/SDG[3, 4] summarized SFR features on decay heat removal system (DHRS). Important safety features of SFR systems include a large margin to coolant boiling during normal operating conditions, a primary system that operates near atmospheric pressure, and an intermediate coolant loop between the radioactive sodium in the primary circuit and the power conversion system. Those features could provide certain grace period for decay heat removal operation. 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. 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 heat transportation systems.

4.2.2 Diversification, active and passive systems

For ensuring reliability of a DHRS, the GIF SFR SDC/SDG[3, 4] requires redundancy and diversity on the DHRS. In order to avoid a common cause failure in case of design basis accidents, including external events, redundancy and diversity should be adequately provided. DHRSs should be designed to have redundancy for ensuring sufficient core cooling capacity, considering loss of off-site power and single failure of components. For instance, three systems at 100% of the capacity required for decay heat removal, or four systems at 50% capacity, would be provided, considering one system failure as an initiating event and a single failure in another system, depending on the design. Diversity in the system configuration and/or operation mode (forced circulation and natural circulation), as well as physical separation, should be introduced. Examples of potential common cause failures are earthquakes, aircraft crashes, flooding, sodium freezing in air coolers, sodium fires on the reactor roof, and loss of all AC power. Diversity in system configurations, components, working fluids (sodium, gas), operational principles (forced circulation and natural circulation), and physical separations, should be adequately introduced.

Natural circulation of a single phase sodium coolant can be effectively utilized 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 Decay Heat Removal Systems (DHRSs) can be placed in different locations, e.g. in the reactor vessel (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.

Several experimental and prototype reactors already demonstrate passive decay heat removal by natural circulation[5].

For design extension conditions, the GIF SFR SDC/SDG also requires alternative decay heat removal system in addition to a decay heat removal system for design basis accidents.

4.3 Confinement of radioactive materials

4.3.1 *Materials*

For radio-active materials in SFR, following items have to be taken into account[6]:

- radio-activation of cover gags
- radio-activation of coolant
- Corrosion products (CP) from core components and structural material
- fission products (FP) from fuel
- Tritium

Since SFRs adopt argon gas for cover gas of coolant sodium, ^{40}Ar could be activated to ^{41}Ar by capturing neutron. For coolant itself, ^{23}Na could be activated to ^{24}Na or ^{22}Na. For ^{24}Na and ^{22}Na halving times are 15 hours and 2.6 years, respectively. For core components and structural materials, SFR generally adopt austenitic stainless, high nickel, high chromium steels. Major radio-active elements could be ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{58}Co and ^{60}Co. For the plant standby conditions, ^{58}Co, ^{60}Co and ^{54}Ma are important, since they have long halving time. FP behaviors depend on fuel burnup, FP distribution inside fuel pin, volatility and solubility in sodium. Detail of the FP will be discussed in the section of source term. Tritium is created from ^{10}B in absorber and involved in FP. Tritium which is created from impurity in sodium such as Li can be neglected.

4.3.2 *Safety barriers*

Safety barriers for radioactive materials in SFRs depend reactor types and designs. However multiple barriers against the release of radioactivity generally consist of fuel elements, fuel cladding, reactor coolant and cover gas boundaries, guard vessels (and pipes in a loop configuration), and containments.

For fuel cladding, SFR generally adopt fuel pins made of steel based materials such as stainless steel and oxide-dispersion strengthened (ODS) materials. Typical indices for design limits of the reactor core are the maximum temperatures of the fuel, cladding, and coolant[3, 4]. Coolant boiling should be prevented to maintain the cooling of the reactor core, and to limit its contribution to the net reactivity. Cladding damage during normal operation and anticipated operational occurrences should be limited so that specified acceptable fuel design limits are not exceeded. To evaluate design limits, accumulation of irradiation data under steady and transient conditions are necessary revealing fuel-cladding interaction under steady state and accidental conditions. SFRs have significant advantage on this point, since irradiation data for various fuel, cladding conditions have been already accumulated for pas experimental and prototype reactors.

For the reactor coolant and cover gas boundaries, SFR adopt stainless steels and the compatibility with sodium coolant is excellent. The integrity of the reactor coolant boundary should be maintained during abnormal temperature increases by limiting the maximum coolant temperature and the time at that temperature.

For the containment including guard vessel, the configurations of the barrier strongly depend on the reactor type such as pool and loop types. For pool type reactors, combination of a guard vessel and top dome could consist a containment. For loop type reactors, containments covering a reactor vessel room and primary cooling system rooms are designed. Typical indices for the containment are the containment temperature and internal pressure.

The GIF SFR SDC/SDG[3, 4] requires integrity of containment against sodium fire and other challenges on penetrations. The containment structure and the systems and components affecting the leak tightness of the containment system shall be designed and constructed so that the leak rate can be tested after all penetrations through the containment have been installed and, if necessary during the operating lifetime of the plant. The design basis for the containment shall consider pressure increase and thermal loads due to sodium fire. And the number of penetrations through the containment shall be kept to a practical minimum and all penetrations shall meet the same design requirements as the containment structure itself. The penetrations shall be protected against reaction forces caused by pipe movement or accidental loads such as those due to missiles caused by external or internal events.

Additionally, design features to control fission products, sodium, hydrogen and other substances that might be released into the containment shall be provided as necessary:

- To reduce the amounts of fission products that could be released to the environment in accident conditions;
- To prevent or mitigate sodium combustion, sodium-concrete reaction, and debris-concrete interaction and to control the concentration of hydrogen in the containment atmosphere in accident conditions so as to prevent thermal, deflagration or detonation loads that could challenge the integrity of the containment.

For a further barrier, some design adopt containment surrounded by a confinement area in which an emergency gas treatment system is installed. The function of the confinement area is to reduce the release rate of radioactive materials through the penetrations on the containment boundary.

4.3.3 By-pass

Lines that penetrate the containment, as part of the reactor coolant boundary and the reactor cover gas boundary, and lines that are connected directly to the containment atmosphere shall be fitted with at least two adequate containment isolation valves or check valves arranged in series, and shall be provided with suitable leak detection systems for preventing the containment bypass of radioactive materials. Containment isolation valves or check valves shall be located as close to the containment as is practicable, and each valve shall be capable of reliable and independent actuation and of being periodically tested.

For severe conditions, complete loss of heat removal, steam generator tube rupture, prompt recriticality in the RV and molten fuel-coolant interaction in the RV could cause bypass of the containment. Complete loss of heat removal causes coolant boundary failure due to thermal loads. Steam generator tube rupture, prompt recriticality in the RV or molten fuel-coolant interaction in the RV causes primary/secondary boundary failure due to mechanical loads. Those causes of containment bypass should be taken into account in the design and prevented[3, 4]. Measures for prompt recriticality and molten fuel-coolant interaction in the RV are discussed in reactivity control sections and measures for complete loss of heat removal is discussed in decay heat removal sections.

Since heat exchange tubes of the intermediate heat exchanger are involved in containment boundary, intermediate heat exchanger (IHX) tube failure means containment bypass. When a steam generator tube fails, sodium-water reaction causes. In large sodium-water reaction, the pressure in the secondary coolant system, which includes primary/secondary coolant interface in the IHX, increases typically due to initial spike-shaped pressure and quasi-steady pressure. The former is produced just after the heat exchange tube failure in a steam generator, while the latter is produced by hydrogen and heat generated by sodium-water reaction for relatively longer duration in the failure propagation process. It could lead to failures of the secondary coolant system components, piping, and heat exchange tubes of the intermediate heat exchanger should not fail against these pressure loadings.

To evaluate effect of sodium-water reaction, both leak initiation and its propagation and by providing substantial margins in the leak magnitude. Propagation of sodium-water/steam boundary failures should be evaluated and considered in determination of the design basis leak, assuming a range of initial boundary failures (e.g., from a small leak to guillotine rupture of a steam generator tube), and taking function of mitigation systems such as leak detection and water/steam isolation/pressure relief, into account with conservative manner. High temperature and corrosive sodium-water reaction jet generated by sodium-water reaction has thermal and chemical effects on adjacent heat exchange tubes. If early termination of water leak from the initial failure position cannot be achieved, the failure would propagate to the adjacent tubes, enlarging the failure. The failure propagation typically has two mechanisms: wastage type failure and overheating type failure. In the former, high temperature and corrosive sodium-water reaction jet from an initial failure site hits an adjacent tube and makes it thin and fail (target wastage). The secondary tube leak causes additional target wastage on another adjacent tubes (multi-wastage). As a result of such kind of failure propagation, temperature of the reaction area goes up and the overheating type failure will happen since mechanical strength of the tubes will be reduced due to the overheating. The overheating type failure tends to have a larger failure area because it affects larger area than a target wastage. In order to minimize the failure propagation, leak detection and steam/water release from the failure component are essential. Following points should be considered in the design of steam generators and detection and mitigation systems.

- Failure propagation and detection time depends on the initial leak rate.
- Small leaks, typically a pin hole, take long time to propagate and also take long time to detect.
- Large leaks, typically a double ended break, can detect and mitigate by rupture disks in a short time.

Middle leaks in between small and large leaks have a potential to cause the overheating type failure since its propagation time is shorter than that of small leaks.

The containment structure should be designed to prevent the mechanical failure by the over-pressurization due to sodium leaks. In addition, the containment structure should be designed to prevent hyperthermia (thermal failure) of a storage structure steel sheet and the frame. During accidents involving sodium leaks, it is necessary to evaluate the pressure and the thermal load to the containment structure caused by sodium combustion in the atmosphere in the containment structure. In addition, if necessary, in order to mitigate the effect of the pressure and the thermal load, a single or a combination of the following measures should be applied to the containment structure.

- To install floors, pits, and retention tanks, etc. in order to retain sodium in limited areas.
- To mitigate sodium combustion reaction by filling containment with inert gas.

Facilities which contact or retain leak sodium should be designed to withstand thermal influence from sodium combustion and chemical influence from sodium products. In addition, influence with concrete temperature rising should be considered and thermal insulation materials or cooling facilities should be installed as needed.

Possibility of hydrogen generation and hydrogen combustion in accident conditions involving sodium leaks should be evaluated. It should be confirmed that hydrogen generation and hydrogen combustion in the containment boundary are prevented. In addition, if the possibility of contact between sodium and concrete cannot be eliminated, countermeasures for the combustion of accumulated hydrogen should be taken.

4.3.4 Source term

SFRs have significant advantages since there have been wide range of experiences on accidents/incidents with fuel damage, experiments with cladding failure fundamental experiments on material data and numerical codes development on transportation of radioactive materials that provide specific insights into the behavior of radionuclide release and transport from the core.

For oxide fuel[6], FP behaviors depend on volatility and solubility in sodium as follows;

- Noble gas (Volatile and non-soluble in sodium): Kr, Xe
- Volatile and soluble in sodium: Cs, Rb, I, Br, Te, As, Se, Ag, Cd, In. Sn, Sb
- Non-volatile and non-soluble metal in sodium: Nb, Mo, Tc, Ru, Rh, Pd
- Non-volatile and non-soluble oxide in sodium: Sr, Ba, Y, Zr, La, Ce, Pr, Nd, Sm, Eu, Gd

For source terms, noble gas and volatile materials are important. In fuel pin failures, noble gas such as Kr and Xe accumulated in gas plenum is rapidly released to coolant sodium. Most of released noble gas reaches to the cover gas because of their non-solubility in sodium. For other volatile materials, cesium and iodine are important as source term. In sodium coolant, cesium exists as metal and gas-liquid equilibrium partition coefficient in cesium-sodium system is 20 to 100[7]. With this large partition coefficient, cesium basically released from sodium. On the other hand, iodine which has a smaller partition coefficient (0.02 to 0.5 at 450deg-C and 0.3 to 0.8 at 650deg-C)[7] and certain amount of iodine could remain in coolant sodium as NaI. For other materials, experimental data is also accumulated.

For metal fuel, the SRE, EBR-II and Fermi 1 accidents/incidents experiences involving metal fuel showed that no radionuclides other than the noble gases of xenon and krypton were found in the cover gas region. This implies that significant retention occurred, whether in the fuel or in the primary sodium, of many of the important radionuclides that are commonly a concern during light-water reactor core damage accidents. Metal fuel cladding breach results in a release of fission gases (including Xe and Kr) and bond sodium (which may contain Cs), but does not imply a release of fuel material. Metallic fuel does not react with the sodium coolant, limiting the likelihood of fuel damage propagation. The retention of radionuclides, other than the noble gases, in the fuel pin and primary sodium appears to be very high, (even during significant fuel melting and relocation, such as at Fermi 1 and the EBR-II experimental capsule) [8, 9].

In case of sodium combustion. radio-active materials in primary sodium could be source term. SFRs also accumulated experiences on studies about sodium combustion with radio-active materials. Experimental data on radio-active material release with sodium vaporization, sodium combustion and sodium-concrete reaction have been accumulated. And numerical calculation codes for transportation of radio-active materials under those conditions have also been developed. For example, the development of CONTAIN[10] was initiated at SNL early in 1980s as a part of safety evaluation studies in the case of hypothetical core disruptive accident for the Clinch River Breeder Reactor Project (CRBRP). Improvement of CONTAIN adding FBR specific models was conducted by 1993 under the international collaboration among SNL, PNC, and KfK and GRS in Germany. Recently CONTAIN was improved to evaluate the ex-vessel accident progression of level-2 PRA for LMFRs. This code can treat many important phenomena consistently such as sodium fire, radioactive aerosol behavior, a water release from heated concrete, hydrogen burn, sodium-concrete reaction and core debris-concrete interaction occurred in the accident[11].

5 Management of design extension conditions (severe accidents)

Design extension conditions have been introduced in the requirements for the design of nuclear power plants for the purpose to further improve safety by enhancing the plant's capability to withstand accidents that are more severe than design basis accidents.

According to the IAEA definition design extension conditions are postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include conditions in events without significant fuel degradation (DEC prevention) and conditions with core melting (DEC mitigation).

5.1 Prevention and Mitigation

Safety studies based on deterministic and probabilistic approaches provide information to identify design extension conditions. For SFR, severe accident sequences, which may lead to core damage and radioactive materials release, are categorized into two major types, namely ATWS and Loss of Heat Removal System (LOHRS). These two major types can be sub-categorized: e.g. Unprotected Loss-of-Flow (ULOF), Unprotected Transient Overpower (UTOP), Unprotected Loss of Heat Sink (ULOHS) for ATWS, and Loss Of Reactor Level (LORL) and Loss Of Heat Sink (LOHS) for LOHRS[3]. Detail prevention and mitigation measures against those events are described in the previous sections.

5.2 Situation to practically eliminate

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 vapor into the containment. Such situations should be practically eliminated by design measures for enhanced core cooling capabilities as follows:

- For DBA, a decay heat removal system should be provided to deal with postulated initiating events typically caused by single failure of an SSC.
- For DEC, design measures should be provided against initiating events, which are more severe than DBAs, or which originate from multiple failures of SSCs.
- Proven technology, based on the design, construction and operation experience of SFRs, should be applied to the basic design of decay heat removal systems.
- Extension of capabilities to deal with DECs, e.g. additional decay heat removal systems, increased capacities of heat removal, and operation with natural as well as forced circulation, should be considered. Application of mobile power sources and manual operations in case of loss of power are examples of extensions of such capabilities.
- Ensuring diversity in systems is essential for improving the overall reliability. Duplication of systems does not bring the same reliability benefits. It is required to maintain heat removal functions, even under postulated severe external hazards, such as earthquakes, flooding, tsunami and missiles leading to a common cause failure.
- An SFR should proactively utilise its natural circulation capability to an ultimate heat sink (atmosphere), since this can significantly contribute to improving the reliability of the heat removal capability, even under long-term loss of power supplies. Natural circulation can be used as a measure for DBAs, as well as for DECs.
- Robust demonstrations of practical elimination should consider independence between safety systems for DBAs and decay heat removal capabilities for DECs. If necessary, additional independent decay heat removal systems should be installed as an ultimate measure.
- It is necessary to clarify all credible factors leading to loss of decay heat removal function and to confirm that measures can be implemented to overcome all of them.
- Each system, related to decay heat removal, should be able to demonstrate that it can perform its function as expected.

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. Reactor Vessels (RVs) and Guard Vessels (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 as follows:

- The RVs and GVs should be designed, manufactured, installed, maintained and inspected to have the highest level of reliability.
- Due design considerations should be taken to prevent dependent failures and common cause failures between RVs and GVs, even under postulated severe external hazards, such as earthquakes.
- In case double failures of RVs and GVs cannot be practically eliminated, provisions should be available to retain the coolant level to keep the reactor core covered.

The GIF-SFR-SDC/SDG[3, 4] showed specific examples of situations to be practically eliminated for SFRs as shown in Table 1.

Table 1: Examples of Situations to be practically eliminated for an SFR

SFR	
Practically eliminated situations	**Reason for choice**
Power excursions for intact core situations	Since the core is not in the most critical configuration, positive reactivity insertion might happen due to extreme initiating events, such as a large gas flow through the core, large-scale core compaction, or collapse of the core support structure.
Complete loss of heat removal function that could lead to core damage and failure of the reactor coolant boundary	If no heat sink is available, a coolant boundary failure will occur, either due to creep damage anywhere in the primary and secondary coolant systems, or from melt-through by degraded core materials.
Core uncovering due to sodium inventory loss	If the core becomes uncovered, it is impossible to avoid a core melt. Depending on the course of the accident and under some circumstances, significant radioactive materials would be released into the containment atmosphere.
Core damage during maintenance, or spent fuel melting in the storage	If a core damage or fuel melting occurs during maintenance (when the containment is not functional), or at the fuel storage outside the containment, significant radioactive release might happen.

6 Safety of the fuel cycle

6.1 Type of fuel

6.2 Management of waste (quantity, quality)

- ***Pyroprocessing***

Pyroprocessing is able to recover transuranic (TRU) elements, such as Np, Pu, Am, etc., along with U in a molten salt system due to their thermodynamic and electrochemical characteristics; TRU elements have quite similar deposition potentials with U, which allows TRU elements to be codeposited with U on the liquid cadmium cathode. Recovered product from pyroprocessing can then be used as a fuel for the fast reactor and this inseparable product significantly increases intrinsic proliferation resistance. Aqueous processes can also recover groups of elements. However, in the case of metal fuels, pyroprocessing would be more appropriate technology since the molten salt used in pyroprocessing is known to be stable under a strong radioactive environment and pyroprocessing can handle short-cooled spent fuel having high heat and high radioactivity. This high heat and radioactivity environment increases intrinsic barriers to human intrusion. Therefore, pyroprocessing facility should be comprised of hot cells, implying consequent application of better containment and surveillance concepts with a strengthened physical barrier.

Pyroprocessing research and development must be continued to increase process efficiency and throughput, and to enhance economic viability and proliferation resistance, which also develop the range of interrelated technologies needed; The development of safeguards technology is essential to ensure proliferation resistance and NDA and DA technologies, online and offline monitoring for tracking special nuclear materials, will be stringently deployed.

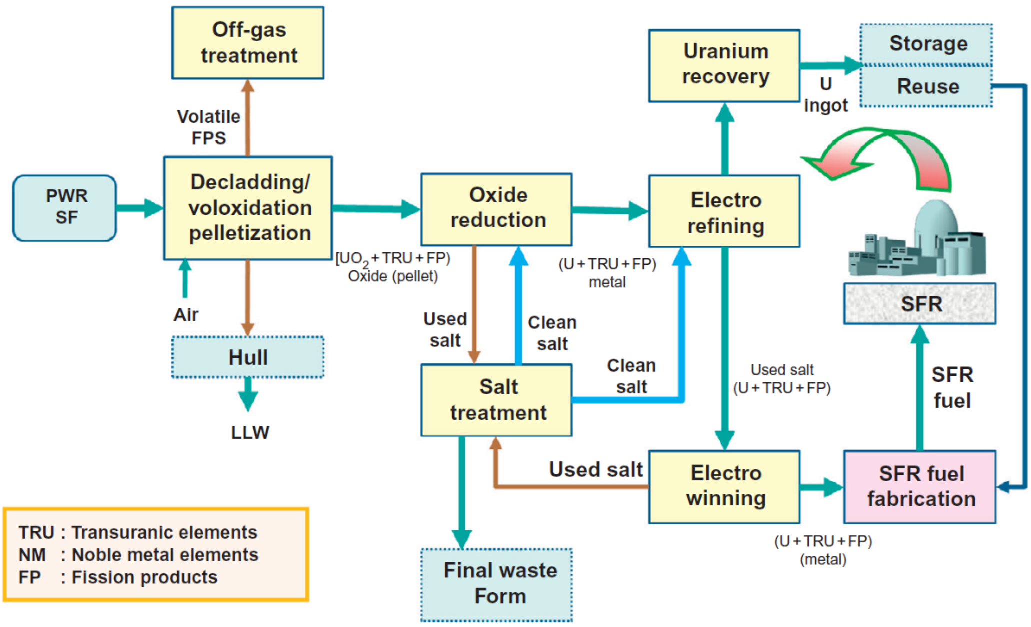
Pyroprocessing flow sheet

Figure 3 shows pyroprocessing flow sheet. Pyro-chemical processes are used to group-recover actinides using electrochemical processing. As a metal phase is required to flow electric current, an upfront oxide-reduction process is necessary to convert oxide spent fuel to metal.

In the headend stage, separated oxide spent fuel from the assembly is followed by a voloxidation process that changes the physical form of oxide spent fuel to powder, and volatile elements, in which process iodine and cesium are released. This powder is compacted to a pellet, followed by sintering at high temperature, which is transferred to the following oxide-reduction process.

Most strontium and residual cesium in spent fuel are dissolved into the molten salt in the oxide-reduction process and uranium is mainly recovered on a solid cathode in the electrorefining process. The rest of dissolved elements in a molten salt such as residual U, TRU, and REE (Rare Earth Elements) are recovered in the electrowinning process with the liquid metal cathode. Molten salts from oxide-reduction and electrowinning processes are recycled after purification of contaminated salt by appropriate methods.

Figure 3: Pyroprocessing flow sheet



Waste treatment

In a pyroprocessing for PWR used fuel, there is a specific waste stream, compared with other wet reprocessing processes. Fission products in a used fuel are group-separated due to the characteristics of each process during the pyroprocessing. Volatile (C-14, H-3, Kr/Xe), semivolatile (Cs, I, Tc and Se), alkali earth metal (Sr/Ba), lanthanide group (La~Lu) and Noble metal group (Tc and etc) are generated as main wastes without a reaction medium like electrolyte or solvent. This is one of unique characteristics of the pyroprocessing.

Metal wastes, disassembly parts, hull waste and anode sludge, are generated from the disassembling/decladding of used fuel and electrorefining process. These wastes are treated by a super-compaction or melting with some metal alloys for final disposal. Especially, noble metal waste in anode sludge containing Tc is destined to be immobilized into SS-15Zr alloy matrix well known in the literature. Hull containing minor quantity of TRUs is expected to be classified as an intermediate-level waste that is consolidated into a specific alloy like Zr-based matrix under development for final disposal. Disassembly parts have some activation products and they are heat-generating waste. They are consolidated by a super-compaction as a well known technology for final disposal.

Gaseous wastes containing Cs, Tc, I or other volatile FPs are selectively captured by a specific filter system and they are generated as a filter waste containing each FPs. The filter wastes are destined to be consolidated by HIP (Hot Isostatic Pressing), vitrified by CCIM(Cold Crucible Induction Melter) or in a specific storage system. Cs filter has mainly an aluminosilicate species (Al-O-Si) to capture Cs to produce Cs-aluminosilicate compounds. It is converted to pollucite-based ceramic or aluminosilicate glass for final disposal. Tc filter consists of calcium oxide and other inorganic materials, where Tc or other volatile transition metals is reacted with CaO to produce stable compounds like Ca(TcO4)2. This is consolidated with some inorganic binder during HIP process. Commercial iodine adsorbent is Ag-X molecular sieve that is very effective to capture gaseous iodine. However, iodine filters including Ag-X has problems on the consolidation for final disposal. Iodine has high volatility at relatively low temperature and it is not easy to suppress some vaporization during conventional consolidation process like vitrification. Some noble works are under development for stable immobilization of iodine with much lower vaporization during the consolidation. At this moment, iodine filter is destined to be consolidated by HIP process. Other gaseous FPs has low amount and their activity is not considerable. Cementation or a specific storage system for them is well defined as a commercial level.

Waste salt is a main problematic waste in a pyro-chemical process for a recovery of U/TRUs from a used fuel. In the pyroprocessing, two kinds of waste salts are generated, LiCl waste from an oxide reduction process and LiCl-KCl waste from an electro-refining/winning process. There are two approach concepts on the waste salt treatment. One is a direct immobilization of waste salt without purification. It is required to secure the stability of pyro-system that generates waste salt being difficult to apply to a conventional consolidation process. In order to reduce the waste volume for final disposal, U-SAP(SiO2-Al2O3-P2O5-Fe2O3-B2O3) is used as a de-halogenation and consolidation material. By this matrix, metal element in waste salt is immobilized into a silicate-phosphate glass matrix and halogen (especially chlorine) is captured with alkali metal oxides to produce a metal chloride compound that return to use as an electrolyte in an electrolytic process. The other is the purification for minimization of waste salt for final disposal. The purification of waste salt is performed by a melt crystallization using a liquid-solid phase-equilibrium and a reactive distillation using a liquid-gas phase-equilibrium. Sr(Ba) in LiCl waste and lanthanide group in LiCl-KCl waste have relatively lower vapor pressure than LiCl and KCl. In this case, a selective distillation with some chemical agent at proper process condition can separate a pure LiCl or LiCl-KCl salt for reuse in a electrolytic process and Sr(Ba) or lanthanide oxides for final disposal by vitrification. In case that the amounts of some volatile FPs like Cs, Se or Te in waste salt is not negligible, the melt crystallization is very effective to concentrate the FPs in salt. From this process, a salt concentrated with FPs is generated and it is immobilized by U-SAP material. The cleaned salts with relatively lower amount of FPs return to an electrolytic process for reuse.

In waste treatment in the pyroprocessing, the most important issue is to regenerate waste salt for minimizing waste salt. A challenging issue is to develop a metal alloy with hull waste for final disposal and a recycling process for Zr source in the hull that has the largest amount in the wastes generated from the recovery process for U/TRUs.

7 Other risks

7.1 Chemical risk

As described in 4.3, SFRs have to take care sodium-air and sodium-water reaction. Then the GIF-SFR-SDC/SDG requires as measures and evaluations to show integrity of containment[3]. It requires that design extension conditions shall include potential significant sodium chemical reactions (e.g. combustion resulting from leakage, sodium-water reaction resulting from steam generator tube failure, and sodium-concrete interactions resulting from leakage) so as to avoid affecting the safety of the reactor core or loss of containment function. The capability of ensuring containment integrity will be required for design extension conditions. Therefore, containment will be required to withstand thermal and mechanical loads generated during the event transient. Sodium combustion, sodium concrete reaction, debris-concrete interaction, and combustion of accumulated hydrogen, which have the potential to load or otherwise threaten the integrity of the containment, must be prevented or mitigated.

During sodium leak accident, following hazards have to be taken into account:

- Skin burn due to metallic sodium contact
- Alkaline burn due to sodium hydro-oxide
- Respiratory system burn due to sodium aerosol

Experiences on human protection against sodium combustion have been and are being accumulated through past and existing sodium facilities and reactors.

7.2 Radiation protection

As discussed in 4.3.1, major sources of radio-active material without fuel failure are radio-activation of coolant and CP from core components and structural material. For 24Na and 22Na halving times are 15hours and 2.6years, respectively. From radiation protection point of view, 22Na is important since it has longer halving time. For CP, 58Co, 60Co and 54Ma are important, since they also have long halving time.

8 Others

9 Summary of progress needed

9.1 Reactivity control

Fast reactors including SFR have a characteristic that a core is not in the most reactive configuration, and thus they have possibility to result in positive reactivity changes when assuming coolant boiling, clad discharge and fuel concentration resulting from a core disruptive accident (CDA). The GIF SFR SDC/SDG requires following items:

- Two diverse and independent active reactor shutdown systems
- Passive shutdown systems such as SASS, HSR and GEM
- Design measures preventing prompt criticality, potentially leading to large mechanical energy release during core degradation

And additional to them, design tools to evaluate performances of those measures are also have to be improved.

9.2 Decay heat removal

Timely unlimited failure of decay heat removal function has to be “practically eliminated” because this leads in particular, to whole core damage combined with damage of the in-vessel and potentially containment structures. This requires R&D:

- “Practical elimination” of unacceptable primary sodium draining, which is assessed considering the risks associated to the loss of primary circulation through intermediate heat exchangers, the DHR heat exchangers uncovering, the core uncovering.
- To assure a sufficient sodium circulation through the core in particular, in natural convection.
- “Practical elimination” of the failure of all systems needed for decay heat removal. In particular, this requires assessing potential common cause failures.

9.3 Confinement challenge

Within the current R&D phase, the possibility to robustly mitigate consequences of whole core accident has to be investigated. Therefore, a special design R&D has to be performed regarding:

- The investigation on core designs, and complementary safety features, allowing minimization of mechanical energy releases: the risk limitation for unacceptable core criticality potentially resulting from sodium voiding and core melting and the risk of energetic interaction between molten fuel and sodium.
- The research of a robust core-catcher design and associated decay heat removal capability. Attention should notably be paid to ensure that the availability of these devices is not threatened by a possible mechanical energy releases and/or by unacceptable increase of load carrying temperatures.
- The research of a robust confinement capability including assessment of possible loads, weak points due to potential by-pass of the confinement structures and accident management capability at long term.

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