GIF/RSWG/2023/002

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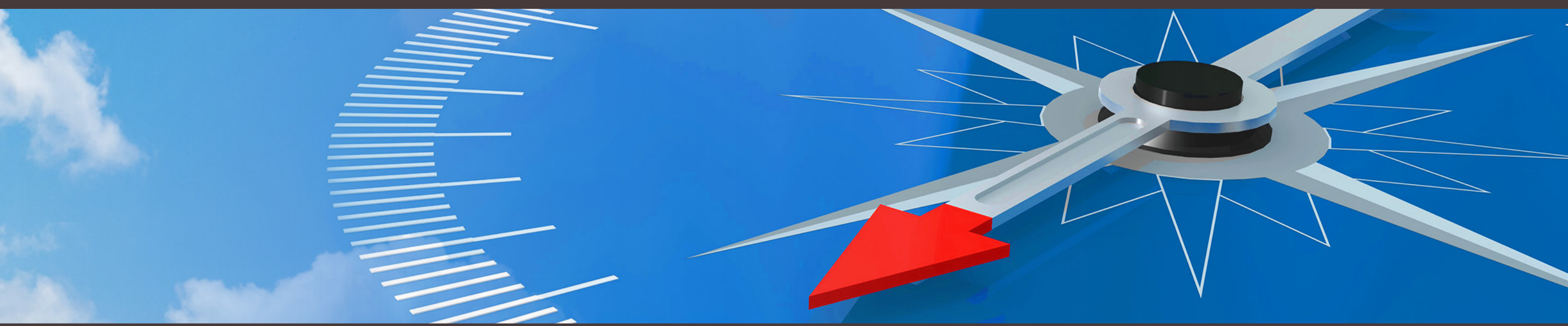
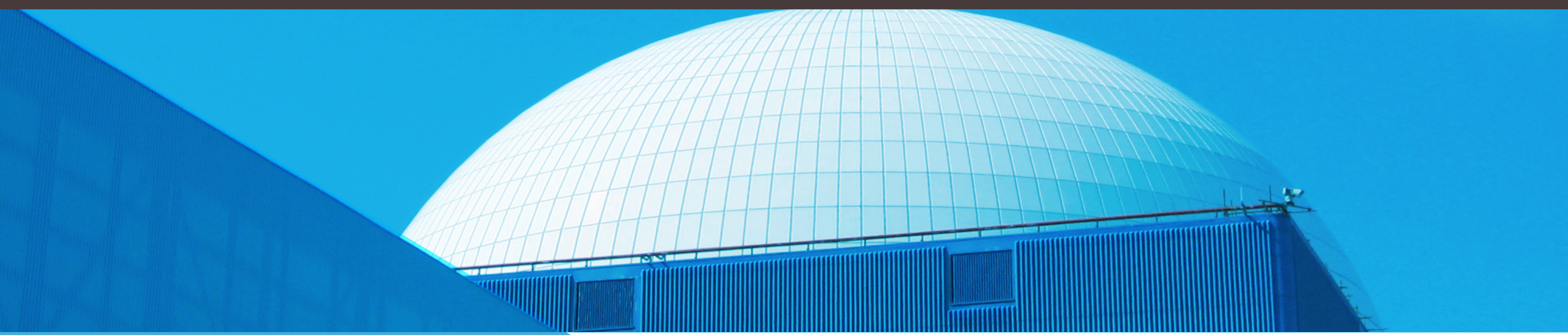
SAFETY DESIGN GUIDELINES ON STRUCTURES, SYSTEMS AND COMPONENTS FOR GENERATION IV SODIUM-COOLED FAST REACTOR SYSTEMS

March 2024





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1. INTRODUCTION

1.1. Background and Objectives

Generation IV International Forum (GIF), under the auspices of the Policy Group (PG), established Safety Design Criteria Task Force (SDC TF) for Generation IV (Gen-IV) sodium-cooled fast reactors (SFRs), and the SDC TF developed Safety Design Criteria Report (SDC) for SFRs in May 2013 (revised in September 2017) [1-1]. To implement the SFR SDC on SFRs, Safety Approach Safety Design Guidelines (SDG) [1-2] was developed in March 2016 (revised in August 2019) for the clarification of the requirements important to the safety. The Safety Approach SDG provides recommendations on measures to prevent accidents and mitigate their consequences and on design measures for practical elimination. It also shows technical considerations such that an SFR core is not in its most reactive configurations for safety. The SDG is placed below the SDC in the hierarchy of safety standards as depicted in Figure 1. Following approval by the PG, the SDC Report and the Safety Approach SDG were distributed to international organisations and national regulatory bodies for review.

Subsequently, the SDC TF has started developing SDG for systems, structures and components (SSC) (SSC-SDG), in order to cover the safety design recommendations dealing with all of plant states, i.e., normal operation, anticipated operational occurrences (AOOs), design basis accidents (DBAs) and design extension conditions (DECs), more comprehensive manner. This guideline makes clear connection between the recommendations in the Safety Approach SDG and each SSC design. In addition, SSC-SDG describes recommendations to the requirements of the SDC Report that were not mentioned in detail in the Safety Approach SDG. The recommendations in this guideline include measures against Anticipated Transient Without Scram (ATWS) for reactivity characteristics, and the measures for practical elimination of core uncovering and complete loss of decay heat removal function. The recommendations which have been out of the scope of the Safety Approach SDG include those for fuels and materials under high temperature and radiation conditions, reactor component design under high temperature and low pressure conditions, measures against various hazards such as sodium fire, sodium-water reaction, and loading on the containment system. Recommendations for addressing these issues are provided in the chapters with guidelines for reactor core, reactor coolant and containment systems. Figure 2 shows the consideration process for the SSC-SDG. The objective of the SSC-SDG is to provide detailed guidelines for SFR designers to support the practical application of the SDC in design process to ensure the highest level of safety in SFR design. Note that this SSC-SDG focusses on the reactor and, therefore, excludes consideration of out-reactor fuel handling and fuel storage. These issues may be addressed in the future with extended SDG.

The SSC-SDG shows recommendations and guidance to comply with the SDC Report and the Safety Approach SDG with examples, which can be applied to Gen-IV SFR systems in general.

Designers don’t need to cover all of statements in the SSC-SDG in their own design. Because the SSC-SDG intends to show concrete design solutions based on design practice among GIF SFR member states, it includes different design options that may not always be applicable to all designs.

The future SFR design depends on the designers’ choice, applicable national regulation and so on. Risk and Safety Working Group safety documents provide the safety basis and approaches for the future SFR safety design [1-3 to 1-6]. The GIF SDC TF expects that these recommendations and examples will be appropriately considered in design according to each design characteristic. Although some of the recommendations in this document are expressed as ‘should’ statements (to emphasize their importance), the future SFR designs will obviously depend on the designers’ choices and applicable sovereign regulations.

1.2. Scope and Structure

The SSC-SDG describes recommendations on the three fundamental safety systems of Gen-IV SFRs, namely, reactor core, reactor coolant system and associated systems, and containment system and associated systems. Table 1 shows the SFR-specific safety features for each system; the selected 14 focal points are also described in this document. The recommendations on specific SSC are developed for accounting features of the GIF SFR systems, with referring IAEA SSG series [1-7 to 1-9] on descriptions, definitions, and formats.

This guideline covers the recommendations of the Safety Approach SDG, i) to equip active and passive reactivity reduction mechanisms, ii) to take preventive measures against significant energy release in a severe accident case, iii) to have design measures to maintain the reactor coolant level and coolability of the core, iv) to utilize the natural circulation of sodium, and v) to ensure its reliability for decay heat removal from the core. In addition, SFR-specific measures against leakage and combustion of sodium and sodium-water reaction are included as internal hazard countermeasures. Designs of fuels and components of the reactor coolant system and associated systems to withstand the high temperature conditions are also addressed in association with the matters that are currently being developed in GIF’s SFR projects.

Examples of design measures proposed by GIF SFR System Steering Committee member countries for the 14 focal points are presented. Common ones are described in the main text, while design concepts that can differ from one country to another are collected in Appendix or Annexes, or noted in footnotes, as defined in the IAEA SSG series “*An appendix, if included,* *is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text*". Individual examples with design concept drawings are shown in Annex, also same as the IAEA SSG series "*Annexes and footnotes are not integral parts of the main text. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful*".

The contents of the SSC-SDG are grouped into the following four parts.

- Chapter 1, Introduction, describes the background and objectives together with the scope and structure of the SSC-SDG.
- Chapter 2, Guidelines for reactor core system, includes recommendations on fuel elements and fuel assemblies for the integrity maintenance of the reactor core system, active reactor shutdown system, and reactor shutdown under a DEC with reactivity control.
- Chapter 3, Guidelines for reactor coolant system and associated systems, provides recommendations on component design and reactor cover gas and its boundary, coolant level maintenance, and measures against sodium leakage and combustion in the primary coolant system. Fundamental functions, decay heat removal under a DBA, and decay heat removal under a DEC are presented to give recommendations on the decay heat removal systems. Application of natural circulation and safety considerations of tests and inspections are provided. Measures against sodium leakage and combustion along with measures against sodium-water reaction in the secondary coolant system are also described.
- Chapter 4, Guidelines for containment system and associated systems, presents their safety functions, general design basis, and design of containment system and associated systems against accident conditions. Tests and inspections for the whole system are included in this chapter. Topics regarding the confinement function of the secondary coolant system, which is one of the SFR characteristics, under an accident condition are also shown in this chapter.

References

[1-1] Generation IV International Forum, "Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System (Rev.1)," SDC-TF/2017/02 (30 Sept. 2017).

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[1-3] Generation IV International Forum, "Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems," Revision 2, GIF/RSWG/2021/001, 2021 July.

[1-4] Generation IV International Forum, "An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems," GIF/RSWG/ISAM Report Version 1.1, 2011 June.

[1-5] Generation IV International Forum, "Guidance Document for Integrated Safety Assessment Methodology (ISAM) (GDI)," GIF/RSWG/2014/001

[1-6] Generation IV International Forum, "A Risk-Informed Framework for Safety Design of Generation IV Systems," GIF/RSWG/2023/001, 2023 June.

[1-7] IAEA, "Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants," SSG-56 (2020).

[1-8] IAEA, "Design of Reactor Containment and Associated Systems for Nuclear Power Plants," SSG-53(2019).

[1-9] IAEA, "Design of the Reactor Core for Nuclear Power Plants," SSG-52 (2019).

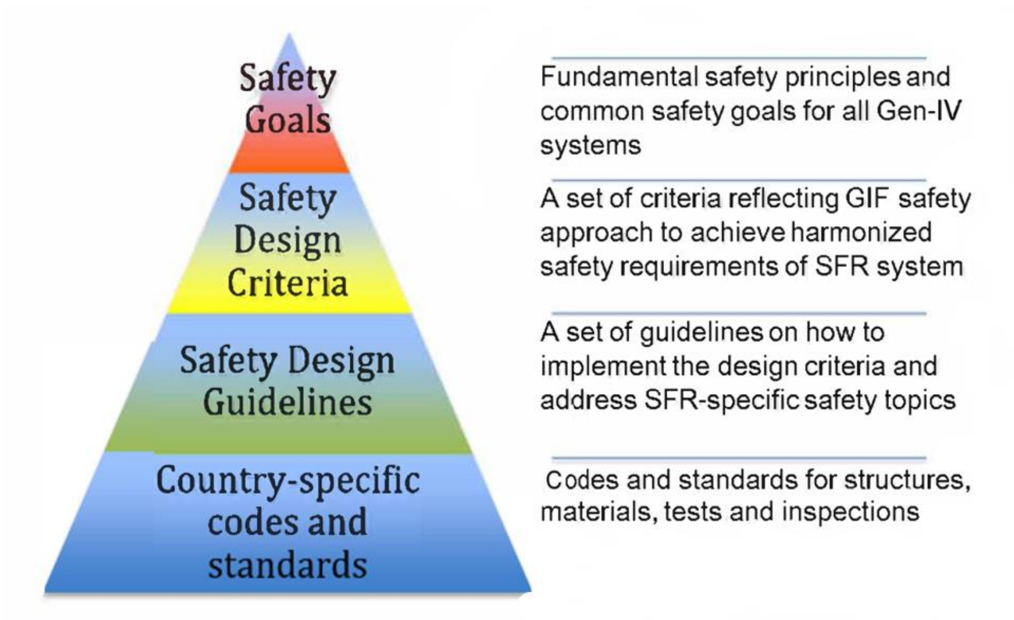

Figure 1 Hierarchy of safety criteria

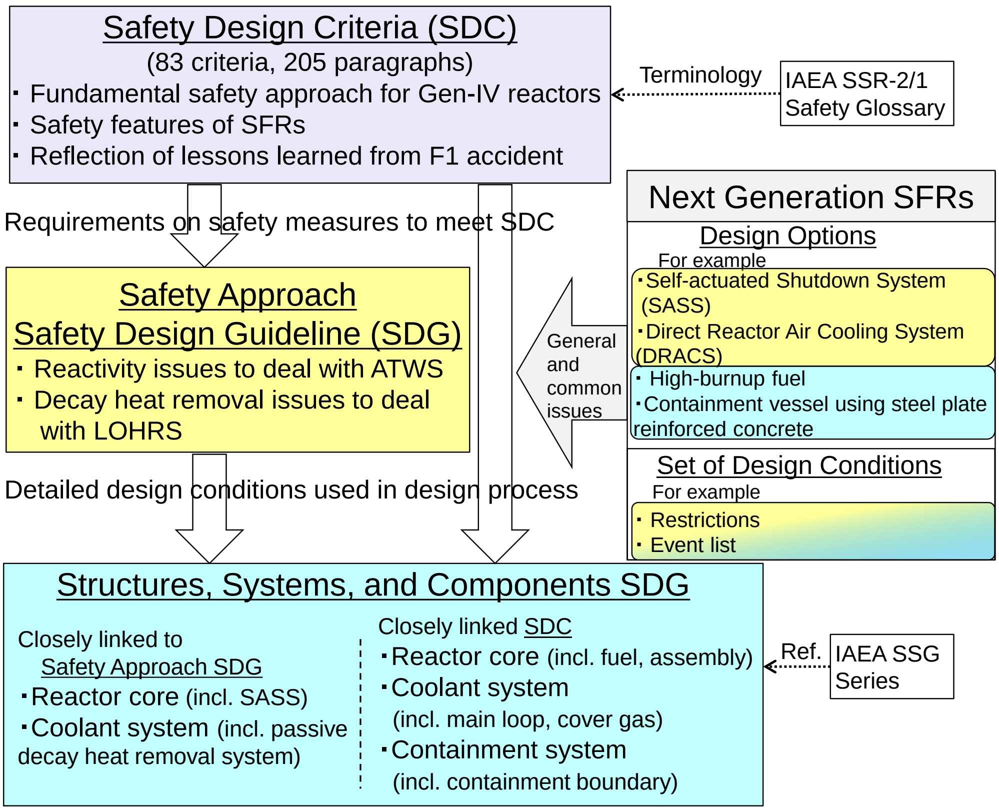

Figure 2 Consideration process of the SSC-SDG[1]

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[1] *"F1" in this document refers to the accident at the Fukushima Daiichi ("number one") Nuclear Power Plant following the Tōhoku earthquake and tsunami of March 2011.*

Table 1 14 focal points in the SSC-SDG[2]

Systems	Categories	Issues	SDC	Safety approach SDG	SDG statement's number
Reactor Core	Integrity maintenance of core fuels	1. Fuel design to withstand high temperature, high inner pressure, and high radiation conditions	✓		2.1, 2.2, 2.4～2.15
		2. Core design to keep the core coolability	✓	✓	2.18～2.23, 2.28～2.34, 2.43～2.46
	Reactivity control	3. Active reactor shutdown	✓	✓	2.47～2.59
		4. Reactor shutdown using inherent reactivity feedback and passive mechanisms	✓	✓	2.16, 2.17, 2.24～2.29, 2.32～2.35, 2.60～2.75
		5. Prevention of significant energy release during a core damage accident, In-Vessel Retention	✓	✓	2.32～2.37, 2.44～2.46
Reactor Coolant System and associated systems	Integrity maintenance of components	6. Component design to withstand high temperature and low pressure conditions	✓		3.1～3.16, 3.22～3.26, 3.33～3.35, 3.49, 3.52～3.54, 3.56～3.71
	Primary coolant system	7. Cover gas and its boundary	✓		3.18, 3.72～3.81
		8. Measures to keep the reactor coolant level	✓	✓	3.3, 3.9, 3.48, 3.62, 3.68, 3.70, 3.82～3.97
	Measures for prevention and mitigation of sodium chemical reaction	9. Sodium leakage and combustion	✓		3.17, 3.33～3.36, 3.48～3.50, 3.110, 3.133～3.146
		10. Sodium-water reaction	✓		3.36, 3.147～3.155
	Decay heat removal	11. Securing decay heat removal by natural circulation of sodium	✓	✓	3.1, 3.3, 3.9, 3.39, 3.123～3.128, 3.132
		12. Reliability maintenance (diversity and redundancy)	✓	✓	3.11, 3.22～3.32, 3.102, 3.103, 3.105～3.109, 3.111～3.122, 3.129～3.131
Containment and its associated systems	Design concept and loading	13. Formation of containment boundary and loads on it	✓		4.1～4.39, 4.49～4.52, 4.57～4.59
	Containment boundary	14. Containment function of secondary coolant system	✓		4.13(d), 4.53～4.56

[2] *The role of the retaining radionuclides is included in focal point1 "Fuel design to withstand high temperature, high inner pressure, and high radiation conditions" and focal point 6 " Component design to withstand high temperature and low pressure conditions". Sodium freezing is included in "Integrity maintenance of components" and "Decay heat removal" as the key issue for long-term cooling of the reactor core after reactor shutdown.*

2. GUIDELINES FOR REACTOR CORE

2.1. Integrity Maintenance of Reactor Core Fuels

2.1.1. Fuel elements and fuel assemblies

(1) General safety considerations in design

2.1 Design limits on parameters such as the maximum linear heat generation rate, the peak fuel temperature and the cladding temperature should be set in such a way that there are sufficient margins in operational states (i.e., normal operation and AOO) to prevent the cladding failure and to keep the failure rates of fuel elements acceptably low level under DBA conditions.

2.2 The design should ensure that the structure of fuel assemblies (i.e., their geometry) and integrity of fuel elements (i.e., their ability to retain radionuclides) is maintained in the aforementioned operational states throughout its lifetime.

2.3 Fuel elements and fuel assemblies should be designed to withstand handling loads during transport, storage, installation and refuelling operations.

(2) Specific safety considerations in design

<u>(2-1) Thermal and burnup effects</u>

2.4 During operational states (i.e., normal operation and AOOs) including erroneous control rod withdrawal, the peak fuel temperature should be lower than the fuel melting temperature by a sufficient margin, with allowance for uncertainties including manufacturing tolerances, to prevent any fuel melting. In the evaluation of the temperatures of fuel in operational states, account should be taken of the changes in the thermal conductivity of the fuel and of the dimensional changes of fuel and cladding due to irradiation effects. In determining the melting temperature of fuel, the changes in the composition and microstructure of the fuel due to burnup effects should be taken into account.

2.5 Stress and strain of the cladding caused by the swelling or thermal expansion of the fuel or by an increase in the internal gas pressure should be limited so that the integrity of the fuel element cladding is not compromised during operational states. The margin-to-failure assessments for the cladding should account for all effects that may occur during irradiation, including any fuel-cladding mechanical and chemical interactions, increases in the internal pressure, and changes in cladding mechanical properties (strength, creep and stress relaxation). During operational states, the limits for stress and strain, including that caused by increased fission gas pressure and differential swelling and thermal expansion of the fuel and cladding, should not be exceeded.

2.6 In the design of fuel elements, account should be taken of the effects of solid and gaseous fission products, the production rates of which depend largely on the power history during their in-core residence. The effects of gaseous fission products on the internal pressure of a fuel element should be limited to keep creep damage of the cladding within an acceptable range. The effects of fission products on the thermal conductance of the fuel-to-cladding gap should also be included, if appropriate. Additionally, the chemical reaction of fission products with the cladding and the potential inner corrosion of the cladding should be considered in the design. Swelling of the fuel material as a consequence of the formation of fission products should be taken into account in the design to avoid the excessive fuel-cladding mechanical interaction. If the fuel pellet fragmentation and relocation are expected to occur, their effects on the cladding integrity should be evaluated.

(2-2) Effects of irradiation

2.7 The effects of irradiation, in particular, the effects of fast neutrons on fuel assemblies, on metallurgical properties such as the tensile strength of the cladding, ductility and creep behavior, fuel dimensional changes (in radial and axial directions), and on the geometrical stability of all materials should be considered in the design.

(2-3) Effects of variations of power levels

2.8 Account should be taken in the design of the effects on the integrity of the cladding of local and global power transients due to fuel shuffling, fresh fuel loading, movements of control rods or other reactivity changes.

2.9 The power distribution in the core and the fuel assemblies changes during the fuel cycle owing to the burnup of fuel. Accordingly, the excess reactivity of the core and the reactivity coefficients of the core also change. These phenomena should be taken into account in the design of the core and the fuel.

(2-4) Corrosion of fuel assemblies

2.10 Fuel assemblies should be designed to be compatible with the coolant environment in all operational states, including shutdown and refuelling, and in the storage. Corrosion depends on the material properties of the cladding and wrapper tube, and on an environmental condition. The environmental conditions for liquid coolant, such as conditions of sodium purity, temperature should be taken into account. In practice, corrosion is controlled by means of appropriate sodium chemistry (e.g., maintaining a low oxygen content by removing impurities with a cold trap).

(2-5) Thermal-hydraulic effects in fuel assemblies

2.11 In normal operation, steady state power and coolant flow should be maintained at levels that allow for certain margin considering core temperature distribution, to avoid coolant saturated boiling during the DBAs. Provisions should be made in the design to prevent or limit changes in fuel element spacing so that thermal-hydraulic behavior and fuel performance are not significantly affected. Thermal-hydraulic effects that depend on the fuel element spacing by the wire-wrap or grid spacers, the fuel element power, and coolant flow rate should be taken into account to avoid flow-induced vibration, fretting or any other conditions that could degrade coolant flow, impact reactor power, or compromise integrity of fuel pins. The design should also account for the effects of power gradients between and within fuel assemblies, power shifts due to burnup, fuel shuffling, movements of control assemblies, or other reactivity changes, on the integrity of the fuel elements and assemblies.

2.12 The fuel assembly should remain in an adequate position on the core support structure to keep the adequate coolant flow in the fuel assembly in consideration of coolant hydraulic force even in case of unexpected increase of primary coolant flow.

(2-6) Considerations of mechanical safety in the design

2.13 The fuel assembly should be designed to withstand mechanical stresses as a result of the following possible examples:

(a) Fuel handling and loading;
(b) Power variations;
(c) Temperature gradients;
(d) Hydraulic forces;
(e) Irradiation effects (e.g., irradiation induced growth and swelling);
(f) Vibration induced by coolant flow and fretting wear of fuel rods;
(g) Creep deformation of the fuel assembly structure (which could lead to distortion of fuel assemblies);
(h) Seismic loading;
(i) Postulated initiating events (i.e., AOOs and DBA conditions) and design extension conditions without significant fuel degradation.

2.14 For normal operation and AOOs, the design considerations for the fuel assembly include the following:

(a) The clearance within and adjacent to the fuel assembly should provide space to allow for irradiation swelling and creep deformation such that sufficient clearance remains to permit fuel assembly removal;

(b) Fatigue should not cause the failure of a fuel assembly;

(c) The fuel assembly should be able to withstand the mechanical forces, which may be caused as a result of fuel handling and loading, power variations, thermal gradients, hydraulic forces including the pressure difference between inside and outside of an assembly duct, without unacceptable deformation;

(d) The performance of the functions of the fuel assembly and the support structure should not be unacceptably affected by damage due to cavitation, vibration or fretting induced by coolant flow;

(e) The fuel assembly should be able to withstand irradiation and its materials should be compatible with the coolant conditions;

(f) Any deformation of the fuel element or the fuel assembly, which could affect the capability for the insertion of control rods for the safe shutdown of the reactor, should be avoided in all operational states, DBA, and DECs without significant fuel degradation (see criterion 44 of SDC).

(2-7) Provision for inspection and testing

2.15 Provision should be made for the inspection of fuel assemblies before and after irradiation. The manner and frequency of inspection will be established by operational and regulatory requirements.

2.1.2. Reactor core[3]

(1) General safety considerations in design

(1-1) Neutronic design

2.16 The design of the reactor core should be such that the feedback characteristics of the core rapidly compensate for an increase in reactivity. The reactor power should be controlled by a combination of the inherent neutronic characteristics of the reactor core, its thermal-hydraulic characteristics, and the capability of the control and shutdown systems to actuate for all applicable plant states. When rapid-acting control or shutdown systems are necessary, their capabilities (e.g., speed and reliability) should be assured.

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[3] *The design guideline for the fuel elements and fuel assemblies, which are part of the core, is elaborated in Section 2.1.1.*

2.17 The maximum degree of positive reactivity and its insertion rate in operational states and accident conditions not involving degradation of the reactor core should be limited or compensated to prevent any resultant failure of the boundary of the primary coolant system, to maintain the capability for cooling and to prevent any significant degradation of the reactor core.

(1-2) Mechanical design

2.18 Structures and components of the reactor core should be designed, fabricated, erected, constructed, tested, and inspected in accordance with codes and standards appropriate to SFR design specificities or approved corresponding quality management system, commensurate with the significance of the safety functions.

2.19 The structural integrity of the core should be ensured so that the core can be safely controlled, shut down and cooled for operational states, DBAs and DECs without significant fuel degradation under various damage mechanisms caused by, for example: vibration (mechanical vibration or flow induced vibration) and fatigue; debris effects; thermal, hydraulic and mechanical loads (e.g., seismic events); and chemical and irradiation effects (including radiation induced growth).

2.20 The core assemblies (fuel assembly and other assemblies such as control rod assembly, shielding assembly and blanket fuel assembly) and its associated components should be designed to be compatible under the effects of irradiation and chemical and physical interactions, e.g., avoid excessive assembly withdrawal forces for refuelling by allowance for radiation-induced creep and swelling.

2.21 The gap between adjacent core assemblies should be designed to have large enough space to allow refuelling before and after irradiation without causing excessive fuel assembly withdrawal and insertion forces.

2.22 In plant states ranging from normal operation to DECs without significant fuel degradation, the design should prevent any interaction between fuel elements, fuel assemblies, and fuel assembly support structures that would impede safety systems from performing their function or inhibit the proper cooling of the core.

(1-3) Coolant

2.23 Safety considerations associated with the coolant in the core should include

(a) Detecting and removing foreign objects, fluids and debris from the coolant system prior to the initial startup of the reactor and during the operating lifetime of the plant;

(b) Keeping the activity of the coolant at an acceptably low level by means of purification systems and the removal of defective fuel as appropriate;

(c) Ensuring a sufficient inventory of coolant for operational states, DBAs, and DECs without significant fuel degradation;

(d) Ensuring that the core is designed to prevent flow instabilities and cavitation within the core and the consequent fluctuations in cooling and reactivity;

(e) Purifying the primary coolant to remove chemical impurities that may cause corrosion of SSCs; and

(f) Suppressing contamination of the coolant with foreign objects, fluids and debris that may cause coolant flow blockage or deterioration of fuel assemblies and its associated components.

(1-4) Core reactivity characteristics and means of control of reactivity

2.24 On the basis of the geometry and the fuel composition of the reactor core, the nuclear evaluations for design should provide steady state spatial distributions of neutron flux and power, core neutronic characteristics and efficiency of the means of reactivity control for normal operation at power and at shutdown conditions.

2.25 The control rod assembly and its support structure should be designed so that the displacement caused by hydraulic forces such as flow-induced vibration and seismic loads is within the specified limit value.

2.26 Adequately conservative assumptions should be made for reactivity coefficients in the analysis of all DBAs and AOOs. Best-estimate assumptions should be made for reactivity coefficients in the analysis of DECs.

2.27 Key reactivity parameters such as reactivity coefficients should be evaluated for each core state and for the corresponding strategy for fuel management. Their dependence on the core loading and the burnup of fuel should be taken into account. Design should account for excess reactivity and reactivity insertion rates to ensure adequate margins for safety.

(2) Specific safety considerations in design

(2-1) Reactor core support structures

2.28 The core support structures should be designed to array and support the fuel assembly in the desired geometrical position to ensure control device insertion and to prevent excessive reactivity changes, e.g., core restraint concepts such as the "limited-free-bow" design that utilizes assembly load pads, core restraint rings, etc., in operational states DBAs, and DECs without significant fuel degradation.

2.29 The design should provide the necessary flow rate to core components for the core thermal-hydraulic design in operational states, DBAs, and DECs without significant fuel degradation.

(2-2) Prevention of flow blockage

2.30 Equipment of the primary circuit should be designed to avoid the generation of loose parts that may cause obstruction of the coolant flow.

2.31 Coolant flow paths to the core fuel assembly should be designed to prevent any obstruction of the coolant flow due to the release of loose parts or structure, or the presence of foreign solid or fibrous material, so as to prevent core damage in operational states, DBAs, and DECs without significant fuel degradation, by using multiple flow channels to allow inflow of the coolant from different positions and directions (e.g., multiple circumferential inlet openings to the assembly) to preclude total blockage of the assembly flow caused by any loose structure. Other core components also should have such design to prevent the blockage.

(2-3) Practical elimination of severe reactivity insertion during normal operation

2.32 For practical elimination of large-scale core compaction, which may cause excessive positive reactivity insertion leading to prompt criticality, the design should

(a) Provide adequate structural stiffness of the core fuel assemblies through the use of suitable materials for the core assemblies. Core support plates and the core restraint system should be ensured;

(b) Provide appropriately small gap between adjacent core assemblies to allow the core assembly positions to be solidly established along the entire assembly length during operation, which is a part of the function of the core restraint system;

(c) Provide core restraint system to limit any fuel assembly motion and deformation in response to potential events, e.g., earthquakes, to prevent uncontrolled reactivity changes and subsequent power changes; and

(d) Ensure control rod insertion with sufficient reactivity margin even under a severe earthquake. In order to maintain subcritical conditions after control rod insertion, the upward movement of the inserted control rods should be limited or prevented.

2.33 For practical elimination of collapse of the core support structure, which may cause unmanageable massive simultaneous control rod withdrawal since the control rods are suspended from the reactor roof in general, the core support structure should be designed to ensure sufficient design margin against mechanical and thermal loads, and should incorporate structural tolerance against potential flaws Radiation dose to the core support structure and temperature around the core support structure should be limited to the level at which those influences on the core support structure are acceptable. Detection of potential core support deformation or failure should be provided in order to confirm through life integrity of the core support structure.

2.34 For practical elimination of prompt criticality by excessive reactivity insertion due to ingress of coolant with a large gas volume fraction into the core, any potential for gas accumulation upstream of the core in the coolant flow should be limited so as to prevent the large gas bubble formation. Therefore, the core support structure should be designed to reduce the accumulation and, if necessary, to have gas release paths.

(2-4) Application of In-Vessel Retention

The following recommendations are design measures for the in-vessel retention, if necessary.

2.35 For the in-vessel retention, the following should be considered in the initiating phase (i.e., accident phase from intact state up to inter-subassembly material motion onset by subassembly duct failure) of core degradation from an unprotected transient with core damage according to descriptions of the Safety Approach SDG below:

(a) Limiting the total reactivity during unprotected transients

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 should be limited so that negative reactivity effects are sufficient to counteract the positive reactivity effects. Relevant reactivity effects depend on the design and accident conditions. Design parameters such as sodium volume fraction, core height and other geometric parameters should be based on the relative importance of relevant reactivity effects to the overall net reactivity during transient.,

(b) Facilitating fuel reactivity effects

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.

2.36 For the in-vessel retention, the following should be considered in the transition phase (i.e., an accident phase after initiating phase up to the establishment of stable cooling conditions) of core degradation from an ATWS according to descriptions of the Safety Approach SDG below:

(a) Limiting the total reactivity during unprotected transients

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 consider the use of inherent phenomena occurring in the course of core degradation[4].

(b) Establishment of a stable cooling condition

Measures should be provided to establish a stable cooling condition of a degraded core. Due consideration should be taken to the coolability of the remaining fuel inside the core region and any relocated molten core materials. Prompt criticality, potentially leading to large mechanical energy release, should be prevented during the relocation process.

2.37 For the situation of core degradation, means to diagnose the conditions of the reactor core, i.e., criticality and cooling conditions during core damage sequences should be provided. For instance, criticality can be judged by monitoring neutron flux, and core cooling condition can be determined by coolant temperature and coolant level.

(2-5) Core management

2.38 The objective of core management is to fulfil the requirements for the safety of the reactor core and the economic utilization of the nuclear fuel.

2.39 A fuel cycle should be selected with appropriate levels of enrichment and appropriate means of controlling the core reactivity and the power distribution so as to extract energy from the fuel in the most economic manner within the safety limitations.

2.40 The specified design limits for normal operation should be taken into account in the design for core management.

2.41 Means should be provided to prevent misloading of fuel assemblies and other core components that may cause fuel failure and radioactive release. For example, a design to change a fitting shape of the entrance nozzle in regions with different fissile enrichment.

2.42 For operational states, the goal is that no cladding failures should occur. However, certain conditions (e.g., manufacturing defects in fuel elements, wear due to debris fretting) may make it extremely difficult to meet this no-failure goal. In practice, some fuel cladding failures can be expected in operational states, and the design should provide means to reduce the concentration of radioactive material in the

[4] *For instance, providing a molten fuel discharge path is an effective measure in oxide fuel cores. The design of a molten fuel discharge path should prevent blockage due to freezing of relocated molten cladding and/or fuel, should be accessible prior to formation of large amounts of molten fuel, and should have enough capacity for timely discharge of molten fuel.*

primary coolant system and cover gas. Cleanup of the reactor coolant and cover gas and other necessary means should be provided in the design to ensure that releases of radioactive material to the environment remain within limits authorized by the responsible regulatory body.

(2-6) Core monitoring system

2.43 Instrumentation should be provided for monitoring the core parameters such as the core power (level and time dependent variation), the conditions and physical properties of the coolant (flow rate, temperature), to ensure that design limits are not exceeded.

(2-7) Detection of fuel failure

2.44 Failure detectors should be provided for, e.g., local fuel pin failure caused by coolant channel blockage in the fuel assembly and incidental fuel pin failure[5].

2.45 For the situation that a failed fuel pin is identified as a cause of the radioactivity levels in the primary coolant and cover gas reaching a permitted level of the operational states, a measure to remove a failed fuel assembly should be provided.

2.46 If fuel pin failure is detected and the signal level of fuel failure detectors exceeds the permitted level of the operational states, reactor shutdown and cooling should be adequately implemented to prevent possible failure propagation. If rapid failure propagation to adjacent fuel pins or assemblies are foreseen, automatic reactor shutdown should be triggered to ensure coolability of the failed fuel.

2.2. Reactivity Control

2.2.1. Active reactor shutdown system

(1) General safety considerations in design

2.47 Means should be provided to ensure that the reactor can be rendered subcritical and held in this state.

(a) The means for shutting down the reactor should consist of at least two different, independent systems to provide diversity and redundancy. At least one of the two systems should be, on its own, capable of quickly rendering the nuclear reactor subcritical and capable of maintaining the safe shutdown state[6] by an adequate

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[5] *e.g., 1) Detector of gaseous fission products released from a failed fuel pin to cover gas; 2) detector of delayed neutrons emitted from delayed neutron precursors released from a failed fuel pin.*

[6] *Safe shutdown state is defined as the state with the reactivity of the reactor kept to a margin below criticality, even for the most reactive conditions of the core, under a prescribed coolant temperature*

margin and with high reliability, even for the most reactive conditions of the core, from operational states, DBAs and DECs without significant fuel degradation, on the assumption of one control rod stuck (the control rod that has the highest reactivity worth cannot be inserted into the core).

(b) The fuel and SSCs (such as the primary coolant boundary and core support structure) design limits, such as peak temperature or peak temperature and duration, should be provided as a set of design conditions to ensure that the shutdown systems provide adequate protection of the reactor.

(c) The first reactor shutdown system should be designed such that the fuel and SSCs design limits of AOO are not exceeded during an AOO, and the fuel and SSCs design limits of DBA are not exceeded during a DBA. The second reactor shutdown systems should be designed at least such that the fuel and SSCs design limits of a DBA are not exceeded during an AOO with assuming that the first reactor shutdown system is not actuated, and the fuel and SSCs design limits of a DEC are not exceeded during a DBA with assuming that the first reactor shutdown system is not actuated.

(d) The system used for reaching a safe shutdown state should be safety classified. The system used for shutdown may also be used as a reactivity control system in normal operation, but such a use in normal operation should not jeopardize the system functioning as a shutdown system.

(2) Specific safety considerations in design

<u>(2-1) Reliability</u>

2.48 The design should include the following measures to achieve a high reliability of shutdown by means of each the following measures, or a combination of these as appropriate:

(a) Adopting systems that are as simple as possible and using a fail-safe design as far as practicable.

(b) Giving consideration to the possible modes of failure of the shutdown systems and adopting redundancy in the activation of shutdown systems (e.g., sensors or actuation devices). Subcriticality should be ensured if a single random failure occurs in the shutdown system. Providing diversity for any part of the systems as far as practicable, for example, by using different physical trip parameters for the

condition in which interventions such as fuel reloading, periodic inspection and repair works in the reactor can be achievable.

different reactor shutdown systems, or two different physical trip parameters for each accident.

(c) Functionally isolating and physically separating the shutdown systems (this includes the separation of control and shutdown functions) as far as practicable, on the assumption of credible modes of failure, including common cause failure.

(d) Ensuring easy entry of the means of shutdown into the core, accounting for the in-core environmental effects of operational states and DBAs.

(e) Designing to facilitate maintenance, in-service inspection and operational testability giving consideration to the issues that sodium is optically opaque and chemically reactive.

(f) Providing means for performing comprehensive testing during commissioning and outages for maintenance giving consideration to the issues that sodium is optically opaque and chemically reactive.

(g) Testing of the actuation process during operation.

(h) Selecting equipment of proven design.

2.49 If the operation of the active reactor shutdown system for maintaining subcriticality is manual or partly manual, the necessary prerequisites for manual operation should be met.

2.50 The effectiveness of each shutdown system should be analyzed to confirm that the consequences on the fuel and the primary coolant boundary are within the prescribed limits for the events being considered and that a subcritical condition can be maintained in the long term.

(2-2) Effectiveness of shutdown

2.51 The effectiveness of the shutdown system should be demonstrated

(a) In design, by means of calculations;

(b) During commissioning and prior to startup after each refuelling, by means of appropriate neutronic and process measurements to confirm the calculations for the given core loading; and

(c) During reactor operation, by means of measurements and calculations covering the actual and anticipated reactor core conditions.

These analyses should cover the most reactive core conditions, and should include the assumption of the failure of the shutdown device(s). In addition, the shutdown margin should be maintained if a single random failure occurs in the shutdown system.

2.52 The requirements for long term shutdown and deliberate actions that increase reactivity in the shutdown state (e.g., the movement of absorbers for maintenance purposes and refuelling actions) should be identified and evaluated to ensure that the most reactive condition is addressed in the criticality analysis.

2.53 The reactor shutdown systems should be designed to keep the reactor subcritical when the reactor is in a shutdown state, considering possible movement of core elements including control rods during an earthquake.

(2-3) Rate of shutdown

2.54 In designing for the rate of shutdown, the response time of the protection systems and the associated safety actuation systems (the control rods) should be taken into account along with the corresponding response time of the reactor. In evaluating the rate of shutdown, the following factors should be considered:

(a) The response time of the instrumentation to initiate the shutdown.
(b) The response time of the signal processing.
(c) The response time of the actuation mechanism of the control rods.
(d) Time required for the shutdown element to achieve shutdown after actuation taking into account of the distance from the control rods to the active region of the core prior to the insertion, ease of entry of the control rods into the core including the fluid-dynamic effects, and the insertion speed of the control rods (as simple as gravity drop or enhanced by measures such as pneumatic pressure).

(2-4) Environmental considerations

2.55 The following environmental effects should be considered in the design of shutdown systems.

(a) *Irradiation effects:* Depletion of the absorber (e.g., boron), swelling, gas release, and heating of materials due to neutron and gamma absorption should be considered.
(b) *Chemical effects:* Compatibility of the materials used with the sodium coolant, and the transport of activated corrosion products through the primary coolant system should be considered.
(c) *Changes in structural dimensions:* Dimensional changes and movements of internal core structures due to temperature changes, or external events such as earthquakes should not prevent the insertion of the control rods.
(d) *Interference with fuel handling:* the control rods insertion should be ensured during core elements loading and unloading.

2.56 The control rod insertion mechanism should provide measures to prevent sodium or sodium oxide deposition on the rod or on the rod insertion mechanism, which could change the speed of control rod insertion or even block insertion.

(2-5) Limitation of reactivity worth and reactivity insertion rate

2.57 In order to limit power increase in case of erroneous withdrawal of control rods caused by e.g., malfunction, and operational error that could lead to exceeding design limits, withdrawal prevention measures such as interlock or rod stop system, should be provided. The rate of reactivity insertion due to erroneous withdrawal of control rods should be limited. The withdrawal of a single control rod with highest reactivity worth is considered. If the rods operate in groups, the withdrawal of the group should be considered.

(2-6) Provision for Testing and monitoring

2.58 During reactor shutdown, insertion time of control rods and their reactivity worth should be confirmed by tests such as scram simulation and control rods operation.

2.59 The control rods position and status of latch and de-latch should be monitored.

2.2.2. Reactor shutdown under ATWS[7]

(1) General safety considerations in design

2.60 The inherent neutronic characteristics of the reactor core, in combination with a passive reactivity feedback mechanism and/or a passive reactivity reduction mechanism (if needed), should prevent significant fuel degradation in case of failure of the active reactor shutdown systems that may lead to ATWS during AOOs.

2.61 Multiple inherent reactivity feedback effects should be considered in the design. These effects can work in tandem to lower reactor power during transients in response to an uncontrolled increase in core and primary coolant temperatures, and/or reactor power. These effects include the reactivity feedback due to Doppler broadening of neutron cross-sections, changes in primary sodium coolant density, fuel axial expansion, core radial expansion, control rod driveline expansion, primary vessel expansion, etc. It should nevertheless be noted that some effects could provide positive or negative feedback depending on the core design, and therefore would not necessarily lower reactor power under all conditions.

2.62 If needed, passive reactivity feedback mechanisms, which provide reversible negative reactivity feedback directly responding to the change of core conditions (such as coolant temperature increase at the core outlet or coolant pressure decrease at the core inlet) without any active signals and drive mechanisms with power source (see Annex II.2), should be designed to complement the reactor's inherent responses

[7] *Related guidelines are described in neutronic design, paras 2.16 "The design of the reactor core ..." and 2.26 "Adequately conservative assumptions should be made for ..."*

for lowering the reactor power and core temperatures, and for ensuring not to exceed the primary coolant boundary limits for DECs. The passive reactivity feedback mechanisms should be designed to achieve an acceptable response time to prevent core damage against accidents concurrent with failure of shutdown systems.

2.63 If needed, passive reactivity reduction mechanism, which provide irreversible negative reactivity directly responding to the change of core conditions (such as coolant temperature increase at the core outlet or coolant flow decrease at the core inlet) without any active signals and drive mechanisms with power source (see Annex II.2), should be designed to provide the sufficient, irreversible negative reactivity within an allowable time necessary to prevent significant fuel degradation and to ensure not to exceed the primary coolant boundary limits for DECs.

2.64 The combined performance of the inherent neutronic characteristics of the reactor core, passive reactivity feedback mechanism and/or passive reactivity reduction mechanism should be evaluated for the full range of DECs (such as unprotected loss-of-flow, unprotected loss-of-heat sink, unprotected transient over power) to demonstrate that they are capable of providing the necessary reactivity to prevent significant fuel degradation , and to assure that the primary coolant boundary limits for DECs are not exceeded. An uncertainty analysis of the inherent neutronic characteristics of the reactor core, passive reactivity feedback mechanism and/or passive reactivity reduction mechanism should also be performed to quantify the effectiveness of the design. These best-estimate analyses should cover the most reactive core conditions, on the assumption of the failure of all active shutdown systems.

2.65 The inherent neutronic characteristics of the reactor core, passive reactivity feedback mechanism and/or passive reactivity reduction mechanism should be designed to maintain their safety function taking into account the environmental conditions such as irradiation, temperature, chemical effects and geometrical changes during operational states and accident conditions up to and including DECs not involving degradation of the reactor core.

2.66 Measures to monitor the plant conditions with plant parameters such as neutron flux and coolant temperature should be provided in case of failure of the active shutdown systems.

(2) Specific safety considerations in design

(2-1) Inherent reactivity feedback

2.67 In order to rely on inherent reactivity feedback to lower the reactor power and core temperatures to a level that can be sustained without core damage or primary coolant system boundary failure during an accident with failure of all of active shutdown systems for the period of time necessary to actuate a complementary reactor shutdown measure, the total power coefficient, isothermal temperature coefficient and power/flow coefficient (see Appendix I.3), should be negative. In addition, the net effect of a reactor's inherent reactivity responses should ensure the insertion of sufficient negative reactivity to the core to prevent core damage.

2.68 Complementary reactor shutdown measures should be provided to achieve the safe shutdown state in the long term (see 2.73).

2.69 Key reactivity parameters such as reactivity coefficients should be evaluated for each core state and for the corresponding strategy for fuel management. Their dependence on the core loading and the burnup of fuel should be taken into account.

2.70 Plant testing can be used to confirm the reactivity feedback parameters of the as-built plant. A pre-startup testing allows the reactivity feedback parameters of the plant to be quantified before full power operation, including the performance of engineered passive reactivity feedback mechanisms. This process may be repeated throughout the lifecycle of the plant.

(2-2) Passive reactivity feedback mechanism and passive reactivity reduction mechanism

The following passive mechanisms should be installed if the inherent reactivity feedback cannot satisfy the recommendation 2.67.

2.71 If needed, the passive reactivity feedback mechanism and passive reactivity reduction mechanism should be designed as simple as possible. A diverse mechanism from reactivity control and active shutdown systems should be considered against common cause failures.

2.72 Since the reactor design should ensure that the accidents involving failure of all active shutdown systems are very unlikely to occur, the reliance on inherent and passive reactivity feedback mechanism and/or passive reactivity reduction mechanism is considered under the DECs as part of the fourth level of the defence-in-depth principle. If relied on, credit for impact of passive reactivity feedback mechanism and/or passive reactivity reduction mechanism is not typically considered in the analysis of DBAs; therefore, these safety features may not be required to be safety systems.

2.73 In general, it is not sufficient to achieve long-term safe shutdown conditions with only inherent reactivity feedback or passive mechanism operation. Thus, there is a need for measures introducing sufficient negative reactivity at a stage where events are stable over time. Recommended measures to achieve and maintain a safe shutdown state include, for example, forced control rod insertion.

2.74 The passive reactivity feedback mechanism and passive reactivity reduction mechanism should be tested in mock-up tests prior to its incorporation into the design. Means for checking the function should be provided.

2.75 In the design of passive reactivity reduction mechanism, the following factors should be considered:

(a) The response time to initiate the means of passive reactivity reduction.

(b) In the case of neutron absorbers, time required for the neutron absorbing element to complete its movement after the actuation. The time corresponds to power level. For example, for a passive control rod insertion system, the factors may include the distance from the control rods to the active region of the core prior to the insertion, ease of entry of the control rods into the core including the fluid-dynamic effects, and the insertion speed of the control rods.

(c) The design should provide appropriate margin for the actuation to avoid spurious activation during normal operation.

(d) If the passive reactivity reduction mechanism is based on passive means of release and insertion of the control rods, the number and the position of the control rods should ensure insertion of sufficient negative reactivity to the core to achieve subcriticality. The design should limit displacement between the control rods and control rod channels, and prevent their deformation to keep needed clearance for passive insertion of the control rods.

3. GUIDELINES FOR REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS

3.1. General Considerations in Design

(1) Objectives of the design

3.1 The primary objective of the reactor coolant system and associated systems is to ensure that an adequate flow and quality of coolant are available to remove heat from the core in all operational states and following DBA conditions. The system may also be used to mitigate the consequences of AOOs, DBAs, and DECs. And, other objective of the reactor coolant system and associated systems is to provide confinement of radioactive material for the protection of workers, the public and the environment.

3.2 All these objectives should be met by means of appropriate design provisions. These provisions may vary with the reactor type, the operating conditions, and the location of the plant (e.g., in terms of environmental conditions).

3.3 To achieve the above-mentioned objectives, the design of the reactor coolant system and associated systems should serve the following purposes:

(a) To provide and maintain a sufficient reactor coolant inventory for core cooling and to transfer the heat generated to the ultimate heat sink in all operational states and accident conditions;

(b) To maintain a sufficient flow of coolant to ensure compliance with fuel design limits; and

(c) To prevent the loss of reactor coolant inventory in the event of a reactor coolant boundary failure.

3.4 The above safety objectives of the reactor coolant system and associated systems related to paras 3.1 and should not be compromised by the failure of the components of the system.

(2) Design basis

3.5 A design basis should be defined for every structure, system and component and should specify the following:

(a) Function(s) to be performed by the structure, system or component;

(b) Postulated initiating events that the structure, system or component has to cope with;

(c) Loads and load combinations the structure or component is expected to withstand;

(d) Protection against the effects of internal hazards;

(e) Protection against the effects of external hazards;

(f) Design limits and acceptance criteria applicable to the design of structures, systems and components;

(g) Reliability;

(h) Provisions against common cause failures within a system and between systems belonging to different levels of defence in depth;

(i) Safety classification;

(j) Environmental conditions for qualification;

(k) Monitoring and control capabilities;

(l) Materials;

(m) Provisions for testing, inspection, maintenance and decommissioning.

(n) Prevention of sodium freezing

3.6 Postulated initiating events and internal and external hazards should be analyzed to set design basis (related to the plant conditions of AOO, DBA and DEC) that should be considered to establish the design criteria for the reactor coolant system and associated systems.

3.7 Structures and components of the reactor coolant system and associated systems should be designed, fabricated, erected, constructed, tested, and inspected in accordance with codes and standards appropriate to SFR design specificities and quality management system, commensurate with the safety classification.

3.8 The design methods, as well as the design and construction codes and standards used, should provide adequate margins to avoid cliff edge effects in the event of an increase in the severity of hazards.

3.9 SSCs in the reactor coolant system and associated systems should be classified according to their functions and safety significance, taking into account, for example, the following safety functions:

(a) Heat removal from the core

(b) Provision and maintenance of a sufficient reactor coolant inventory

(c) Barrier against fission product emission

3.10 The most widely used method in the reactor coolant system and associated systems design is a deterministic approach, whereby SSCs are designed to comply with guiding rules. This approach is generally complemented with a probabilistic risk assessment whose objective is to verify that the plant as designed will have no unacceptable vulnerabilities.

3.11 In order to achieve a well-balanced design, redundancy and diversity of systems and components should be considered appropriately. In designing safety systems, this consideration should be based on a deterministic approach such as the application of the single failure criterion complemented with a probabilistic risk assessment.

3.12 Equipment outages should be considered in the design.

(3) Postulated initiating events and hazards

3.13 From the list of postulated initiating events established for the design of the plant, those events that affect the design of the reactor coolant system and associated systems should be identified and categorized on the basis of their estimated frequency of occurrence.

3.14 For each of the conditions caused by the postulated initiating events, a list of the reactor coolant system and associated systems that are necessary to bring the plant to a safe and stable shutdown condition should be established.

3.15 Bounding conditions caused by the postulated initiating events should be determined in order to define the capabilities and performance of the reactor coolant system and associated systems and related equipment.

3.16 Examples of postulated initiating events that could significantly influence the design of the reactor coolant system and associated systems are as follows:

(a) Sodium leaks of primary and secondary coolant system
(b) Loss of reactor coolant flow
(c) Spurious extraction of the control rod
(d) Loss of off-site power
(e) Failure of an intermediate heat exchanger tube
(f) Failure of a steam generator tube
(g) Turbine trip
(h) Pipe breaks of the steam and feed water system
(i) Internal missiles, including turbine disintegration events
(j) Sodium leaks of auxiliary systems
(k) Fires

3.17 Examples of the external hazards that could significantly influence the design of the reactor coolant system and associated systems are as follows:

(a) Tornado, tropical cyclones
(b) Flying object from the outside
(c) Earthquake
(d) Tsunami, flooding
(e) Extreme atmospheric air temperatures
(f) Snowfall

(g) Volcanic eruption

(h) External fires including a wildfire

(4) Seismic considerations

3.18 The SSCs of the reactor coolant system and associated systems should be classified and assigned to the appropriate seismic significance categories. The SSCs that are necessary for fulfilling any of the following functions should be, irrespective of the safety class to which they are assigned, considered as classified in the highest seismic significance category. The SSCs classified in the highest seismic significance category should ensure sufficient design margin to seismic loads.

(a) Maintaining the integrity of the reactor coolant boundary and reactor cover gas boundary

(b) Achieving and maintaining decay heat removal

(c) Achieving and maintaining shutdown of the reactor

(d) Mitigating the consequences of a seismic event (e.g., Support structure for reducing earthquake load)

3.19 SSCs of the reactor coolant system and associated systems should be designed on the basis of seismic ground motions appropriate to the site and the seismic significance categories. Appropriate restraints, supports and snubbers should be provided so that the relevant limitations on stress and displacement and the no-loss-of-function criteria are met.

3.20 The dynamic effect of flow instabilities and the dynamic loads, such as sodium sloshing and pressure wave of sodium hammer, induced by earthquakes should be taken into account as the design load based on the safety analysis. Some combinations of an earthquake and other initiating events likely to occur independently of an earthquake should be taken into account. Moreover, the appropriate provisions should be made for these combinations.

3.21 It should be ensured in the design that the failure of SSCs of the reactor coolant system and associated systems or other systems designed in accordance with low seismic class has no impact on failure of those systems designed in accordance with higher seismic class.

(5) Reliability

<u>(5-1) General requirements</u>

3.22 To achieve the necessary reliability of the reactor coolant system and associated systems to control the reactivity of the core, to maintain sufficient inventory in the reactor coolant system, to remove residual heat from the core and to transfer residual heat to the ultimate heat sink, the following factors should be considered:

(a) Safety classification and the associated engineering requirements for design and manufacturing;

(b) Design criteria relevant for the systems (e.g., number of redundant trains, seismic qualification, qualification to harsh environmental conditions, and power supplies);

(c) Prevention of common cause failures by the implementation of suitable measures such as diversity, physical separation and functional independence;

(d) Layout provisions to protect the reactor coolant system and associated systems against the effects of internal and external hazards;

(e) Periodic testing and inspection;

(f) Ageing effects;

(g) Maintenance;

(h) Use of equipment designed for fail-safe behaviour.

3.23 Systems that are relied upon to fulfil a safety function should have adequate reliability commensurate with the safety function that they perform. In assessing system reliability, appropriate consideration should be given to both redundancy and diversity.

3.24 For safety systems and safety features to cope with accident conditions, providing the redundancy alone may be insufficient to provide adequate reliability owing to common cause failures; diversity could have the potential to compensate for this[8]. In assessing the potential benefit of diversity, the following should be considered:

(a) The consequence of different operating conditions

(b) The effects of different manufacturing processes on the reliability of components

(c) The consequences for the reliability of components of different work processes based on different physical methods

(d) The potential benefit or detriment resulting from the increased complexity of maintenance and/or the increased burden on operators in the event of an accident.

3.25 Operational errors can have a major influence on the reliability of the systems and components necessary to fulfil safety functions, and therefore in the design of the reactor coolant system and associated systems, adequate consideration should be given to minimize the potential for human errors.

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[8] *This recommendation is not applied to main cooling systems used for normal operation (primary coolant system, secondary coolant system, power conversion system). Diversity is a recommendation for safety systems and safety features to cope with accident conditions.*

3.26 If credit is claimed for operator actions in the initial phase of a transient, an assessment should be made of the consequences of delay and/or error on the part of the operator with respect to predetermined acceptable limits.

(5-2) Systems designed to cope with design basis accidents

3.27 Shutting down the reactor, cooling the core, controlling core reactivity, residual heat removal and transfer to the ultimate heat sink in the event of design basis accidents should all be possible despite consequential failures caused by the postulated initiating event and a single failure postulated in any system necessary to fulfil a safety function. The unavailability of systems due to maintenance or repair should also be considered.

3.28 Systems that maintain the reactor in a safe state in the long term should be designed to fulfil their function despite a single failure postulated in any of those systems (either an active failure or a passive failure[9]). Some component failures might not need to be postulated (e.g., some passive failures) if this is duly justified.

3.29 The on-site power source (i.e. the emergency diesel generator and/or batteries) should have adequate capability to supply power to electrical equipment to be operated in design basis accidents for shutting down the reactor, cooling the core, removing and transferring residual heat to the ultimate heat sink, and maintaining the reactor in a safe state in the long term.

3.30 Vulnerabilities to common cause failures between the redundancies of the safety systems should be identified, and design and layout provisions should be implemented to make the redundancies independent as far as is practicable. In particular, adequate physical separation should be implemented between the redundant trains of the safety systems to prevent or minimize common cause failure due to the effects of hazards considered for design.

(5-3) Safety features for design extension conditions

3.31 The more likely combinations of postulated initiating events and common cause failures between the redundancies of the safety systems should be analyzed. If the consequences exceed the limits given for design basis accidents, the reliability of the safety systems should be improved (e.g., vulnerabilities to common cause failures should be removed) or additional design features should be implemented to prevent such events from escalating to an accident with core damage or offsite consequences.

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[9] *The design shall take due account of the failure of a passive component, unless it has been justified in the single failure analysis with a high level of confidence that a failure of that component is very unlikely and that its function would remain unaffected by the postulated initiating event.*

3.32 The recommendations in paras 3.27–3.30 should also be applied in respect of design extension conditions, taking into account that meeting the single failure criterion is not necessary and that the additional safety features for design extension conditions are supplied by the alternate AC power source and batteries.

(6) Selection of materials

3.33 Materials used for the reactor coolant system and associated systems should be selected to be suitable for the service conditions expected in all operational states and under DBA conditions.

3.34 Materials should be tolerable against severe plant conditions encountered in a DEC.

3.35 Materials used for the reactor coolant boundary (including joining materials such as welding materials) should be compatible with the contained reactor coolant, adjoining components, and overlay or radiolytic products. Materials specified for the reactor coolant system and associated systems should comply with applicable provisions of the used code appropriate to SFR design specificities, including but not limited to the following properties and characteristics:

(a) Resistance to heat loads
(b) Strength, creep and fatigue properties
(c) Corrosion and erosion properties
(d) Resistance to effects of irradiation
(e) Resistance to temper embrittlement
(f) Ductility characteristics (including crack growth rate)
(g) Fracture toughness (brittle failure) characteristics
(h) Ease of fabrication (including weldability)

(7) Layout considerations

3.36 The layout of the reactor coolant system and associated systems should be designed considering the following factors.

(a) Radiological protection of site personnel
(b) Protection against the consequences of pipe failure (sodium fire)
(c) Protection against internal missiles
(d) Provisions to facilitate testing and inspection
(e) Separation and isolation of sodium-containing facilities and water containing facilities

3.37 The layout of safety systems should be designed to maintain the minimum required capability in the event of a failure in one train of protection or in the event of needing to survive any internal and/or external hazards (e.g., earthquake, fire and flooding).

3.38 In preparation for possible internal flooding in the steam generator room, measures such as physical barriers and separation to prevent water ingress and direct contact with sodium should be considered to prevent sodium-water reaction.

3.39 Layout of the reactor coolant system and associated systems should be designed to ensure the removal of decay heat by the natural circulation of the reactor coolant in the event of a total loss of power supplies to pumps.

3.40 The layout of the reactor coolant system and associated systems should be designed to enable inspection, maintenance, repair, and replacement of SSCs with the radiological protection of site personnel.

(8) Interface considerations

3.41 Appropriate interface devices should be provided for connections between systems or components belonging to different safety classes. These interface devices should prevent the loss of the safety function of the system or component with the higher safety classification and should prevent the release of radioactive material. An interface device should have the same safety classification as the system or component with the higher safety classification to which it is connected.

3.42 In designing the structures of reactor coolant system and associated systems, their influence on the overall safety of the plant should be taken into account. It should be ensured that the temperatures of the structures and components interfacing with the reactor coolant system and associated systems are maintained within acceptable limits and that a provision is made for in-service inspections. Components and structures that are directly anchored to the containment should be designed that their failure would not cause the loss of containment leak tightness.

3.43 Interface considerations should include flow rates, various loading conditions, response times and heat transfer capabilities.

3.44 Examples of loads on the supporting structures for the reactor coolant system and associated systems are as follows:

(a) The deadweight of components in operational states and DBA conditions
(b) Thermal expansion in steady state or transient conditions
(c) Earthquake loads
(d) Transient loads

3.45 Structures interfacing with the reactor coolant system and associated systems include the following items:

(a) Buildings supporting or housing the reactor coolant system and associated systems
(b) Equipment and piping supports

(c) Snubbers and their anchors
(d) Pipe whip restraints
(e) Building penetrations
(f) Protective structures (e.g. barriers and shields)

3.46 The design of the reactor coolant system and associated systems should also reflect constraints imposed by the support systems and structures. Support systems include, for example, ventilation systems, compressed air systems, electric power systems and the instrumentation and control system.

3.47 In designing a system, appropriate consideration should be given to the consequences of the following design conditions on other systems.

(a) Differences in the scale of damage and their locations in the reactor coolant boundary in the design of the back-up structure (e.g., guard vessel) and/or containment vessel
(b) Configuration of components in the reactor coolant system and associated systems in the design of the ventilation system

(9) Considerations of isolation between systems

3.48 Auxiliary system pipework penetrating the reactor coolant boundary should be equipped with adequate isolation devices to limit any loss of radioactive fluid (primary coolant or cover gas) and to prevent the loss of coolant through interfacing systems so that cooling of the reactor core can be maintained.

(10) Instrumentation and control system

3.49 The reactor coolant system and associated systems should be provided with adequate instrumentation for the following purposes:

(a) Monitoring of the process parameters (e.g., pressure, temperature, coolant level and flow rate) that indicate whether the system or component is being operated within the range specified for its normal operation;
(b) Early detection of abnormal operating conditions;
(c) Automatic operation of systems necessary for the mitigation of the consequences of an accident;
(d) Providing the main control room and the technical support center with appropriate and reliable information for accident management;
(e) Periodic testing of systems and components;
(f) Supporting an understanding of the maintenance state of structures, systems and components.

3.50 Instrument sensing lines should be designed such that the measurement parameters (e.g. magnitude, frequency, response time and chemical characteristics) are not distorted.

3.51 Potential leakage of radioactive material from and into the reactor coolant system and associated systems should be monitored.

(11) Provision for in-service inspection, testing and maintenance

3.52 SSCs of the reactor coolant system and associated systems should be designed to facilitate the performance of inspection, maintenance and testing tasks without undue exposure of the site personnel to radiation throughout the lifetime of the plant.

3.53 Periodic testing, when required, should simulate under the conditions which systems and/or components are expected to operate. Test conditions should not jeopardize plant safety.

3.54 Automated or remotely operated equipment can be used for in-service inspection to keep the exposure to radiation of the inspection personnel as low as reasonably achievable and within any limits specified by legislation or by the regulatory body.

3.55 Provided that LBB condition is satisfied (see 3.67), continuous leakage monitoring should be applied as an inspection measure for the boundary of sodium-containing equipment that have a significant impact on the safety when its sodium-containing capability is significantly deteriorated.

3.2. Primary Coolant System

3.2.1. Component design

(1) Consideration in the designs of components of primary coolant system

3.56 Limitation of thermal stress influence such as creep fatigue damage on reactor structure should be adequately implemented to meet proper codes and standards if necessary. For example, the causes are rapid temperature change and thermal stratification of reactor coolant caused by transient events such as reactor trip, change of reactor coolant level and temperature variations during the start-up/shutdown operations.

3.57 Thermal loads should be confirmed by thermal hydraulic analysis and/or experiments for the primary coolant system.

3.58 Excessive pressure increase, erosion and corrosion should be prevented. Cavitation, in particular in the primary pumps, should be prevented.

3.59 Mechanical and thermal fatigue damage due to flow induced vibration, and thermal striping caused by mixture of different temperature fluids should be eliminated or sufficiently reduced to maintain the expected behavior of the components of the primary coolant system. Examples of measures include below.

(a) Flow induced vibration

- Ensuring structural rigidity to reduce vibration
- Setting flow conditions and designing structures and components to avoid resonance

(b) Thermal striping

- Reducing difference in temperature between mixing fluids
- Adjusting flow conditions such as flow velocity ratio of mixing fluids
- Installing thermal resistance on upper core structures exposed to thermal striping
- Using materials having high resistant against thermal fatigue

3.60 Sodium impurity (sodium hydroxides and oxides content) should be controlled and other impurities accidentally mixed into sodium should be removed so that those impurities do not cause obstruction of the coolant flow and degradation of SSCs' safety functions due to corrosion or plate out.

(2) Consideration in system design for ensuring component structural integrity

3.61 The design considerations for the primary coolant system to prevent excessive thermal load should include the following:

(a) Detection of any degradation of the capability for core cooling or any deterioration of components important to safety such as the reactor vessel and pipes which constitute the reactor coolant boundary, e.g., by means of the measurement of operating parameters for heat transport, monitoring for leaks of reactor coolant, detection of abnormal vibration of pump, and monitoring of displacement of pipe support structures.

(b) The pumps of the primary coolant system should be designed to have adequate flow coast-down characteristics in the event of a reactor trip under transient or DBA conditions to avoid rapid temperature change of the reactor coolant and to ensure the integrity of the structures and components of the primary coolant system considering transient influence of reactor power and reactor coolant flow.

(3) Consideration in structural design of reactor vessel

3.62 The design considerations for the reactor vessel should include the following:

(a) The welds in the reactor vessel should have high reliability. In particular, due consideration should be given to high-temperature (in particular with regard to possible creep effects) and neutron irradiation conditions.

(b) The reactor vessel should be designed to withstand all cyclic loads that are expected to occur over the plant lifetime by limiting fatigue factor and creep fatigue factors below the value defined by standards. The design documentation should include clear specifications of those loads that are necessary for the determination of the cumulative usage factors.

(c) The choice of material, the structural design, the welding and the heat treatment should be such as to ensure ductility and toughness of the material of the reactor vessel throughout the plant lifetime. The ductility and toughness of the reactor vessel should be ensured by limiting the maximum neutron fluence and by the use of base material and weld metal taking account of these chemical compositions and acceptable level of irradiation embrittlement. The temperature of the reactor vessel should be limited in order to ensure acceptable creep effects during operational states and DBA.

(4) Consideration in structural design of reactor vessel internals

3.63 The reactor vessel internals (core support structures and other internals) should be designed

(a) To channel properly the coolant flow through the core and heat exchangers and, if it is equipped, the reactor vessel wall cooling;

(b) To maintain core geometry;

(c) To withstand the effects of design basis earthquakes without loss of capability;

(d) To prevent unacceptable flow induced vibration and thermal striping; and

(e) To ensure that fuel design limits are not exceeded in operational states and DBAs.

3.64 The maximum neutron fluence should be limited to ensure a sufficient ductility and toughness of reactor vessel internals. Adequate provisions should be in place as required to provide shielding against neutron radiation.

3.65 For the in-vessel retention, if necessary, the following should be considered in the post-accident heat removal phase after core degradation resulting from an unprotected transient according to descriptions of the Safety Approach SDG.

(a) Retention of a degraded core

Measures should be provided to retain degraded core materials to facilitate post-accident heat removal. Re-criticality of a retained degraded core should be prevented during the post-accident heat removal phase. The retention structure should resist the thermal load from a degraded core, as well as mechanical loads, including any loads from fuel-coolant-interactions.

(b) Ensuring a coolant circulation path and heat sink for in-vessel retention

A coolant flow path and heat sink should be available for cooling of degraded core materials. Natural circulation capability should be incorporated. Structures and components that form the flow paths should maintain their functions against adverse effects, such as mechanical loads from fuel-coolant interactions and blockage by dispersed fuel debris.

(c) Protection of a degraded core retention structure (if needed)

The reactor structure should facilitate molten fuel dispersion and solidification in the presence of adequate heat removal capability to prevent or mitigate erosion of any structure intended to retain the degraded core materials caused by molten fuel. Depending on the characteristics of the degraded core materials, preventive measures against erosion, such as installing protective layers on core retention structures, should be considered.

(5) Consideration in structural design of primary pipes

3.66 In the design of the primary pipes, including internal pipes connecting the primary pump outlets to core inlet plenum, the structural integrity should be ensured against e.g., thermal expansion, thermal transient, earthquake, flow induced vibration, coupled vibration with pump, etc.

3.67 The leak before break (LBB) concept or techniques of break preclusion should be considered for use in determination of pipe failure size and in preventing pipe rupture.

Following conditions should be fulfilled in adopting LBB concept to safety evaluation and design of the primary coolant system. If the LBB concept is not or cannot be applied, consequences of the pipe rupture should be addressed in the safety demonstration.

(a) Evaluation methods of LBB should be based on proper standards relevant to SFR. Primary pipes should be designed to satisfy the LBB condition, i.e., detectable size of leak is sufficiently small comparing with size of unstable fracture of pipe defined considering the most penalizing loading.

(b) Leak detecting facilities should be provided to detect the primary sodium leaks evaluated based on the LBB concept with a sufficient margin.

(c) Quality of the primary coolant system should be ensured by inspection in manufacturing and pre-service inspection.

(d) Process instrumentation measures should be provided to ensure that operating conditions of the reactor coolant boundary such as radiation dose, primary sodium purity, and thermal loads are within values assumed in design.

3.68 The necessary amount of the primary sodium coolant flow for core cooling should be ensured in case of a postulated internal pipe failure in the reactor vessel.

(6) Testability and inspectability

3.69 Structures and components important to safety should have provisions for inspection during their service life with regard to their capability to perform their intended safety functions as well as their physical integrity, including any changes in the properties and characteristics of the materials used.

3.70 Sodium leak detectors should be installed in the guard vessel and guard pipes and/or rooms in which the primary coolant system is located in order to continuously monitor sodium leaks from the vessel and pipes.

3.71 Accessibility for necessary maintenance and inspection including remote inspection should be considered in the design of the primary coolant system.

3.2.2. Reactor cover gas and its boundary

(1) Basic function

3.72 Inert gas should be used as cover gas on the sodium free surface in the reactor vessel or other vessels such as expansion vessel, if it is equipped, in the primary coolant system to prevent a chemical reaction at the free surface, to accommodate volume changes in sodium due to various operating states/transients, and to reduce heat load on the reactor roof. The reactor coolant boundary should be designed as a barrier against radioactive material release and be closed by the reactor cover gas boundary. The reactor cover gas boundary should be designed to be within acceptably low leak rates and isolation function in the connecting lines should be provided.

3.73 A design leak rate from the cover gas region should be determined, and operational performance should be maintained within the associated limits. A cover gas make-up and clean-up system should be provided to keep impurity in the cover gas below acceptable limit.

3.74 Any systems in which air or water is used as working fluid should not be connected to the reactor coolant or reactor cover gas boundaries including envelope in gas. Measures such as physical barriers and separation for such systems should be considered to prevent accidental air or water ingress into the reactor cover gas boundary.

3.75 To prevent sodium vapor and mist in the cover gas from being released into the atmosphere and the non-sodium equipment on the exhaust side of the cover gas from being affected, equipment should be installed for removing sodium mist and vapor in the cover gas. And the cover gas boundary and structures and components installed inside the reactor cover gas boundary should be designed to maintain their functions under the environmental conditions including sodium vapor and mist, associated radionuclides, and activation products.

3.76 The covers gas should be slightly positive-pressurized (compared to the above roof area) to prevent air ingress. Besides, detectors should be implemented in the cover gas to detect any abnormal air ingress. Due consideration should be given to sodium geyser effect through a small diameter pipe that penetrates the cover gas boundary and whose tip is immersed in sodium, i.e., sodium leak through the penetrating lines due to cover gas pressure. For example, cover gas pressure should not be higher than necessary. Protective measures should be taken against sodium leak such as tight sealing and back-up structure.

(2) Overpressure and negative-pressure prevention

3.77 In the design of primary coolant system, provisions for preventing over- and negative-pressurization of the reactor coolant boundary and the reactor cover gas boundary should be taken. The following measures should be taken so that the reactor coolant boundary and the reactor cover gas boundary can be maintained within the design limits of the corresponding plant condition (i.e., operational states and accident conditions). Risk of the reactor vessel buckling due to negative-internal pressurization and external over-pressurization must be prevented under all the plant conditions.

(a) Monitoring the cover gas pressure in operational states and DBA,
(b) Controlling the cover gas pressure within operational limits in operational states
(c) Providing devices for overpressure relief such as safety valves or relief valves, if necessary (such devices could be used for core damage situations in DEC)
(d) Providing design to avoid excessively large negative cover gas pressure

(3) Isolation function

3.78 Lines that are connected to the reactor cover gas boundary should be provided with adequate leak detection of the reactor cover gas and isolation in order to limit radiological consequences of lines' failure. The single failure criteria should be applied to the detection and isolation. For example, line isolations are performed by the following signals:

(a) Cover gas flow rate high
(b) Cover gas radioactivity high
(c) Radioactivity high within the containment structure

3.79 Consideration should be given to the characteristics and importance of the isolation and its reliability targets. Isolation devices either should be normally closed or should close automatically on demand. The response time and speed of closure should be in accordance with the acceptance criteria defined for initiating events.

(4) Prevention of gas entrainment

3.80 In order to limit entrainment of the cover gas from the sodium free surface into the primary sodium and limit the entrained gas into the primary circuit, limitation of sodium free surface velocity below values leading to free surface fluctuation and formation of vortices, and arrangement of vessel internal structures constituting the primary circuit should be taken into consideration. If sloshing due to a seismic event below the scram level affects gas entrainment, seismic-induced sodium sloshing should be considered in the design.

(5) Testability and inspectability

3.81 Structures and components which constitute the reactor cover gas boundary should have provisions for inspection during their service life with regard to their capability to perform their reactor cover gas boundary function as well as their physical integrity, taking account of any changes in the properties and characteristics of the materials used. Continuous leak monitoring of e.g., radioactivity in the rooms where the primary coolant system is installed, cover gas pressure and cover gas contaminants should be provided to ensure the function of the reactor cover gas boundary.

3.2.3. Reactor coolant level maintenance

(1) Ensuring reactor coolant level for core cooling

3.82 Provision should be made for controlling the free surface level of the primary coolant to ensure that specified design limits of the level change for maintaining the core cooling with coolant circulation are not exceeded in operational states and accident conditions.

3.83 Guard vessels should be designed to maintain the sodium surface of the primary coolant system sufficiently above the level necessary for decay heat removal in the case of a sodium leak accident (Emergency sodium Level (EsL)) in the primary coolant system. Therefore, gap volume between guard vessel and reactor vessel should be limited. Volumetric change of sodium due to temperature change should be considered to achieve a safe shutdown state.

3.84 Provisions, e.g., sodium leak detection, pump trip and isolation of connecting lines to the reactor coolant or cover gas boundaries should be made to reduce the leakage amount of sodium from the primary coolant system in case of reactor coolant boundary failure.

3.85 The gap space between guard vessel and reactor vessel should be inert gas atmosphere to mitigate chemical reaction of leaked sodium. The gas pressure in the gap space should be controlled to be slightly lower than that of primary coolant system so that a reactor coolant boundary failure can be detected by monitoring sodium or cover gas leak in the gap space. The inert gas atmosphere system should be readily inspected, examined and maintained throughout life.

3.86 Provisions should be made to maintain the reactor coolant level well above the level necessary for decay heat removal, in case of failure of pipes connecting with interfacing system (e.g., primary sodium purification system).

(2) Practical elimination of core uncovering due to sodium inventory loss

3.87 Robust demonstration of the practical elimination should be made taking the following recommendations into account.

3.88 Reactor vessel, guard vessel and piping of the primary coolant system should be designed, manufactured, and installed to have a high level of reliability. Guard pipes of the primary coolant system for loop-type design should be designed, manufactured, installed, and maintained, and inspected to have reliability as high as possible.

3.89 For prevention of failure of guard vessel after sodium leakage from the reactor vessel, the following should be taken into account:

(a) The guard vessel should withstand thermal loads due to a sodium leak from the reactor vessel.

(b) The guard vessel should withstand mechanical loads from all possible causes such as earthquakes, while retaining leaked sodium.

(c) The guard vessel should withstand any interference with a failed reactor vessel (even considering thermal expansion, vibration, etc.).

3.90 To prevent failure of the guard vessel prior to an accidental sodium leakage from the reactor vessel, the following should be taken into account:

(a) The guard vessel should be subjected to limited loads during normal operation.

(b) The guard vessel leak tightness should be controlled throughout the reactor lifetime.

3.91 To prevent common cause failure between reactor vessel and guard vessel, the following should be taken into account:

(c) The design should separate the support structures of the reactor vessel and guard vessel to the extent practicable, and prevent failures of common parts of the support structures with the highest level of reliability.

(d) The design should ensure sufficient margins against internal/external hazards including earthquakes.

(3) Measures against sodium leaks from the primary loops (Note: only required for loop-type designs)

3.92 Guard pipes, as well as piping arrangements, should be provided to preclude the risk on reactor vessel draining for ensuring core cooling and for reducing the effects of sodium leaks from any primary loop pipes or components. Design basis leaks should be determined with due consideration taken to direct and indirect consequences of failure, if LBB concept is applied.

3.93 Guard pipes should be designed to withstand loads associated with leaked sodium, pipe whip and fluid force even considering a large break of primary coolant system piping under DECs.

3.94 In order to prevent significant core damage in severe conditions such as multiple leaks from the primary loops, even in a condition such that all the primary loops cannot keep the coolant circulation, a cooling system should be installed in the reactor vessel that should be able to cool the core preferably by natural convection.

(4) Testability and inspectability

3.95 Structures and components important to safety should have provisions for inspection during their service life with regard to their capability to perform their intended safety functions as well as their physical integrity, including any changes in the properties and characteristics of the materials used.

3.96 Reactor vessel, guard vessel, piping of the primary coolant system should have provisions for inspection. Environmental and operational conditions should be maintained within the permissible range and monitored. Their integrity should be assessed based on data obtained by monitoring and/or regular inspection.

3.97 Guard pipes of the primary coolant system for loop-type design should have provisions for inspection.

3.3. Decay Heat Removal Systems

3.3.1. Design objective

3.98 Objective of the safety design of the decay heat removal systems and related safety features is to practically eliminate complete loss of the decay heat removal function that could lead to severe core damage and large failure of the reactor coolant boundary. Robust demonstration of the practical elimination should be made taking the following recommendations into account.

3.99 Proven technology, based on the design, construction and operation experience of SFRs, should preferably be applied to the basic design of decay heat removal systems.

3.100 Expected performance of each decay heat removal system should be ensured throughout the reactor lifetime.

3.3.2. Decay heat removal under an AOO and a design basis accident

(1) Postulated events

3.101 In AOOs and DBAs, decay heat removal systems should be provided for long-term cooling of the reactor core after reactor shutdown. The system configuration and heat removal capacity of decay heat removal systems, as well as transient characteristics, such as flow coastdown of the primary pumps, should be set to meet design limits of AOOs and DBAs, assuming a single failure.

3.102 The decay heat removal systems should have sufficient reactor core cooling function against all AOOs and DBAs with consideration of the loss of off-site power and the single failure of any component.

3.103 The decay heat removal systems should be designed against the consequences of internal and external hazards such as seismic hazards that have potential to jeopardize its safety functions. For the design of the ultimate heat sink using air, external events which affect the air coolers e.g., tornado, hurricane, extreme temperature, volcano ash fall, airplane crash, a forest fire, external missiles, should be considered with due consideration for site specificities. If the ultimate heat sink is water in the sea, river or lake, influence of tsunami, flood and such should be taken into account with due consideration for site specificities.

3.104 The necessary power to perform the function of decay heat removal system should be provided by the emergency power supply system. The start-up time, capacity and durability of the emergency power supply should be adequate to ensure the performance of core cooling function in an accident.

(2) Redundancy and diversity

3.105 As described in general consideration (See 3.1. General Considerations in Design (5) Reliability), the decay heat removal systems should have adequate redundancy and diversity to fulfil their safety functions. An assessment of the adequacy of redundancy and diversity should be made on the basis of deterministic methods supplemented by probabilistic methods.

3.106 For plants at which preventive maintenance at power is intended, the need for considering a postulating initiating event that is coincident with the maintenance of one safety system train should be evaluated.

3.107 Since redundant or diverse systems can be vulnerable to events (e.g., internal fires) resulting in common cause failures, appropriate physical barriers or physical separation or a combination of both should be provided as far as practicable (See 3.1. General Considerations in Design (5) Reliability). Any associated control systems and power supply should also include adequate separation to prevent common cause failures of redundant or diverse systems.

3.108 Diversity in operation mode and component design should be implemented to the extent practicable.

(3) Prevention of sodium freezing

3.109 In order to prevent sodium freezing, measures should be such that;

(a) Control of decay heat removal is appropriately achieved to avoid excessive cooling.

(b) Control of sodium temperature is appropriately achieved by trace heating and thermal insulation system.

(c) Maintaining of a minimum flow rate of sodium is achieved by appropriate operation measures, e.g., forced convection of sodium in stand-by condition.

(4) Measures against sodium leaks

3.110 Design provisions should be made for the detection of sodium (or other types of coolant such as NaK) leaks, e.g., aerosol and contact type detectors, and for mitigation of the effects of chemical reactions between sodium and air or water, e.g., guard pipes or enclosures, sodium drain systems.

3.3.3. Decay heat removal under a design extension condition

3.111 In order to cope with DECs, which are more severe than DBAs, or which originate from multiple failures of SSCs, the design extensions of the decay heat removal systems and/or the alternative cooling measures should be provided.

(1) Design extension of decay heat removal systems

3.112 Adequate independency and diversity should be implemented to ensure avoidance of common cause failure so that the decay heat removal function is ensured under postulated situation in design including DEC.

3.113 Physical separation between decay heat removal systems or protection of some of the decay heat removal systems should be provided against internal and external hazards and against common cause failure mechanisms generated by hazards.

3.114 It is necessary to identify all credible factors leading to loss of decay heat removal function and to confirm that measures can be implemented to overcome all of them. The decay heat removal function should provide adequate margins to avoid cliff edge effects.

3.115 The decay heat removal systems should be available for long-term cooling of a degraded core to avoid reactor coolant boundary failure against unprotected transients with core damage. (see 3.65)

3.116 The capability of the decay heat removal systems should cope with more severe initiating events than DBAs, taking potential internal and external hazards and their possible combinations into account. Accident management provisions, e.g., manual operation of air cooler dampers, utilization of mobile power sources, should be made so that recovery operations can be performed.

3.117 Practical accident management procedure should be established. In postulated abnormal conditions, necessary time margin and operation environment for implementing the accident management provisions should be secured.

3.118 The decay heat removal systems should have adequate redundancy, diversification and design margins to assure sufficient heat removal capacity during DECs. Risk of sodium freezing due to over cooling should be taken into account in the system configuration and operation of the decay heat removal systems.

(2) Alternative cooling measures[10]

3.119 The heat removal capacity of alternative decay heat removal measures against DECs should be set so that the reactor systems do not exceed the design limits for DECs.

3.120 In order to avoid a common cause failure, the alternative cooling measures should be independent, to the extent practicable, from the decay heat removal systems and the main cooling system. The alternative cooling measures should have diversity for the decay heat removal systems to the extent practicable.

3.121 The alternative cooling measures should be designed and located to withstand or protect against internal and external hazards and against common cause failure mechanisms generated by hazards even in case that all of the decay heat systems fail.

3.122 Start-up and operation procedures should be established in line with diagnostic processes for the plant state, even under severe plant conditions, such as after failure of DBA provisions.

3.3.4. Securing decay heat removal by natural circulation of sodium

3.123 In order to enhance the reliability of the decay heat removal function and to maintain the function under long-term loss of all AC power, natural circulation capability should be implemented in the measures for decay heat removal

3.124 DHRSs should consider incorporating a natural circulation function. Designers can decide when to use the natural circulation. For example, some DHRS, which has a forced circulation function, can also operate its natural circulation under DEC, even the system lost its forced circulation function. Another DHRS with only natural circulation capability can even operate under AOO, DBA and DEC. These systems should have sufficient capacity to perform their intended functions and not exceed the design limits of fuel and reactor coolant system and associated systems for AOO, DBA and DEC.

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[10] In order to justify the practical elimination of complete loss of decay heat removal function, the alternative cooling measures are introduced if necessary.

3.125 The alternative cooling measures, if provided, should utilize the natural circulation capability to the extent practicable.

3.126 The arrangement of piping or flow paths should reduce system pressure loss in order to facilitate natural circulation.

3.127 In order that the decay heat can be removed by natural circulation, the primary coolant system should be designed such that

(a) Adequate height differences between core and heat exchangers such as DHX, IHX or height of reactor vessel, and adequate pressure loss of the system and components should be provided to ensure sufficient natural circulation flow through primary coolant system.

(b) Transition from normal power operation to decay heat removal situation by natural circulation should be smooth by e.g., the time constant of the coastdown of the primary pumps should be optimized to limit core overheating during transition from nominal flow to natural circulation regime, and should not depend on the utilization of any AC power.

(c) The natural circulation capability should be ensured even in unbalanced flow conditions, e.g., one primary pump seizure.

3.128 In order that the decay heat can be removed by natural circulation, secondary systems for decay heat removal should be designed such that

(a) For DRACS, PRACS, IRACS, SGACS (see Appendix I.5)

- Primary-secondary coolant heat exchangers should be located at sufficiently higher elevation above the reactor core.
- Secondary coolant-air heat exchangers should be located at sufficiently higher elevation above the primary-secondary coolant heat exchangers.
- Pressure loss of the systems and components should be kept as low as possible by e.g., minimizing or removing valves, shortening pipes.
- Air stacks connected to secondary coolant-air heat exchangers should be tall enough to have sufficient draft effect.
- Unintended draining, unintended isolation of the loop and freezing of the secondary coolant should be prevented to ensure the continuous and effective natural circulation to remove the decay heat.

(b) For RVACS (Components which constitute the air channel outside of guard vessel for decay heat removal)

- The air stacks of the system should be tall enough to have sufficient draft effect.

- The reactor/guard vessel structure should be designed to allow adequate heat transfer from the primary system to the environment for predicted transient and environmental conditions.
- Pressure loss of the systems and components should be kept sufficiently low to ensure natural circulation flow by e.g., minimizing obstructions with such as baffles or grates, shortening flow paths.
- Ingression of external contaminants into air flow paths should be prevented to ensure the natural convection.
- If necessary, insulation of the concrete structure should be provided.

(c) The use of active devices and instruments for the purpose of control should be minimized in order to cope with long-term loss of all AC power.

(d) The use of active devices on decay heat removal loops, such as electromagnetic pumps, should be designed so as not to impede natural circulation flow when faulted or inactive.

(e) From the viewpoint of enhancing the reliability and minimizing the operators' load, the number of devices which need activation and operations should be minimized. If many parameters will be monitored and analyzed, automatic system should be adopted. The necessary automatic or manual operations should be minimized to establish the natural circulation.

(f) For any vessel cooling system to contribute to a viable safety basis for reactor licensing, their ability to maintain the intended safety function throughout the operating life should be assured via availability of pedigreed technical data that can quantify the performance during accident scenarios under degraded conditions that include adverse atmospheric circumstances and passive component failure modes.

3.3.5. Safety considerations of tests and inspections

3.129 The design of the decay heat removal system should be such that periodic functional testing of the active components in the system is possible during normal operation at power.

3.130 The design of the decay heat removal system should be such that implementation of the tests does not impair its functional capability.

3.131 The decay heat removal system should be such that its operation condition is monitored at all times.

3.132 The decay heat removal function by natural circulation should be assured through testing and/or monitoring. For instance, temperature and flow rate should be confirmed by transient tests from forced circulation to natural circulation during a safe shutdown state in the commissioning stage or pre-start-up phase in service. For decay heat removal systems that cannot be tested at full operational capacity (such as a reactor vessel cooling system), separate component testing and system monitoring should be developed to ensure adequate performance during transients.

3.4. Measures for prevention and mitigation of Sodium Chemical Reaction

3.4.1. Measures for prevention and mitigation of sodium leakage and combustion

(1) Prevention of sodium leakage

3.133 Causes of sodium leakage, including thermal stress, fatigue, thermal fatigue, buckling, overstress, flow-induced vibration, fretting, defects of base material, material deterioration, welding defect, assembly defect, corrosion, should be taken into consideration in the design of sodium-containing components and piping. The structure design, selection of materials, manufacturing and inspection should be performed based on appropriate codes and standards to ensure high-quality of the sodium-containing components and piping.

(2) Mitigation of sodium leakage and control of sodium combustion

3.134 Sodium leakage with negligible impact on the core cooling should be detectable. Plant protective actions such as pump stop etc. should be provided so that the plant operators can take the actions according to the predefined procedure. Sodium leakage which may affect core cooling and/or may cause extensive leaked sodium combustion should be detected and the protective actions should be automatically activated to prevent adverse effects on systems with the fundamental safety functions.

3.135 In the case that systems for managing sodium leakage, such as pump stop, have automatic actuation e.g., by detecting sodium leakage, the systems should be designed to ensure their operation even under the single failure of the detector.

3.136 The number of small diameter pipes connecting to the systems containing sodium or other penetrations of the boundary of reactor coolant system and associated systems should be minimized to reduce the likelihood of the leakage as much as possible. Measures e.g., isolation valve, and backup seal should be provided for the connected small-diameter piping to prevent or minimize sodium leak.

3.137 For sodium leakage, provisions should include the control of the amount and severity of sodium combustion, e.g., by designing a protective structure, or by using a number of smaller isolated compartments instead of one large room, or by providing an atmosphere that can mitigate combustion using e.g., an inert gas.

3.138 Sodium leak detectors should be installed in rooms and/or in protective structures. Signals from the detectors which need to manage incident procedure should be displayed in the main control room, and an alarm should be sounded when the leakage is detected.

3.139 Small diameter pipes connecting to the protective structure for controlling its structure's internal inert gas pressure or sampling, if provided, should be designed to be immediately isolated by the sodium leak detection signal to prevent the leaked sodium from spreading out to other systems via those lines. Configuration examples of measures against sodium leakage and combustion are shown in ANNEX II.3.

3.140 Measures against primary sodium leakage are combined with measures described in guidelines for containment system and should be taken account of preventing radioactive materials release to the environment. (See 4.3.1 Control of pressure and temperature (2) Sodium leakage and combustion)

3.141 Measures against the secondary sodium leakage outside of the containment structure should be taken account of preventing human harm when aerosol is released to the environment.

(3) Postulated sodium leakages for DBAs and DECs

3.142 For DBAs, postulated sodium leakage should be determined on the basis of LBB concept. (In case of the primary coolant system, see 3.2.1 component design (5) consideration in structural design of primary pipes) Depending on the design, for pipes, e.g., branch pipes connected to support systems, for which LBB is not feasible, consequences of the pipe rupture should be addressed.

3.143 For DECs which result in more severe conditions than DBAs, a larger leakage than DBAs, multiple leaks, failure of mitigation measures at leaks, etc. should be assumed to design safety measures.

(4) Prevention of adverse effects on safety functions

3.144 Measures such as provision of protective structures, physical separation from systems containing sodium, or resistance to the effects of sodium combustion (hot sodium contact, heat radiation, aerosol deposition) should be considered in the design to protect SSCs important to safety.

3.145 Main pipes of the primary coolant system for a loop type SFR and branch pipes directly connected to reactor coolant boundary and their guard pipes should be independent and physically separated as far as possible to prevent common cause failure. Protection by segregation features e.g., physical barriers should be implemented so far as is reasonably practicable. The guard pipes should maintain their function against postulated pipe break. As for pipes of secondary coolant system (including pipes of decay heat removal system) in the containment structure for a pool type and a loop type SFRs, guard pipes protecting the secondary pipes should maintain the function of the guard pipes to protect SSCs important to safety against postulated pipe break.

(5) Testability and inspectability

3.146 The protective structures should be designed to maintain the internal pressure slightly positive to the ambient pressure in order to prevent air infiltration from outside, and should be designed to be monitored and periodically inspected for their sodium leak mastering function.

3.4.2. Measures for prevention and mitigation of sodium-water reaction

Specific safety considerations in design are described below.

(1) Prevention of sodium-water reaction

3.147 To prevent sodium-water reactions, the following measures should be adopted:

(a) The design limits such as maximum temperature and pressure should be provided as a set of design conditions for the system with a sodium-water/steam interface. These design limits should not be exceeded in any of the operational states or DBAs. Adequate margin should also be provided to cope with DECs. A comprehensive set of load conditions such as design basis earthquake, transient thermal loads, and flow induced vibrations should be considered to ensure the integrity of such system with sufficient margin.

(b) The concentration of impurities in the water/steam systems, as well as the interfacing sodium systems, should be controlled to prevent the boundary failures due to erosion or corrosion.

(2) Mitigation of sodium-water reactions

3.148 The following measures should be adopted for mitigation of sodium-water reactions so that their impact on the reactor and plant, including the outer shell of the steam generator, can be minimized and the safety functions of SSC will be adequately ensured:

(a) A water/steam leak detection system should be installed to detect any interaction with sodium, such as an increase in the secondary coolant system pressure, accumulation of hydrogen gas as quickly as possible to allow timely isolation of the leak and facilitate reactor shutdown. Active components, such as valves, required to prevent further damage and to mitigate consequences should be automatically activated by the detection signals (see Annex II.4).

(b) A pressure relief system, such as rupture disks and connected discharge lines, should be installed in the secondary coolant system to ensure the integrity of primary coolant boundary at the interface with the secondary coolant system as well as the integrity of the secondary coolant system.

(c) Adequate prevention of failure of the steam generator outer shell should be provided (in particular, against wastage effects).

(d) The means for isolation of sodium from steam/water following any failure of a sodium-steam/water interface should be provided (e.g., by installing shutoff valves and relief valves in the water-steam system, and by injection of inert gas).

(e) Accumulation of hydrogen generated by the sodium-water reactions should be prevented to avoid a hydrogen explosion by means of, for example, separation of hydrogen and other sodium-water reaction products, and quick release/reduction of hydrogen (by e.g., vent system and/or hydrogen ignition system).

(f) Potential harmful effects on environment and operators of other sodium-water reaction products and sodium aerosols should also be considered and systems to contain and remove such reaction products should be provided.

(3) Design basis accidents

3.149 Primary-secondary coolant system interface should be kept its integrity under design basis leak accident at steam generator, which may cause unacceptable effects on the primary coolant system due to e.g., sodium-water reaction products.

3.150 The design basis leak should be determined by considering analytical or experimental examination of the physical processes for both leak initiation (see Appendix I.4) and its propagation and by providing substantial margins in the leak magnitude. Propagation of sodium-water/steam interface failure 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 the function of mitigation systems such as leak detection and water/steam automatic shutoff or pressure relief into account in a conservative manner. The determined design basis leak is used as input to design mitigation measures.

3.151 Influence of initial pressure spike propagation due to the steam/water leak and quasi-stationary pressure increase caused by sodium-water reactions should be evaluated by the structural response analysis. Conservative evaluation should be adopted (conservative initial plant conditions, assumption of single failure of the mitigation system, etc.).

(4) Design extension conditions

3.152 The fundamental safety functions should be maintained even under severe sodium-water reactions beyond the design basis leak.

3.153 DEC to be considered should be based on physically possible causes such as multiple failures of the mitigation systems that are not taken into account in the determination of the design basis leak, severe earthquake, external missiles, which could lead to multiple sodium-water/steam interface failures. Prevention measures, such as building arrangement, protection wall against external missiles, can be considered to exclude an event or situation from the causes of multiple failures.

(5) Provisions for inspection and testing

3.154 The system with a sodium-water/steam interface should be designed to enable measures for monitoring and inspection, such as continuous monitoring of water/steam leak, periodic inspections, to check the function of the sodium-water/steam interface to be implemented.

3.155 Mitigation systems of sodium-water reactions should be designed to enable measures for inspection and/or testing, such as periodic calibration of detectors, operational test of valves, and overhaul of rupture discs, to check the function to be implemented.

4. GUIDELINES FOR CONTAINMENT AND ITS ASSOCITED SYSTEMS

4.1. Safety Functions of Containment and its associated systems

4.1.1. Confinement of radioactive substances

4.1 The main requirement for the containment and its associated systems is to envelop those SSCs whose failure could lead to an unacceptable release of radioactive materials to the environment. The containment and its associated systems should include all those components of the reactor coolant boundary and the reactor cover gas boundary and other systems, such as segments of the primary coolant and cover gas cleanup systems that cannot be isolated from the reactor core in accident conditions.

4.2 For operational states, the annual dose received by people living in the vicinity of a nuclear installation is expected to be comparable to the effective dose due to natural background levels of radiation (i.e. the levels that originally existed at the site).

4.3 The structural integrity of the containment and its associated systems is required to be maintained, and the specified maximum leak rate is required not to be exceeded in any condition related to the design-basis conditions for the containment and its associated systems, including any reactor accident that may be included as part of the design basis for the containment and its associated systems. This is required to be achieved by means of containment isolation, management of pressure and temperature, and structural load-bearing capabilities. The management of the radioactive materials should include features to ensure that the release of radioactive materials from the containment and its associated systems is kept below authorized limits.

4.4 Containment isolation features include the valves and other devices that are necessary to seal or isolate the penetrations through the containment envelope, as well as the associated electrical, mechanical and instrumentation and control systems. The design should be such as to ensure that these valves and other devices can be reliably and independently closed when this is necessary to isolate the containment and its associated systems. Table 1 in Appendix I.6. shows examples of isolation valves for piping and ducting systems.

4.5 The pressure and temperature management features should be designed to limit the internal pressures, temperatures and mechanical loading on the containment structure to levels below the design values for the containment structure and for the equipment within the containment envelope. Example of the pressure and temperature management features are sodium combustion control features, thermal capacity by the containment structure, and a free volume in the containment structure.

4.6 The features for the control of combustible gases should be designed to prevent contact of leaked sodium with the concrete and moisture in the containment structure. Typical features for the control of combustible gases are limiting the moisture concentration in the containment structure. Contact of the leaked sodium with concrete in containment structure should be prevented by installing a liner or catch pan. If necessary, inerting in the containment atmosphere for the combustion control of combustible gases is taken into account.

4.7 The features for radionuclide management should operate together with the features for the management of the pressure and the temperature and combustible gases and the containment isolation system to limit the radiological consequences of postulated accident conditions. Typical features for the management of radionuclides are the associated systems such as containment isolation system, inertization system and gas treatment system with activated carbon filters and high efficiency particulate air (HEPA) filters.

4.8 Evaluations of the containment response for pressure and temperature conditions, for combustible gases, and for radionuclide management should be performed considering their relevance to the safety functions of the containment and its associated systems in a conservative or best estimate manner corresponding to the event category.

4.9 In a case where gaseous fission product could be released from failed fuel, heat generation by gaseous fission product should be considered as one of loading on the containment structure.

4.1.2. Protection against internal and external hazards

4.10 The containment and its associated systems should be designed to protect all components of the reactor coolant boundary and the reactor cover gas boundary as well as the safety systems located inside the containment against internal and external hazards challenging from outside the containment.

4.1.3. Radiation shielding

4.11 In operational states and accident conditions, the containment and its associated systems contributes to the protection of plant personnel and the public from undue exposure caused by radioactive materials contained within the containment. Dose limits and dose constraints as well as the application of the ‘as low as reasonably achievable’ principle (for the optimization of radiation protection) should be included in the design basis of the containment and its associated systems. The composition and thickness of the concrete, steel and other structural materials should be such as to ensure that the dose limits and dose constraints for plant personnel and the public are not exceeded in operational states or in the accident conditions that are considered in the design.

4.2. General Design Basis of Containment and its associated systems

4.2.1. Derivation of the design basis

4.12 The design basis for containment and its associated systems should consider all plant states (i.e., any condition arising in normal operation, anticipated operational occurrences, design basis accidents and design extension conditions). Load combinations created by internal and external hazards should also be included in the design basis for the relevant SSCs.

4.2.2. Internal hazards

4.13 Internal events that should be considered in the design of the containment and its associated systems are those events that result from faults occurring within the plant and that may necessitate the performance by the containment of its functions or that may jeopardize the performance of its safety functions. They fall essentially into five categories:

(a) A failure in a sodium-containing system located in the containment

The containment should be able to withstand pressures and temperatures that may be generated by the failure.

(b) A failure in systems or components containing radioactive material located in the containment

The containment should be able to confine the radioactive material.

(c) System transients causing limiting loads (e.g., pressure, temperature and dynamic loads) on the containment structure

The containment should be able to withstand these loads.

(d) Containment bypass events such as intermediate heat exchanger tube failure

Appropriate provisions for confinement capability should be in place.

(e) Internal hazards

It should be verified that the containment functions should not be impaired by internal hazards such as

- Fuel handling accidents in the containment envelope;
- Internal missiles; and

- Internal fires.

4.14 Typical internal events that should be considered in the design of the containment and its associated systems are as follows:

(a) A failure of the reactor coolant boundary or the reactor cover gas boundary; and

(b) Failure (breaks or leakage) within the containment in a transfer system for radioactive liquid or gases.

4.2.3. External hazards

4.15 External events that should be considered in the design of the containment and its associated systems are those events arising from human activities in the vicinity of the plant, as well as natural hazards that may jeopardize the integrity and the functions of the containment. All the events that are to be addressed in the design should be clearly identified and documented on the basis of historical and physical data or, if such data are unavailable, on the basis of sound engineering judgement.

4.16 The postulated relevant external events should be evaluated to determine their possible effects on the functions of the containment and its associated systems, to determine any safety systems needed for prevention or mitigation of the consequences of the external events, and to ensure that the systems are able to withstand the expected effects.

4.17 The containment and its associated systems should be designed to protect the reactor coolant boundary and the reactor cover gas boundary from the postulated natural and human induced external events.

4.18 Typical natural external events and human induced events that should be considered in the design of the containment and its associated systems are given below. Design considerations for the typical events are described below.

Natural external events

- Earthquake
- Hurricane and/or tropical cyclone
- Flood
- Tornado
- Wind
- Blizzard
- Tsunami (tidal wave)
- Volcanic eruption
- Extreme temperature (high and low)
- Forest fire

- Human induced events
- Aircraft crash
- Explosion of a combustible fluid container (e.g., in a shipping accident, an industrial accident, a pipeline accident or a traffic accident)

4.19 The structural integrity of the containment and its associated systems is required to be ensured with appropriate margins, taking into account the loads or combinations of loads originating from the hazards or prevailing in the plant states during which such structures are required to operate. Margins provided by the design of the containment and its associated systems should be adequate, such that the integrity and operability of those systems would be preserved in the event that natural hazards cause loads exceeding those derived from the hazard evaluation of the site.

4.20 The containment and its associated systems should be designed to withstand impacts of missiles generated by internal and external events (including aircraft crash). The protective features of reactor buildings may be considered in missile impact evaluation. In evaluating aircraft crash impacts on the structure, the effects of an aircraft fuel fire should also be included.

4.21 The containment and its associated systems should be designed to maintain its functional requirements against wind loads (i.e., the wind pressure, pressure generated by the difference of atmospheric pressure, and impact loads of the missile) considering site specific conditions.

4.22 The containment and its associated systems should be designed to prevent flooding, e.g., a flood against postulated tsunami height by adopting seawalls and/or watertight construction and to withstand tsunami loads. The design should consider a combination of external events as appropriate for the site, e.g., earthquakes and tsunami loads.

4.23 The containment and its associated systems should be able to withstand external events arising from marine- or industrial-related accidents at neighboring facilities while maintaining the ability to satisfy safety function requirements.

4.24 A nuclear power plant site should be located at sufficient distance from active volcanos and take measures to prevent ingress of volcanic ash.

4.25 If appropriate, external events related to surrounding vegetation, such as fires in forests or grasslands, should be considered to ensure that safety function requirements are satisfied by e.g., siting measures such as physical separation of the containment, and the surrounding vegetation should be taken into account to reduce thermal load on the containment in case of forest or grassland fire.

4.2.4. Design basis accidents and design extension conditions

4.26 Critical load conditions on the containment structure are determined considering DECs, because the conditions for the DBAs such as sodium pipe failure can be relaxed by adopting protective measures e.g., double boundary.

4.27 Events which may cause loads on the containment structure include the following.

(a) Sodium leakage and combustion
(b) Sodium-concrete reaction
(c) Heat generation by gaseous fission product
(d) Hydrogen combustion
(e) Mechanical energy release induced by core melting and re-criticality
(f) Core debris-concrete interaction

Prevention and/or mitigation measures should be taken against all of these events to reduce the uncertainty about conditions in the containment and the containment response. Design load conditions for the containment structure should be determined taking the effects of these prevention and/or mitigation measures into account.

4.2.5. Loading considerations

4.28 In-vessel retention strategy should be applied to reactor core and primary coolant system in order to reduce the potential loading on the containment structure.

4.29 For a DEC, sodium leakage and combustion should be considered as one of loading on the containment structure in the postulation of sodium leakage caused by multiple failures of coolant system boundary. In addition, if the leakage may lead to sodium-concrete reaction followed by hydrogen generation and accumulation, hydrogen combustion should also be considered as one of loading on the containment structure.

4.30 In a case where gaseous fission product could be released in the process of core damage sequences, heat generation by gaseous fission product should be considered as one of loading on the containment structure.

4.31 The application of In-vessel retention strategies should reduce the following potential loading on the containment structure:

(a) Mechanical energy release induced by core melting and re-criticality which should be prevented by taking preventive measures against severe re-criticality in core damage sequences resulting from an unprotected transient; and
(b) Core debris-concrete interaction which should be prevented by design measures for in-vessel retention in core damage sequences resulting from an unprotected transient and by practical elimination of complete loss of heat removal function and core uncovering due to sodium inventory loss.

4.3. Design of Containment and its associated systems against Accident Conditions

4.3.1. Control of pressure and temperature

(1) Gaseous fission product heat generation

4.32 In case gaseous radioactive materials are released into the containment structure, increase in pressure and temperature in the containment structure by decay heat may become loading on the containment structure. The containment function should accommodate the heat load generated by the gaseous radioactive materials.

(2) Sodium leakage and combustion

4.33 The containment structure should be designed to prevent mechanical failure caused by over pressurization due to sodium combustion from any sodium leakage and to prevent thermally-induced failure of the containment structure.

4.34 Pressure and thermal loads on the containment structure caused by sodium combustion in the containment structure atmosphere should be evaluated. In addition, mitigation measures applying one or a combination of the following, depending on the situation, should be provided to the containment structure.

(a) Design measures to prevent contact between sodium and concrete, and to collect the leaked sodium to reduce the possibility of further reactions, e.g., installation of catch pans, pits, retention tanks, and transportation pipes to prevent sodium-concrete contact, and to retain leaked sodium in limited areas. In addition, sodium should not be mixed with other highly reactive liquids during collection of leaked sodium.

(b) Design measures to minimize sodium combustion by making the atmosphere in the containment structure essentially inert against sodium combustion, e.g., lowering the oxygen concentration by using inert gas.

4.35 Structures that may have contact with or retain leaked sodium should be designed considering the thermal load associated with the sodium, including combustion in addition to influence of fission product on the systems. Effects of increased concrete temperature, i.e., water release from concrete, should be considered as needed.

4.36 The design should subdivide and potentially inert compartments where sodium leakage can occur to limit the available oxygen that can react to leaked sodium. The degree of this subdivision and the design of the segregation features may vary depending on the operational state, the expected radioactivity of the leaked sodium and the challenge that the sodium release and combustion would place on the compartment segregation features.

4.37 To reduce the effect of sodium combustion on the temperature of the containment structure, installation of thermal insulations or cooling systems to the containment structure and its internals should be considered taking account of the effects of leaked sodium and the combustion on the structures as necessary.

4.38 Considerations of sodium leakage should include the potential for sodium to leak from any part of the systems containing sodium, e.g., not only the major pipes and the components but also from small penetrations of the sodium-containing boundary such as those associated with instrumentation lines for thermocouples, neutron detectors, and failed fuel detectors.

4.39 The safety systems located inside the containment should be designed to prevent functional failure by aerosol deposition and/or corrosion due to interaction with reaction materials generated by sodium leakage and combustion.

4.3.2. Control of radionuclides

(1) Containment source term

4.40 To assess the overall containment performance and in particular the measures for radionuclide management, the amount and isotopic composition of the radionuclides postulated to be released from the containment (the source term) should be assessed for the various accidents to be considered. For DBAs, this should be done by means of a conservative analysis of the expected behaviour of the core and of the safety systems. Consideration should be given to the most pessimistic initial conditions for the relevant parameters (e.g., for the inventory of radionuclides in systems and for leak rates) within the framework of the allowable limits specified in the technical specifications for the plant.

4.41 The anticipated evolution of the physicochemical forms of the radionuclides in the containment should be assessed.

(2) Leaktightness of the containment

4.42 An effective way to restrict radioactive releases to the environment is to maintain the leak rate below conservatively specified limits throughout the plant’s operating lifetime. As a minimum, leak rates should be small enough to ensure that the dose and/or dose rate limits are not exceeded during normal operations or in accident conditions.

4.43 At the design stage, a target leak rate should be set that is well below the safety limit leak rate, i.e., well below the leak rate assumed in the assessment of possible radioactive releases arising from accidents.

4.44 To limit the number of potential leak paths, the number of penetrations should be kept as low as possible. The external extensions of the penetrations should be installed in the secondary confinement building, at least until the first isolation valve, in order to collect and filter any leaks before a radioactive release occurs.

4.45 Leak rates of isolation devices, air locks and penetrations should be specified with account taken of their importance to safety and the integral leaktightness of the containment envelope.

(3) Deposition on surfaces

4.46 In typical SFR design, reduction of radionuclide in the containment atmosphere is expected by deposition effects on the surface of the internal structures and inner wall of containment structure and/or by ventilation systems with filters.

4.47 The containment structure and its internals can be the primary means for the removal of released radioactive materials because of their wide surface area for the absorption. Coefficients of deposition and desorption of the radioactive materials should be calculated based on the available best knowledge about the surface absorption and in a conservative manner.

(4) Secondary confinement building

4.48 In a case where secondary confinement building is provided outside the containment structure, the following should be considered.

(a) The objective of the secondary confinement building is collection of the leaks and release of filtered materials via vent stacks, not taking over the containment functions if the containment envelope fails.

(b) The systems associated with the secondary confinement building should be designed to collect, filter, and discharge gases and radioactive materials leaked from the containment envelope in accident conditions or to pump those leakages back into the containment envelope.

(c) A filtered ventilation system should be provided to maximize the efficiency of the secondary confinement. The ventilation system should quickly reduce the pressure in the secondary confinement building to negative pressure against atmospheric pressure after initiating events. The negative pressure should be kept even under the most severe external environment (e.g., wind loads).

(d) The secondary confinement building should be designed to prevent direct leakage from the containment envelope (direct leakage to outside from the containment envelope without passing through the secondary confinement building) as much as possible.

4.3.3. Control of combustible gases

(1) Hydrogen combustion (Prevention of hydrogen generation and combustion)

4.49 Sodium leakage can cause the formation of a hydrogen-air mixture in the containment atmosphere as a result of the following phenomena. All these contributing phenomena should be evaluated.

(a) Interaction between leaked sodium and water in the containment.

(b) Interaction between leaked sodium and concrete in the containment.

4.50 The possibilities of hydrogen generation and combustion resulting from sodium leakage should be evaluated. It should be confirmed that hydrogen generation and combustion in the containment envelope are prevented. In addition, if the contact of sodium with concrete cannot be eliminated, countermeasures should be provided to reduce the volumes generated and the risks of deflagration to as low as reasonably practicable.

(2) Management of water and other liquids and gases in the containment

4.51 Systems containing water (or any other liquid or gas) that are installed in the containment should use multiple barriers to reduce the possibility of contact of sodium with water (or other liquids or gases). Any liquid or gas used in equipment in the containment should be evaluated and necessary measures should be implemented taking account of any potential interactions of the liquid or gas with sodium.

4.52 In a case where a steam venting system, which releases steam from concrete heated as a result of sodium leakage or from any other causes, is included in the design, it is required to discharge the steam to areas other than any compartments containing sodium. Increased pressure and behavior of condensed water due to the discharge should be considered in the design.

4.3.4. Mechanical features of the containment envelope

Design of isolation valves and containment function of a secondary coolant system should follow the recommendations below.

4.53 Each line penetrating the containment envelope should have at least two isolation valves arranged in series. This requirement applies to any line that is either (a) connected to the reactor cover gas boundary or (b) connected directly to the containment envelope atmosphere. Each valve either should be normally closed or should have provisions to close automatically when required for containment isolation, and should be reliably and independently actuated. Isolation valves should be located as close as practicable to the containment envelope to minimize the potential for failure of the line outside of the containment envelope.

4.54 Loops that are closed either inside or outside the containment envelope should have at least one isolation valve outside the containment envelope at each penetration.

4.55 Exceptions to the recommendations for containment isolation stated above should be permissible when the following three cases of containment isolation is not considered to be reasonably practicable.

(a) Small tubes such as measuring instruments

The leak amount of coolant and/or radioactive substances can be limited within acceptable amount due to small opening.

(b) A line of secondary coolant system which has interface with the primary coolant system

It forms a closed loop in the containment envelope. The leak amount of primary coolant and/or radioactive substances can be limited within acceptable limit in case of the primary-secondary coolant interface (typically, heat exchange tube in the intermediate heat exchanger) failure by means of liquid sodium sealing effect[11], provided that the pipes and components including the primary-secondary interface of the secondary coolant system inside the containment envelope are also defined as containment envelope and its integrity can be confirmed by monitoring the condition or periodical inspection. Also, the primary-secondary interface failure should be detectable.

(c) A line that potentially reduces the reliability of the decay heat removal function by applying isolation valves and that the above countermeasures are taken.

4.56 Performance of containment isolation valves and related devices should be ensured even under the severest conditions expected.

4.4. Tests and Inspections

The containment and its associated systems should have provision for conducting the following commissioning tests and in-service tests and inspections.

(1) Commissioning tests

4.57 Commissioning tests, including structural integrity test, integrated leak tests, local leak tests of isolation devices, for the containment should be carried out prior to the

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[11] *Liquid sodium sealing effect: Effect that can limit radioactive substance release to the environment in case of the primary-secondary interface failure (a) by keeping the pressure of the secondary coolant system higher than that of the primary side in the normal operation and (b) by retaining the radioactive substances in the secondary coolant after reaching equilibrium pressure between the primary and the secondary sides.*

first criticality of the reactor to demonstrate the structural integrity of the containment, to determine the leak rate of the containment envelope and to confirm the functioning of related equipment.

(2) In-service tests and inspections

4.58 Periodic in-service tests and inspections, including structural integrity test, integrated leak tests, functional tests of the equipment and visual inspection, should be performed to demonstrate that the containment and its associated systems continue to meet the requirements for design and safety throughout the operating lifetime of the plant.

4.59 Where it is technically feasible, the design should provide for a complete visual inspection of containment structures, penetrations and isolation devices.

I. APPENDIX

I.1. Fuel Characteristics (oxide, metal, and nitride fuels)

Metallic fuels composed of uranium, plutonium, or their alloys were used in the early stage of sodium-cooled fast reactor development. However, their burnup was limited due to excessive swelling. Oxide fuels composed of mixed uranium and plutonium dioxide have been developed and experience of irradiations has been accumulated since 1970s. Meanwhile, U-Pu-Zr alloy fuel was developed in the 1990s based on the vast experience of U-Fs and U-Zr fuels. High burnup potential of metallic fuels is evaluated to be comparable to oxide fuels. Development of nitride fuel and carbide fuel has also been conducted for the SFR system [I-1] [I-2].

Achieving high burnup is essential to bring out the economic competitiveness of an SFR. Demonstration of stable fuel behavior up to high burnup needs to take the following characteristics of different fuel types into account.

(1) Oxide fuel (U, Pu) O_x

The oxide fuel is made up of a large number of short pellets. Uranium dioxide has a cubic crystal structure and is isotropic as far as its physical properties are concerned. It can exist over a wide range of composition, i.e., it can accommodate an excess or a deficiency of oxygen atoms and properties vary with the oxygen-metal ratio. UO_2 and PuO_2 have similar properties and are mutually soluble. They have high melting points (UO_2 = 2,800 degree C), low thermal conductivities (0.023 W/cm/C) and good chemical stability. The low thermal conductivity gives rise in operating fuel elements to high thermal gradients from the fuel surface to the center, often reaching over 2,000 degree C at the center. This gives rise to fuel restructuring during irradiation and to the migration of fission products down the thermal gradient. Thus, oxides tend to initially release their gaseous fission products rather than swelling. In the evaluation of the temperatures of fuel pellets in operational states, account should be taken of the changes in the thermal conductivity of the pellets and in the thermal conductance of the gap between pellet and cladding due to burnup dependent effects such as oxide densification, swelling, accumulation of fission products and other changes in the microstructure of pellets. The increase of oxygen potential and the change in the chemical balance of fission product will favor chemical reactions of some fission product such as cesium with chromium oxide. Simultaneously, other fission product as volatile tellurium and iodine, made free by the association of cesium with chromium or molybdenum can chemically react with the major components (Fe, Cr, Ni) of the cladding.

(2) Metal fuel (U-Pu-Zr)

The metal fuel is made up of a fewer number of long slugs. The metallic fuel has a higher thermal conductivity (0.22 W/cm/C) and low melting point (1,160 degree C). To accommodate fuel swelling upon irradiation, the metallic fuel design features an as-fabricated smear density of 75%. Because the metal fuel alloy is chemically compatible with sodium, the radial gap of fuel rod is filled with liquid sodium inside the cladding. Higher thermal conductance across the bond sodium and higher thermal conductivity of metal fuel causes relatively small radial temperature gradients over the fuel rod [I-3][I-4][I-5]. Metallurgical inter-diffusion occurs between the fuel alloy and the stainless-steel cladding during normal operation. A liquid phase due to fuel-cladding eutectic reaction may be penetrated into the cladding at a higher temperature under transient conditions [I-6].

(3) Nitride fuel (U, Pu)N

Nitride fuels were identified as candidates for SFR, nearly three decades back, on the basis of their attractive physical and chemical properties e.g., a high heavy metal density, a strong thermal conductivity connected with a high melting temperature (>2,700°C) as well as a good compatibility with stainless steels and sodium and aqueous reprocessing. Due to the attractive physical properties, improved performances of nitride fuel core such as a larger breeding ratio and higher linear heat rates (in comparison to oxide fuel) will be expected. These features will serve large safety margin in operational states, although fuel dissociation of (U, Pu) N fuels, whose temperature is substantially lower than the melting point if nitrogen overpressure is not maintained, has been identified as a critical issue in case of severe accidents. The relative drawback of nitride fuel is neutron absorption in $^{14}N(n,p)^{14}C$ reaction causing some deterioration of neutron balance and formation of carbon ^{14}C with long half-life. Nitride fuel has fairly extensive and adequate performance demonstrated at the pin, sub-assembly and assembly level for driver fuel. With regard to high burnup and transmutation capability, no critical issue has been identified to-date but this technology is in the early stage of the development.

I.2. Mechanical Design of Fuel Assemblies

A typical design of fuel assembly and fuel element of an SFR is shown in Figure I-1. Typically, oxide, metal, nitride, and carbide fuels are employed in SFRs. This Appendix describes the mechanical design of the fuel assembly for oxide and metal fuel.

(1) Fuel assembly for oxide fuel

The fuel assembly for oxide fuel contains numbers of fuel elements (called a fuel bundle) and neutron shields at upper and lower parts of the fuel bundle in the hexagonal wrapper tube. The entrance nozzle and handling head are installed in lower part and upper part of the wrapper tube respectively. The fuel element may contain core fuel pellets and upper and lower blanket fuel pellets filled with helium gas in a cladding. The fuel elements are densely arranged in the form of an equilateral triangle in order to increase the fuel volume fraction. A wire wrap type spacer winds around each fuel element in a spiral to keep the spacing between the fuel elements. Figure I-1 shows the example of the wire wrap spacer. Besides the wire wrap spacer, there is an example of honeycomb type grid spacer located at specified axial levels.

The entrance nozzle has orifices on the outer surface in six directions to control the coolant flow in combination with core support structure and to prevent simultaneous blockage of the inlet holes by obstacles. The space between bundle of wired fuel elements and inner surface of the wrapper tube should be kept to have an appropriate distance so that excessive flow-induced oscillation will not cause fretting.

The handling head is arranged at upper part of the wrapper tube and it should have a configuration which can be grasped by the refuelling machine in loading, drawing out and shuffling the fuel assemblies.

The entrance nozzle and the handling head preferably have self-orientation mechanisms that enable to load fuel assemblies easily and automatically by adjusting the angle of adjacent assemblies. The neutron shields installed at the upper and lower parts of the fuel bundle suppress the fast neutron irradiation at the upper core structure and the core support plate. Pads should be placed at the handling head and the wrapper tube to keep a distance to the adjacent assemblies.

The main issues that need to be addressed in the fuel element design are: high-burnup structure; fission-product migration, pellet-cladding chemical interaction (FCCI), and pellet-cladding mechanical interaction (FCMI). The fuel element has a reservoir (gas plenum) to prevent increase of inner gas pressure. In case of fast reactor fuel, inner gas pressure in the fuel element can be high because fission gas release rate is large, and the fuel is used until high burnup. From the beginning of fuel lifetime, inner gas pressure in the fuel element is higher than outside coolant pressure because coolant sodium, whose boiling point temperature is high, remains at several atmospheric pressures. There is a possibility that cladding has creep failure due to pressure difference between inner gas and outside coolant of the fuel element. For this reason, the length of gas plenum should be set to prevent excessively high inner gas pressure in the fuel element. The cladding material is stainless steel who exhibits a superior creep resistance at high temperature. The position of gas plenum is on the top or bottom of the stack of fuel pellets. In the latter case, length of gas plenum is shortened because the temperature of gas plenum is lower.

Oxygen is excessively generated in the fuel pellet as nuclear fissions progress for oxide fuel. The cladding inner surface can corrode due to the excessive oxygen, iodine, and a part of fission products such as Cesium. The outer surface of cladding may also corrode slightly by e.g., dissolved oxygen in sodium. To evaluate stress on cladding, reduction of cladding thickness due to corrosion should be taken into account.

Prevention of fuel melting in normal operation and abnormal operational transient needs to be considered because fuel pellet temperature is relatively high in a fast reactor because power density is high. There is helium gas and fission gas in gap between the fuel pellet and inner surface of cladding. The gap should not be too large considering influence on heat transfer rate because thermal conductivity of gas is relatively low.

In the fuel pellet, remarkable structure changes occur due to pore migration based on evaporation and condensation mechanism caused by large thermal gradient in a radial direction. The central hole in a large diameter, generated after the structure changes by the mechanism, has an effect of suppressing the maximum temperature of the fuel pellet. To prevent excess fuel pellet cladding mechanical interaction (FCMI) and to increase linear heat rate, annular fuel pellets are adopted in some of the fast reactors.

Regarding fission products generated in the fuel pellet, most gaseous products are released, however solid products remain in the pellet. The fuel pellet is swelled due to accumulation of such fission products. Such pellet swelling causes the FCMI. Considering that the fuel pellet of fast reactor is in a high temperature and tends to be creep-deformed, the smear density in the fuel element should be appropriately adjusted to prevent excessive FCMI.

The relocation of fuel pellet fragment produced by the pellet cracking due to a temperature gradient in radial direction occurs during irradiation. This fragment relocation toward cladding inner surface used to affect the excessive FCMI taking place during irradiation in LWR, however, in fast reactors, the cladding hoop stress caused by FCMI can be mitigated by higher creep rate of fast reactor fuels due to its higher temperature than LWR fuels. Also, the fuel fragment relocation in fast reactors is expected to improve a heat transfer in a gap between fuel pellet and cladding, and might allow to prevent the fuel melting even in a transient event. Furthermore, the results of irradiation experiment with annular pellets in EBR-II indicated that an influence of fuel fragments dropping into a central-hole on the increase of fuel temperature and cladding stress caused by FCMI would be limited and negligibly small.

The cladding has swelling by voids, accumulating defects formed by fast neutron in metal crystal structure. If the cladding is excessively swelled, cooling performance deterioration, embrittlement of cladding material, and bundle-duct interaction (BDI) become problems. Because of this, a material with high resistance property to swelling should be selected for cladding.

Some of factors that determine the lifetime of the fuel assemblies include the behavior of the fuel bundle in the wrapper tube and the interaction between the fuel assemblies. An example of the former is the BDI. The BDI is caused by the difference in the swelling of the fuel bundle and the wrapper tube according to fast neutron irradiation. If the swelling of the fuel bundle is very large, the local temperature rises because the spacing between the fuel elements is reduced. Therefore, it is important to limit the dose with sufficient margin by understanding the temperature dependence and fast neutron irradiation dose dependence on the swelling of the material.

The interaction of the assemblies occurs by the expansion and the bending of the wrapper tubes. The expansion of the wrapper tube results from overlapping of the isotropic swelling of the wrapper tube material and the irradiation creep deformation due to coolant pressure in the wrapper tube. The bending of the wrapper tube occurs due to the face-to-face differences of the thermal expansion and the swelling caused by the radial temperature and fast neutron flux distributions in the wrapper tube. Because of the expansion and the bending, the interference occurs between the adjacent wrapper tubes and the contact load is generated at the pads. The irradiation creep deformation occurs in the wrapper tube so as to mitigate this load. In order to avoid the excessive contact load, it is important to set the appropriate spacing between the adjacent wrapper tubes and to install the pads on the wrapper tube and the handling head in the appropriate position to keep the space. In addition, the bending of the wrapper tube due to swelling and irradiation creep remains even when the reactor shutdown, which is important in the evaluation of the handling load at the time of refuelling.

From the above, an excellent material having resistance to swelling characteristics should be selected for core materials e.g., wrapper tube, cladding. Development of such materials has been conducted to aim for the high burnup.

(2) Fuel assembly for metal fuel

The metal fuel is injection cast as binary (U-Zr) or ternary (U-Pu-Zr, U-TRU-Zr) alloys of long rods. The metallic alloy is stabilized typically using 5 to 30% addition of zirconium to increase the melting point, improve the structural strength, and minimize the potential for fuel/cladding chemical interaction (FCCI). The fuel is thermally bonded to the cladding using sodium inside the cladding, providing a high thermal conductivity medium to facilitate almost unimpeded transfer of the heat generated in fuel to the cladding and reactor coolant. Most metallic fuel forms maintain a low fuel-smeared density at or below 75% to provide room for early swelling and development of interconnected porosity to allow the escape of fission gases from the fuel matrix and to accommodate fuel swelling at higher burnup.

To assure adequate performance, metal fuel designs should consider applicable failure modes depending on various irradiation effects such as fuel-alloy constituent redistribution, porosity formation, fission gas retention and release, irradiation-induced radial and axial swelling of fuel slugs, and formation of low-melting temperature eutectic formation at the fuel-cladding interface resulting in a gradual thinning of the cladding. Since the primary failure mode for the conventional metal fuel forms is fission gas induced breach of the cladding that is weakened due to eutectic thinning (FCCI), the design should consider the impact of this slow process at the operating temperatures, as well as the accelerated cladding failure at elevated temperatures (below the fuel and cladding melting points) during postulated accidents. The impact of low melting temperature of the metallic fuel (lower than that for the oxide fuel and the melting point of the cladding material) should also be factored in the design to limit the coolant outlet temperature and assure acceptable transient fuel performance.

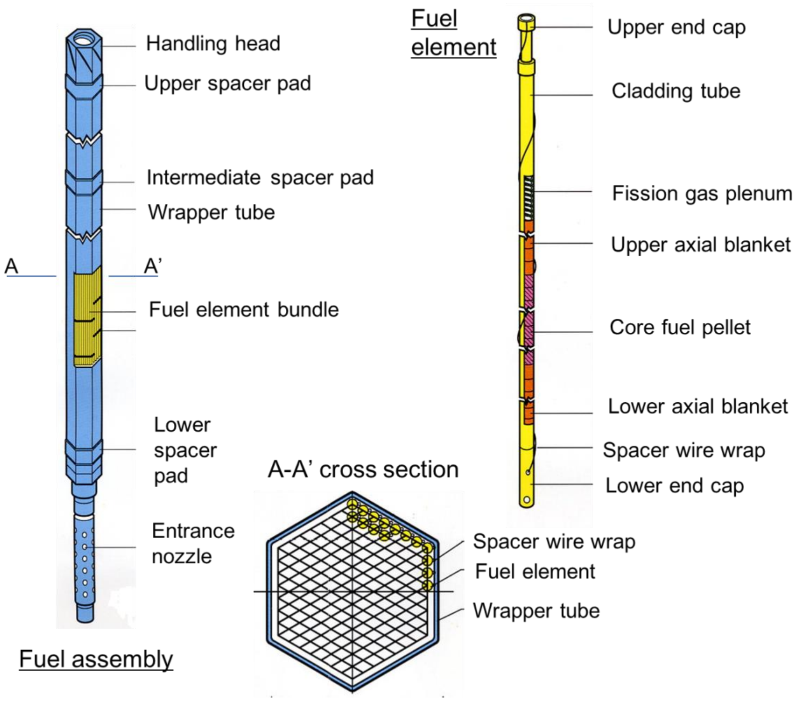


Figure I-1 Typical SFR core fuel element and fuel assembly

I.3. Reactivity Coefficients

One of the important features of the behavior of the reactor core in an Anticipated Operational Occurrence from steady state power operation is the rate at which the transient progresses. This rate depends on the combined effects of the core nuclear characteristics discussed in this Appendix. The factors of importance are

- The total power coefficient of the core;
- The power/flow coefficient of the core;
- The isothermal temperature coefficient of the core;
- The delayed neutron fraction; and
- The prompt neutron lifetime.

The total power coefficient is defined as the change in reactivity with normalized power change. The power/flow coefficient is defined as the change in reactivity with normalized power to normalized core coolant flow ratio change. The isothermal temperature coefficient is defined as the change in reactivity with core inlet coolant temperature change. The three coefficients above determine reactivity behavior in transients because transient events would influence the reactor core with abnormal power, abnormal power-to-flow ratio, and abnormal core inlet coolant temperature [I-7].

Listed below are typical phenomena that cause reactivity change in the core [I-8]. Combination of these phenomena provides the three coefficients. Due consideration should be paid primarily for uncertainty and change of the coefficients during operation for application of these effects to the design and evaluation. Moreover, these should be treated with care so that conservative results can be obtained. In general, reactivity components resulting in temperature change in the core or density change in reactor core materials, i.e., Doppler effect, coolant density coefficient, axial and radial expansion, and control-rod driveline expansion are taken into account with conservative application of the uncertainty under anticipated operational occurrences and design basis accidents (DBAs). When utilizing the effects of core axial or radial expansion and control rod drive line expansion under design extension conditions, these effects should be quantified with implementation of design provisions for enhancing these effects.

- Doppler Effect;

  Owing to the increase in the neutron resonance absorption cross-section of 238U with temperature, the temperature coefficient of the fuel is normally negative. Different fuel types have different magnitudes of fuel Doppler feedback, e.g., the absolute value of the Doppler coefficient for oxide fuel is larger than for metallic fuel because of the presence of oxygen molecules and a softer neutron spectrum.

- Coolant Density;

An increase in coolant temperature reduces the coolant density, reducing the moderating and reflecting effects of the sodium coolant. The reactivity effect of a coolant density reduction varies across the core, being more positive in the core interior where the moderating effect is dominant and negative at the core boundaries where the leakage effect is more important. Depending on design choices affecting the relative importance of moderation and leakage, the reactivity feedback from increases in coolant temperature (reduction in coolant density) can be either positive or negative.

- Fuel Axial Expansion;

  An increase in fuel temperature causes the fuel to expand axially based on the thermal expansion coefficient of either the fuel or cladding, or both, depending on the chemical and stress conditions at the fuel/cladding interface. The fuel axial expansion decreases the fuel density at active core region and increases the radial neutron leakage at overall active core region, thus introducing negative reactivity.

- Core Radial Expansion;

  Fast reactor cores have a significant neutron leakage fraction due to the large neutron mean free path length, resulting in a large gradient of fuel reactivity worth at the edges of the core, making the reactivity of the core sensitive to changes in core geometry. If core assemblies move outward in the radial direction, increasing the effective diameter of the core and moving fuel from a region of higher worth to one of lower worth, negative reactivity feedback is generated. Conversely, if the core assemblies move inward, positive reactivity feedback is generated.

- Control Rod Driveline Expansion;

  The relative motion between the core and the control rods is caused by a change in temperature of the control rod drivelines (CRDLs) and by changes in the temperature of the reactor vessel. The drivelines are normally located at the coolant outlet, or hot plenum, and respond to changes in core outlet temperature. An increase in CRDL temperature will cause the control rods to move further into the core, introducing negative reactivity feedback.

I.4. Sodium-Water Reaction

Sodium is chemically active. It reacts with air and water, generating corrosive reaction products, hydrogen, and reaction heat. In SFRs with steam turbines for power conversion, a failure of a boundary between sodium and water (e.g., heat exchange tube in steam generator (SG)) would cause injection of high-pressure water/steam into sodium, resulting in sodium-water reactions. Following phenomena should be taken into account in the design of SGs and detection/mitigation systems.

When reusing the components after accidents (a failure of a boundary between sodium and water, sodium-water reaction), their integrity and performance should be confirmed before use. The sodium-water reaction products may cause stress corrosion cracking (SCC).

(1) Enlargement of failure due to propagation (Thermochemical effect)

High temperature and corrosive jet generated by sodium-water reactions 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 as shown in Figure I-2. The wastage type failure is synergistic tube thinning of corrosion and erosion which is caused by sodium-water reaction jet. In the wastage type failure, there are two types of tube wastage. Target wastage is that high temperature and corrosive jet from an initial failure site hits an adjacent tube and makes it thin and fail in a smaller leak event. Multi-wastage is that a larger leak affects some of adjacent tubes. As a result of such kind of failure propagation, if no mitigation action is taken, 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 wastage type failure. Therefore, it is important to prevent the overheating type failure by means of quick detection and mitigation of the failure propagation.

In order to minimize the failure propagation, leak detection and reaction mitigation in the failure component are essential. Following points should be considered in the design of SGs and detection and mitigation systems.

- Failure propagation and detection time depend on the initial leak rate.
- Small leaks, typically a pinhole, take longer time to propagate and also take longer time to detect.
- Large leaks, typically a double-ended break, can be detected and mitigated by actuation of rupture disks in a short time.

- Middle leaks in between small and large leaks can cause the overheating type failure since the temperature rise around the SG tubes is faster than that of small leaks.

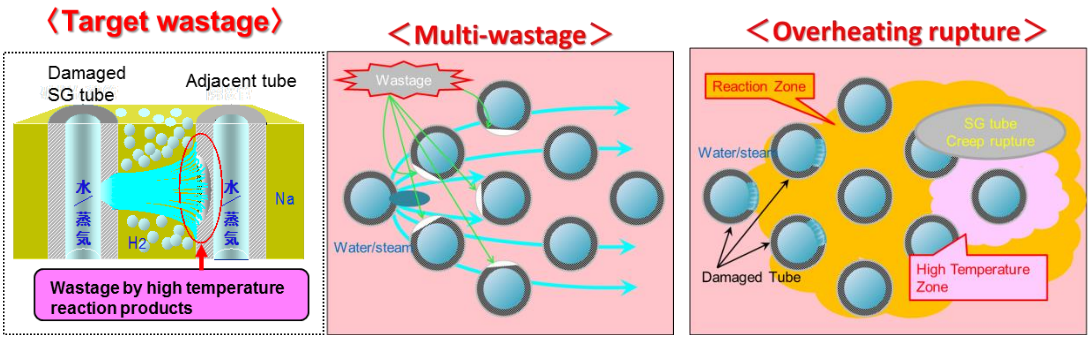


Figure I-2 Wastage type and overheating type failures of steam generator tubes

(2) Influence on the components and piping in the secondary coolant system (Mechanical effect)

When the failure area is large, the pressure in the secondary coolant system, which includes primary/secondary coolant interface in the intermediate heat exchanger, increases typically due to initial spike-shaped pressure and quasi-steady pressure as shown in Figure I-3. The former is produced just after the heat exchange tube failure in a SG, while the latter is produced for relatively longer duration in the failure propagation process. It could lead to failures of the secondary coolant system components and piping. The heat exchange tubes of the intermediate heat exchanger should not fail against these pressure loadings. The images of these pressures are shown below.

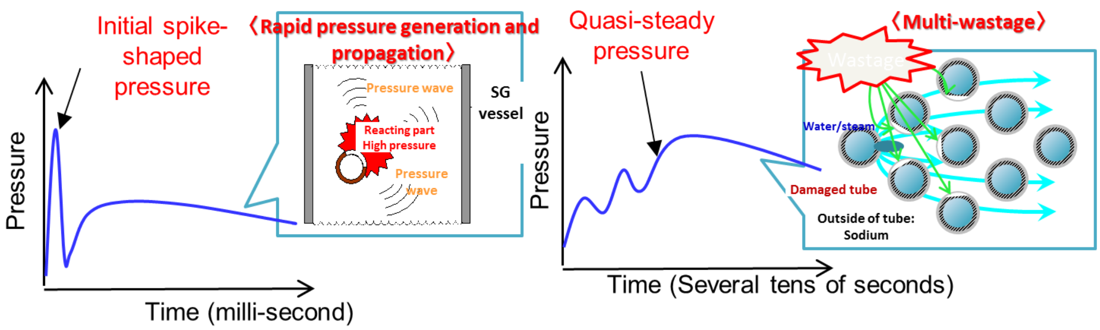


Figure I-3 Pressure in secondary coolant system due to sodium-water raction

I.5. Decay Heat Removal System

The safety of SFRs is achieved by decay heat removal functions having highly reliable independency and diversity. Normal decay heat removal is usually accomplished via balance-of-plant (BOP). For a steam turbine power conversion system, a bypass line which diverts steam flow from the turbine to heat sink can be used for the decay heat removal. BOP is usually not a safety-grade system. In the event BOP path is not available, decay heat removal should be achieved via safety-grade Decay Heat Removal Systems (DHRS).

Features of typical DHRSs in SFRs are described below. Taking the feature of the reactor design and the postulated accident conditions into account, DHRS should be composed by proper combination of the following sub-systems and the alternative cooling measure should be adequately selected.

(1) DRACS (Direct Reactor Auxiliary Cooling System)

A DRACS consists of a heat exchanger (DHX) installed in a Reactor Vessel (RV) and its secondary coolant loop which transfers decay heat from the primary coolant to an Ultimate Heat Sink (UHS), generally atmosphere, as shown in Figure I-4. DRACS may be installed in the hot pool or the cold pool. Depending on the position and capacity of DHX, primary coolant flow and temperature distribution may change from the normal operation condition in RV. The DRACS should be designed that such change does not jeopardize natural circulation capability of the primary coolant system.

The features of the DRACS compared to the other types of DHRSs are as follows:

- Available in case of failure of a primary coolant circuit for loop-type design and in case of failure of normal primary coolant flow between hot plenum to cold plenum for pool-type design,
- Inter-wrapper sodium flow can be utilized for enhancing the passive heat removal capability if DHX is installed in the hot plenum.
- Radiation shielding of the DHX is needed.

(2) PRACS (Primary Reactor Auxiliary Cooling System)

A PRACS consists of a heat exchanger (PHX) installed in a primary coolant circuit for loop-type design and in a main flow stream path for pool-type design, and its secondary coolant loop which transfers decay heat from the primary coolant to a UHS as shown in Figure I-4.

The features of the PRACS compared to the other types of DHRSs are as follows:

- Suitable natural circulation capability of the primary coolant due to a large elevation difference between the core and the PHX,
- Not available in case of failure of a primary coolant circuit for loop-type design, and in case of failure of normal primary coolant flow between hot plenum to cold plenum for pool-type design
- The necessity for radiation shielding of the PHX is taken into account for pool-type design.

(3) RVACS (Reactor Vessel Auxiliary Cooling System)

An RVACS transfers decay heat from the primary coolant in RV through a Guard Vessel (GV) to a UHS as shown in Figure I-4. The heat transfer from GV outer surface to UHS is accomplished by atmospheric air or dedicated liquid cooling circuits. A heat exchanger in the primary coolant is not needed, and heat transfer between RV and GV is conducted by radiation heat transfer. Natural circulation between the core and RV wall should be established inside RV for its operation. Air flow path is provided surrounding GV so that natural circulation capability can be utilized for air cooling RVACS.

The features of the RVACS compared to the other types of DHRSs are as follows:

- Enhanced diversity in combination with other types of DHRSs
- The heat removal capacity is small, especially for large SFRs, because the heat transfer is limited by the radiation heat transfer between RV and GV, and by the heat transfer area of the outer surface of GV.

(4) IRACS (Intermediate Reactor Auxiliary Cooling System)

An IRACS consists of an auxiliary cooling loop branched from a main loop of a Secondary Coolant System (SCS) and a heat exchanger installed in its loop which transfers decay heat from the primary coolant to a UHS as shown in Figure I-4. The IRACS removes decay heat from the primary coolant via the IHX. The number of the IRACSs installed in an SFR depends on the number of the main loops of the SCS. In order to change the coolant flow from the SG to the IRACS, valves are provided in the main loop of upper stream and downstream of SG.

The features of the IRACS compared to the other types of DHRSs are as follows:

- Flexible in the location of installation,
- Available in natural circulation capability,
- Not available in case of failure of flow path of the PCS and/or the SCS,
- The SCS should be designed as safety-grade.

(5) SGACS (Steam Generator Auxiliary Cooling System)

A SGACS is classified into two types. The first type is that water passes inside heat exchanger tubes in the SG for cooling. Decay heat is removed by an auxiliary cooling loop branched from a main loop of a steam and feed water system (SFWS) as shown in Figure I-4. The second type is that wind blows around the SG for cooling. Decay heat is removed by e.g., dedicated cooling circuits installed around the SG.

The features of the SGACS compared to the other types of DHRSs are as follows:

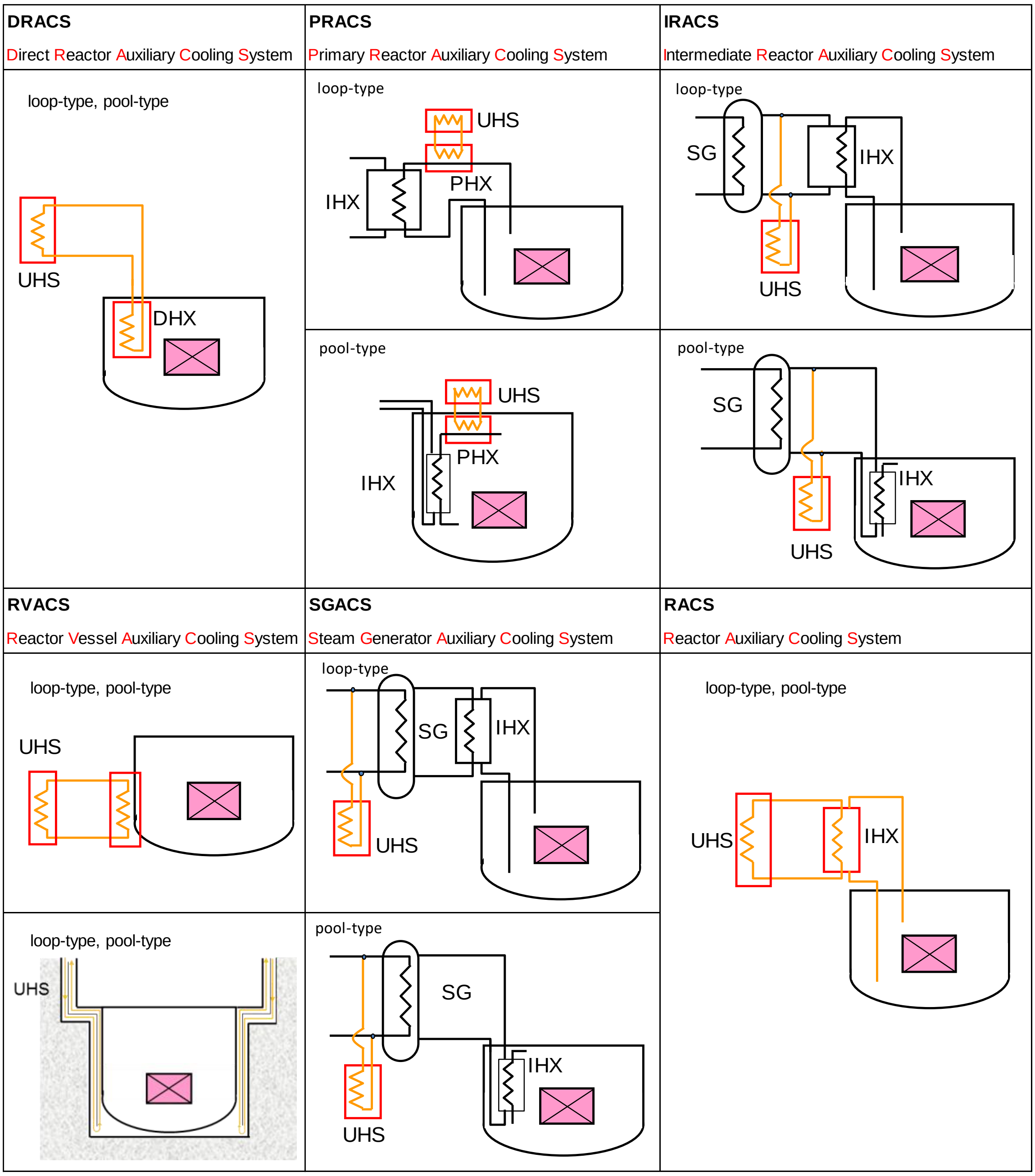
- Ease of access to the SGAHRS for the maintenance,
- Not available in case of failure of circuit of the PCS and/or the SCS,
- The SCS should be designed as safety-grade.
- The SFWS or the dedicated cooling circuits should be designed as safety-grade.

(6) RACS (Reactor Auxiliary Cooling System)

A RACS is a set of auxiliary cooling circuits for decay heat removal, which is independent of the main heat transport systems. This system connects to RV by dedicated primary piping and has dedicated IHX and secondary coolant system.

The features of the RACS compared to the other types of DHRSs are as follows:

- Independent of the main heat transport systems and other decay heat removal systems
- Available in natural circulation capability



DHX : heat exchanger of DRACS	IHX : intermediate heat exchanger
PHX : heat exchanger of PRACS	SG : steam generator
UHS : ultimate heat sink	

Orange and red lines show decay heat removal systems.

Figure I-4 Concept of typical decay heat removal system[12]

[12] *This figure indicates a concept of typical decay heat removal system but does not indicate detailed design such as elevation.*

I.6. Configuration Examples of Isolation Valves

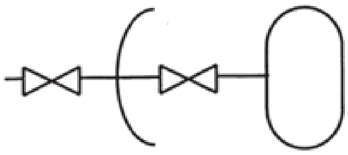
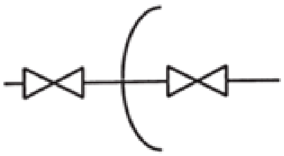
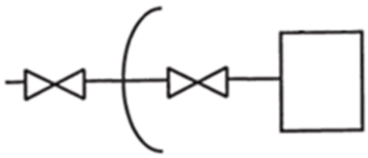
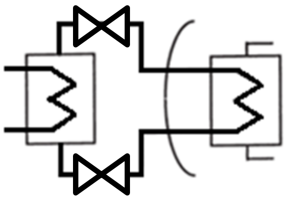
Table 1 elaborates on means of isolation for piping and ducting systems.

Each line penetrating the containment that is not part of a closed loop and that either (a) directly communicates with the reactor coolant during normal operation or in accident conditions or (b) directly communicates with the containment atmosphere during normal operation or in accident conditions should be provided with two isolation valves in series. Each valve either should be normally closed or should have provisions to close automatically. Where the line communicates directly with the reactor coolant or the containment atmosphere, one valve should be provided inside the containment and one valve outside. If two valves either inside or outside the containment can provide an equivalent barrier (i.e., can meet all the design requirements) in certain applications, this may also be an acceptable arrangement. Each valve should be reliably and independently actuated. Isolation valves should be located as close as practicable to the structural boundary of the containment.

Loops that are closed either inside or outside the containment should have at least one isolation valve outside the containment at each penetration. This valve should be an automatic valve, a normally-closed valve or a remotely-operated valve. Where the failure of a closed loop is assumed as an initiating event or as a consequence of an initiating event, the recommendations in the previous paragraph will apply to each line of the closed loop.

Loops that are closed both inside and outside the containment envelope should have at least one isolation valve, an automatic valve, a normally-closed valve or a remotely-operated valve outside the containment envelope at each penetration.

Table 1 Categories of isolation features

See para.	Schematic configuration	Example
1.(a)		Cover gas system
1.(b)		Ventilation system
2.		Cold trap Cooling system (opening type)
3.		Air conditioning cooling system

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II. ANNEX

II.1. Active Reactor Shutdown System

(1) System configuration

An SFR typically has two active reactor shutdown systems of control rods. Each system comprises control rod insertion mechanisms and control rods, which works with protection system, i.e., detectors, logic circuits that process detected signals, actuation circuits that activate control rod insertion. The reactor shutdown system, which also has reactor control function, is designed to allow a change in the control rods position for the reactor start-up and shutdown, and power control. The control rod assemblies are reactor core elements and independent from fuel assemblies. The control rods, which consist of pin bundle of neutron absorbers are located at the upper part (fully withdrawn) or in the middle of the height of the core under normal operation inside control rod guide tubes. Insertion of control rods is conducted so that vertical position of control rods can be rapidly changed from the initial position to fully inserted position inside the control rod guide tubes.

(2) Measures for enhancing reliability

Two shutdown systems are designed to have independence and diversity to the extent practicable to prevent common cause failures. Examples of diversities are as follows.

- Actuation: mechanical latch, electromagnet
- Driving force: fast drive-in motor, gas acceleration, gravity drop
- Insertion: control rod with / without driving shaft

Detection parameters for the reactor shutdown systems should be diverse to the extent practicable. Typical detection parameters are neutron flux, reactor or core outlet coolant temperature, primary pump rotation, primary flow rate, off site electric power voltage.

The reactor shutdown systems is designed to ensure reactor shutdown considering failure of an element of entire system including detectors, circuits for signal processing and activation, insertion mechanisms, and control rods, and influence of any abnormal events which may happen in the reactor power plant. The detectors, circuits for signal processing and activation are redundant by e.g., application of two out of four voting logic, considering single failure and maintenance.

Reactor shutdown systems are a fail-safe design. The control rods are inserted when electric power supply of holding control rod is lost.

Withdrawal of control rod of another reactor shutdown system is prevented when one reactor shutdown system is actuated.

II.2. Passive Reactivity Mechanisms

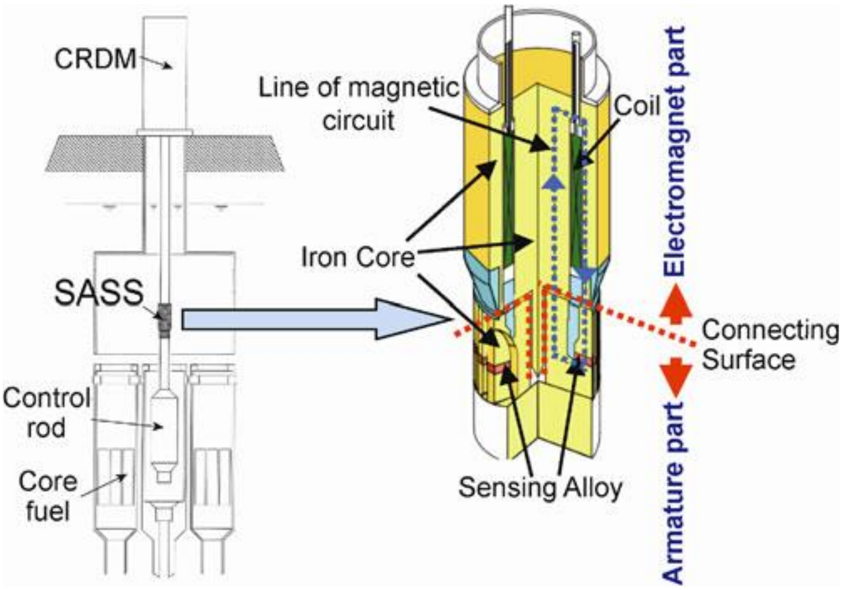
Various passive reactivity reduction mechanisms and passive reactivity feedback mechanisms have been developed [II-1]. Depending on the reactor core design, these mechanisms are designed to ensure core damage prevention in the case of failure of active reactor shutdown systems under following conditions: abnormality in power, abnormal coolant flow rate and temperature, and other abnormal events including combinations of them. It can be achieved by utilizing inherent reactivity feedback characteristics and appropriate combination of these mechanisms.

Passive mechanisms that are incorporated into the design are such that 1) common cause failures with the active reactor shutdown systems are prevented, 2) proven technology based on the results of research and development programs are applied, 3) provisions for monitoring and testing are provided, and 4) it is possible to resume normal operation in case of their erroneous activation.

Examples of passive reactivity reduction mechanisms are

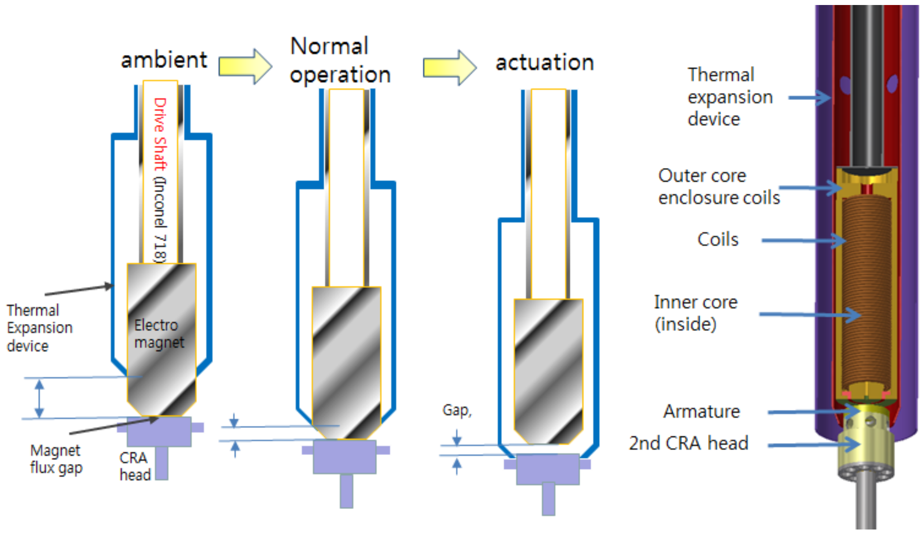
- Passive control rod insertion by gravity achieved by their release due to **magnetic property change of temperature sensing alloy** when the reactor coolant temperature reaches the Curie-Point as shown in Figure II-1. [II-2][II-3]
- Passive control rod insertion by gravity achieved by **thermal expansion-based release** of control rods as shown in Figure II-2. [II-4][II-5]
- **Hydraulically levitated absorbers** that lower a neutron absorber into the core region when primary sodium flow is reduced due to pump trip. [II-2][II-6]
- Hydraulically levitated boron balls. [II-7]

An example of passive reactivity feedback mechanism is the Gas Expansion Module (GEM) that increases neutron leakage from the core when primary sodium flow is reduced due to pump trip. [II-8]



Control rods are passively inserted utilizing magnetic property change of temperature sensing alloy at Curie point in the case of reactor coolant temperature increase against ATWS events. This passive mechanism is introduced above the core.

Figure II-1 Curie point magnetic alloy type



Control rods are passively inserted utilizing an enhanced thermal expansion device in the case of reactor coolant temperature increase. This passive reactivity reduction mechanism is introduced above the core at the end of the driveline of the secondary control rod system.

Figure II-2 Thermal expansion device type

II.3. Configurations and Measures against Sodium Leakage and Combustion

There are examples below for design measures against sodium leakage and combustion inside and outside of containment structure.

(1) Inside of Containment Structure

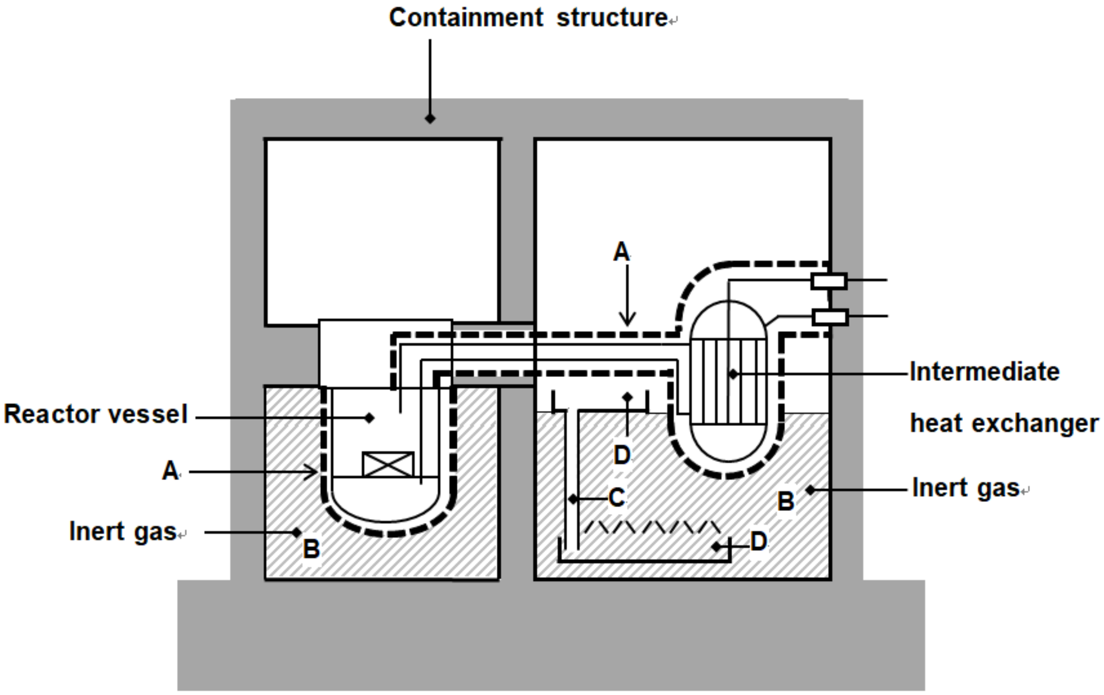
As shown in Figure II-3, examples of measures against sodium leakage and combustion related to inside of containment structure are as follows.

A) Installing protective structures filled with inert gas such as a guard vessel enclosing major components and guard pipes for pipes in the primary coolant system,

B) Compartments filled with inert gas,

C) Transporting leaked sodium to compartments filled with inert gas using transportation pipes, and

D) Installing catch pans and/or combustion restraint plates to protect concrete structure.



A) Guard vessel, Guard pipe, B) Compartments filled with inert gas

C) Transportation pipe, D) Catch pan, Combustion restraint plate

Figure II-3 Example of measures inside of containment structure

(2) Outside of Containment Structure

As shown in Figure II-4, examples of measures against sodium leakage and combustion related to outside of containment structure are as follows.

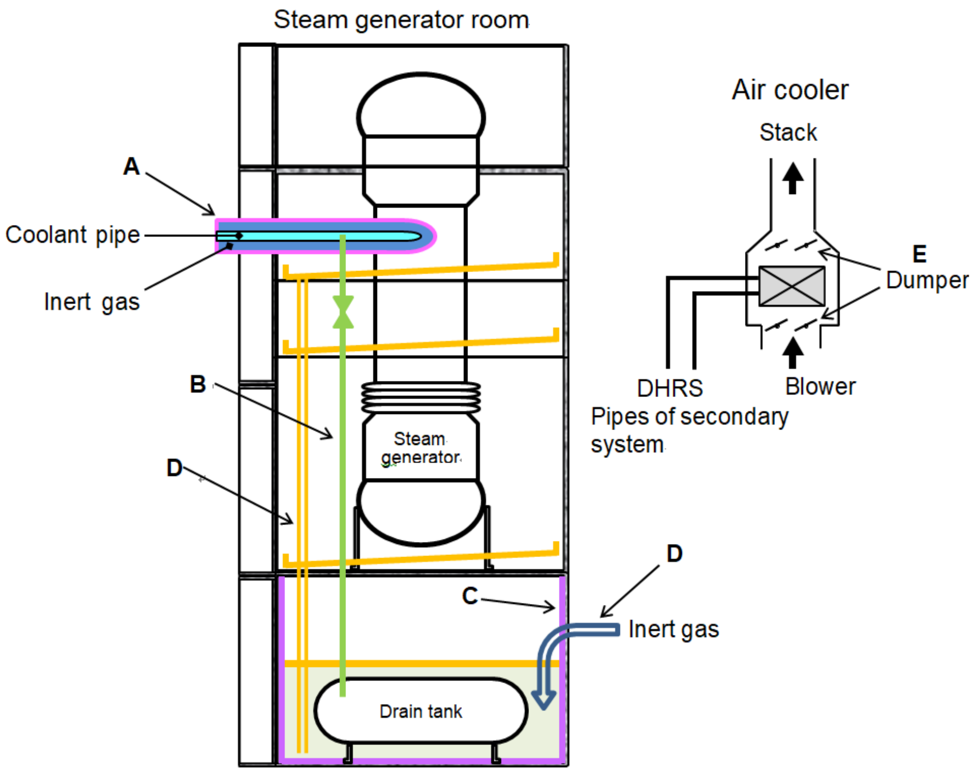
A) Installing protective structures filled with inert gas such as an enclosure in the secondary coolant system,

B) Draining sodium in components in emergencies to restrain sodium leakage,

C) Installing catch pans and/or combustion restraint plates to protect concrete structure

D) Transporting leaked sodium to compartments using transportation pipes and injecting inert gas into the compartments, and

E) Closing dumper in emergencies to restrain sodium combustion.



A) Enclosure, B) Draining sodium in components,

C) Catch pan, Combustion restraint plate,

D) Transportation pipe and inert gas injection, E) Dumper

Figure II-4 Example of measures outside of containment structure

II.4. Design Measures Against Sodium-Water Reaction

There are two examples as shown in Figures II-5-1 and II-5-2 for design measures against sodium-water reaction of water leak detecting system, water and steam shutoff system, pressure relief system, sodium-water reaction product treatment system, and nitrogen injection system. Aside from these conventional design measures, double wall heat transfer tube has a potential to reduce probability of water leak and consequences.

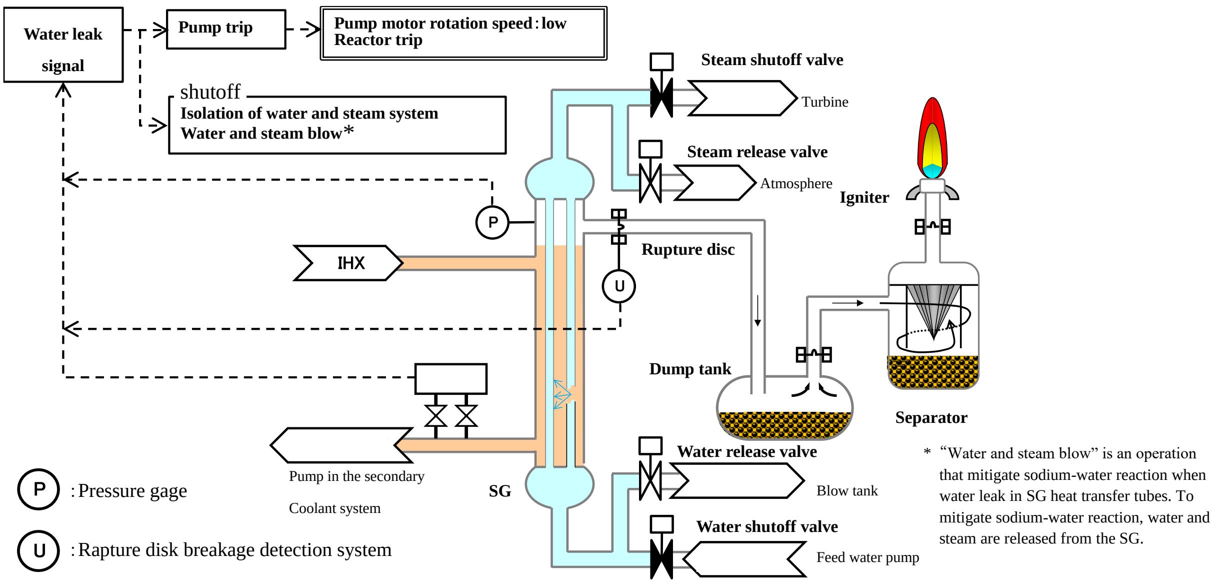


Figure II-5-1 Design measures against sodium-water reaction (1)

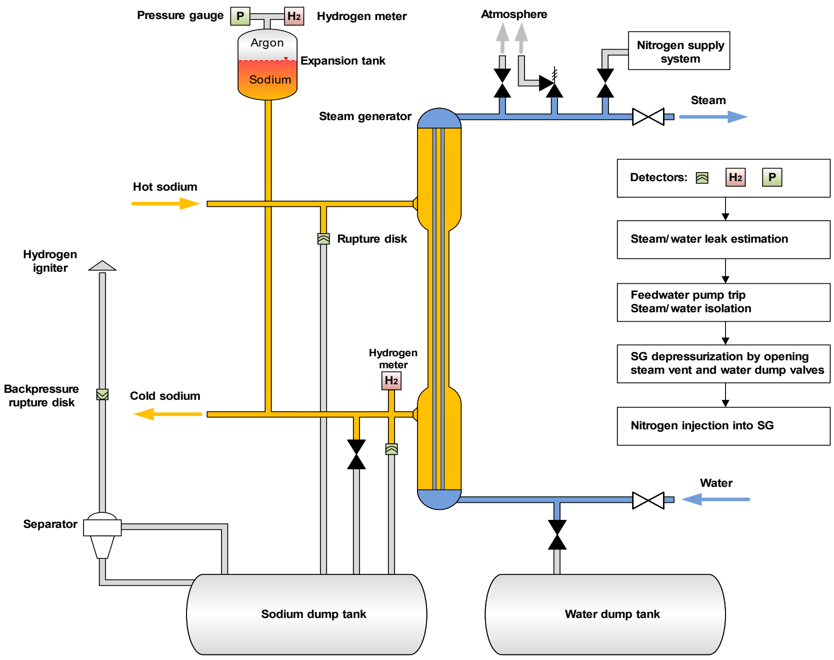


Figure II-5-2 Design measures against sodium-water reaction (2)

II.5. Containment and its Associated Systems

This annex provides some examples of design concepts of the containment and its associated systems for SFRs, focusing on the relation between associated systems for achieving all the functions of the containment properly. Although the design concepts are described briefly, readers will be able to grasp the whole picture of each containment and its associated systems.

(1) Steel containment structure (e.g., Monju)

A typical containment and its associated systems (Figure II-6-1) is composed of the following.

- A steel containment vessel, typically cylindrical and semi-spherical structure, which serves as a containment envelope and encloses all the primary system components.
- Concrete walls surrounding the containment steel vessel, forming secondary confinement building
- A filtering system

The secondary confinement building provides the following functions.

- Protection of inner systems and components against postulated external events
- Retention of leaked materials from the containment envelope

The temperature and pressure inside the containment vessel are maintained below the design limits by suppressing sodium combustion by inert gas atmosphere of the cell with steel liner. To control leaked gas from the cell or sodium combustion induced by directly released sodium into containment atmosphere, the peak pressure and peak temperature in the accident conditions will be limited by the free volume and heat capacity of the containment (the containment steel vessel and internals). A vacuum breaker can be placed as needed to cope with negative pressure in containment atmosphere due to decrease in oxygen pressure that can occur after the termination of sodium combustion. A catch-pan and sodium transfer tube can also be provided in the cell considering possible level of sodium leakage to mitigate heat effects on concrete and reduce the risk of sodium-concrete contact.

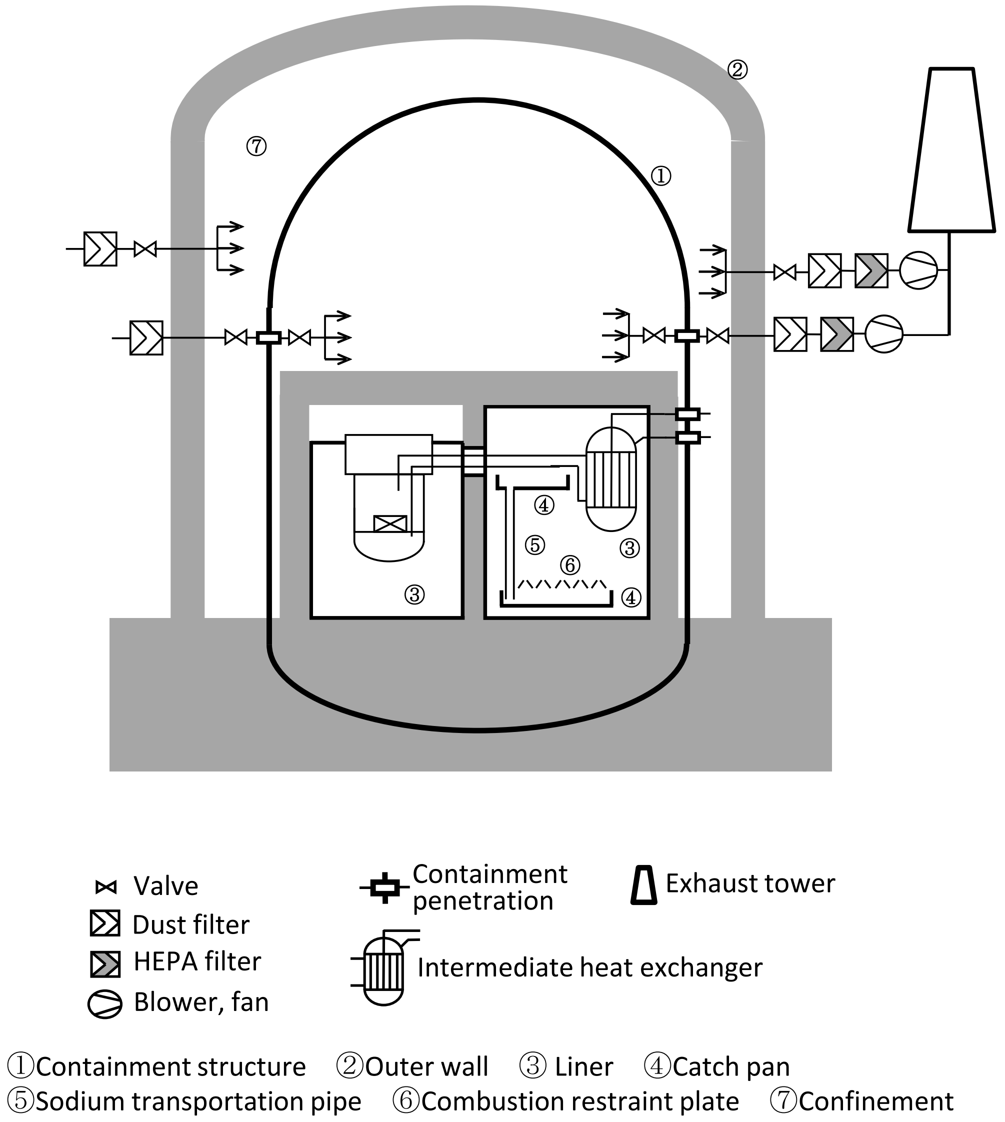


Figure II-6-1 Steel containment structure

(2) Concrete building with protective cover containment structure (e.g., PFBR, BN800)

Some cases of this design concept utilize the radiation shielding and upper structure of the concrete building as the containment. In pool-type design, specifically, a cylindrical or semi-spherical protective cover is installed in the upper part of reactor vessel deck to form a containment together with the radiation shielding as shown in Figure II-6-2. Steel liner is installed onto the inner wall of the radiation shielding as necessary. Cooling measures on the concrete surface can be provided to reduce the risks of heat effects from the reactor vessel and temperature rise in concrete due to high radiation in normal operation.

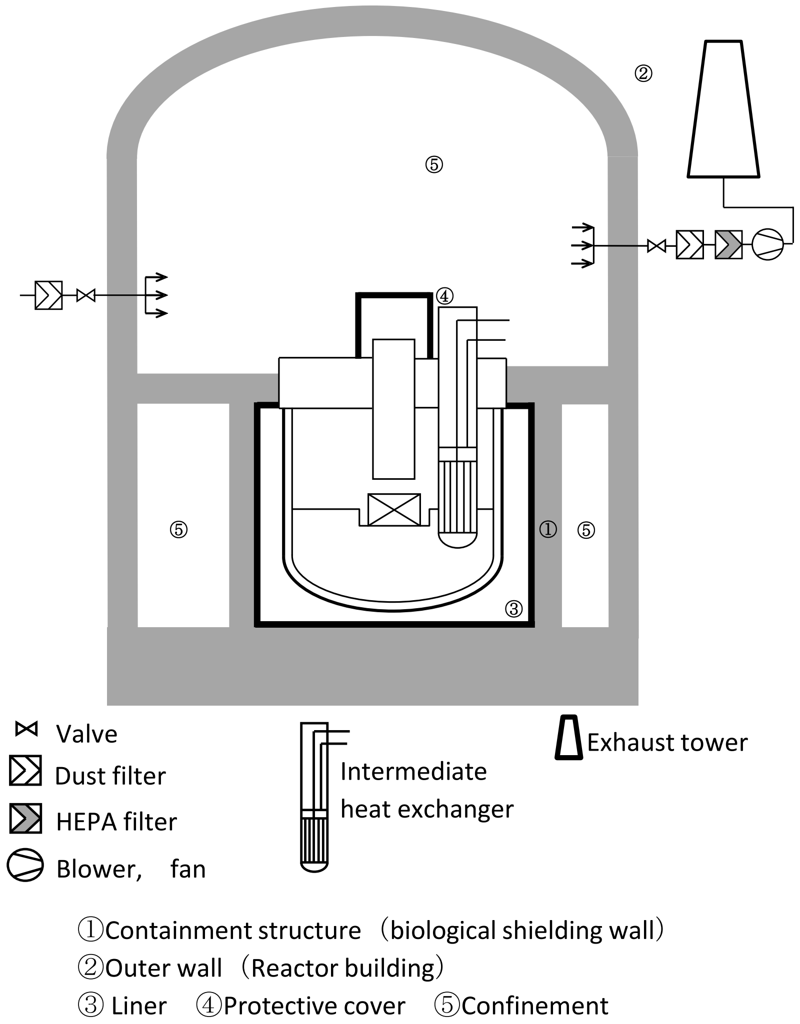


Figure II-6-2 Concrete building with protective cover containment structure

(3) Guard vessel with top dome containment structure (e.g., Superphénix)

In this concept, the guard vessel in the reactor building also serves as containment as shown in Figure II-6-3. The facility's upper part forms a containment by the upper part of the building or a dome placed on the upper part of the reactor building.

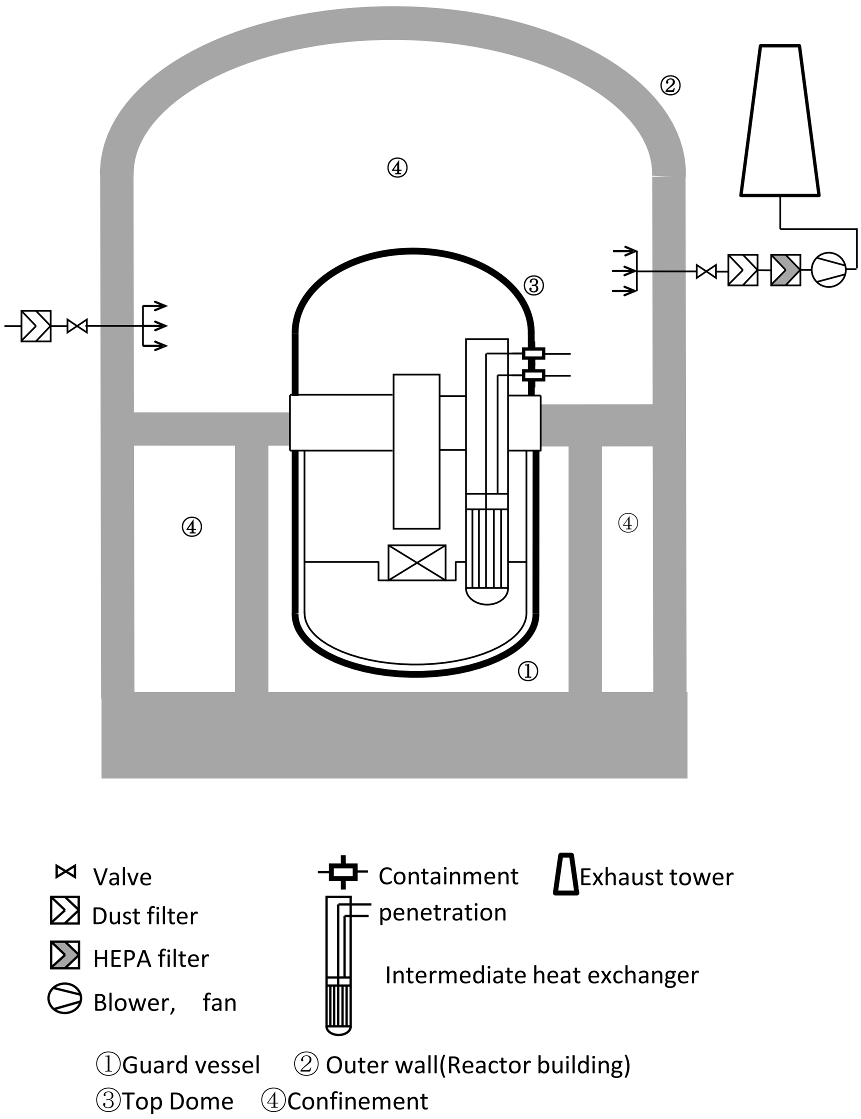


Figure II-6-3 Guard vessel with top dome containment structure

(4) Concrete building containment structure (e.g., JSFR)

This concept utilizes reinforced cells of the reactor building as containment. Measures to control the temperature and pressure against sodium composition are therefore the same as shown in Figure II-6-1. Figure II-6-4 shows a design example to reduce the loads on the containment and risks of sodium-concrete contact in the case of sodium leakage by adding pressure-resistant guard vessel, guard pipes, and enclosures to components containing sodium.

The design concept of the secondary confinement building and its function are also the same as the example of the steel containment structure.

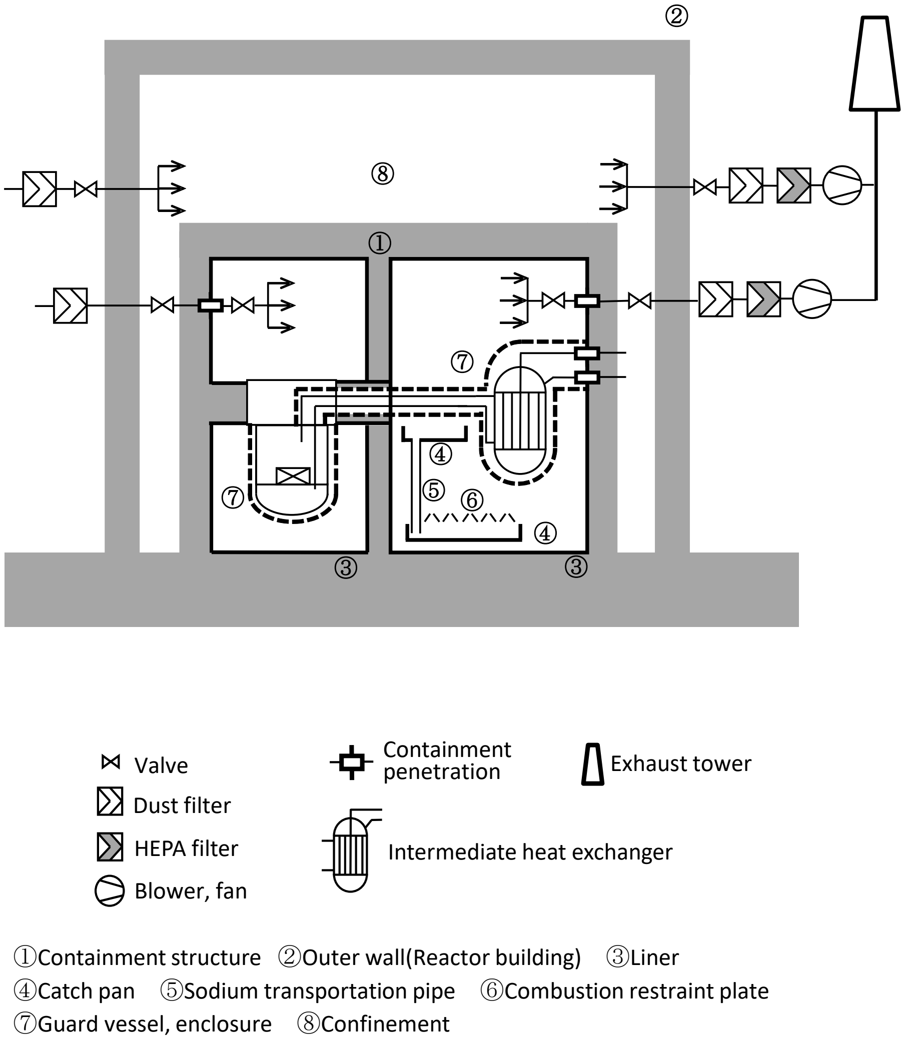


Figure II-6-4 Concrete building containment structure

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[II-3] S. NAKANISHI et al., Development of advanced loop-type fast reactor in Japan – (5): adoption of self actuated shutdown system to JSFR," Proc. of ICAPP 2008, Anaheim, USA, (2008).

[II-4] D. FAVET et al., "Third shutdown level for EFR project," Proc. of a Technical Committee meeting on Absorber Materials, Control Rods and Designs of Shutdown Systems for Advanced Liquid Metal Fast Reactors, p. 173, IAEA-TECDOC-884, Obninsk, Russian Federation (1995).

[II-5] Design Study for Passive Shutdown System of the PGSFR, Technical Meeting on Passive Shutdown Systems for Liquid Metal-Cooled Fast Reactors (2015)

[II-6] Yu. E. BAGDASAROV et al., "Development of passive safety devices for sodium-cooled fast reactor," Proc. of a Technical Committee meeting on Absorber Materials, Control Rods and Designs of Shutdown Systems for Advanced Liquid Metal Fast Reactors, p.97, IAEA-TECDOC-884, Obninsk, Russian Federation (1995).

[II-7] E. Specht et al., "Hydraulically Supported Absorber Balls Shutdown System for Inherently Safe LMFBRs," Proc Intl. Mtng on fast reactor Safety and Related Physics, Chicago, Illinois, October 5-8, 1976.

[II-8] "Pre-application Safety Evaluation Report for the Power Reactor Innovative Small Modular (PRISM) Liquid Metal Reactor," US. Nuclear Regulatory Commission report NUREG-1368.

III. GLOSSARY

- **accident conditions**

  Deviations from normal operation that are less frequent and more severe than anticipated operational occurrences.

  [IAEA Safety Glossary (2018)]

- **anticipated operational occurrence**

  A deviation of an operational process from normal operation that is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

  [IAEA Safety Glossary (2018)]

- **controlled state**

  Plant state, following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to effect provisions to reach a safe state.

  [IAEA Safety Glossary (2018)]

- **core assembly**

  core assembly is the general term of fuel assembly, control rod assembly, shielding assembly and blanket fuel assembly.

- **core restraint system**

  Specific structure which is provided for limiting horizontal allowable space from the core peripheral zone, e.g., core formers connected with the core barrel.

- **design basis accident**

  A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.

  [IAEA Safety Glossary (2018)]

- **design extension conditions**

  Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.

[IAEA Safety Glossary (2018)]

- **fuel assembly**

  Fuel assembly contains numbers of fuel elements (called a fuel bundle) and neutron shields at upper and lower parts of the fuel bundle in the hexagonal wrapper tube. They are loaded into and subsequently removed from a reactor core as a single unit.

- **fuel element**

  A rod of nuclear fuel which consists of solid fuel pellets or slugs, its cladding and any associated components necessary to form a structural entity.

- **guard pipe**
- **guard vessel**

  Guard pipe is placed outside of the coolant pipe where sodium coolant flows.

  Guard vessel is placed outside the reactor vessel containing the sodium coolant.

  Both are installed to maintain sodium coolant level for reactor cooling in case of sodium leakage.

- **item important to safety**

  An item that is part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public.

  [IAEA Safety Glossary (2018)]

- **leak tight configuration**

  Configuration that ensures leak-tightness and gas-tightness of the reactor coolant boundary and cover gas boundary.

  [Based on IAEA NS-G 1.10 (2004)]

- **maximum leak rate**

  Containment leak rate for the structure design. This leak rate should be lower than "safety limit leak rate".

  [Based on IAEA NS-G 1.10 (2004)]

- **misloading**

  Improper loading of a fuel assembly into a reactor core. Misloading will cause abnormal levels of effective multiplication factor, neutron flux and power distribution, coolant velocity, and temperature distribution.

- **normal operation**

  Operation within specified operational limits and conditions.

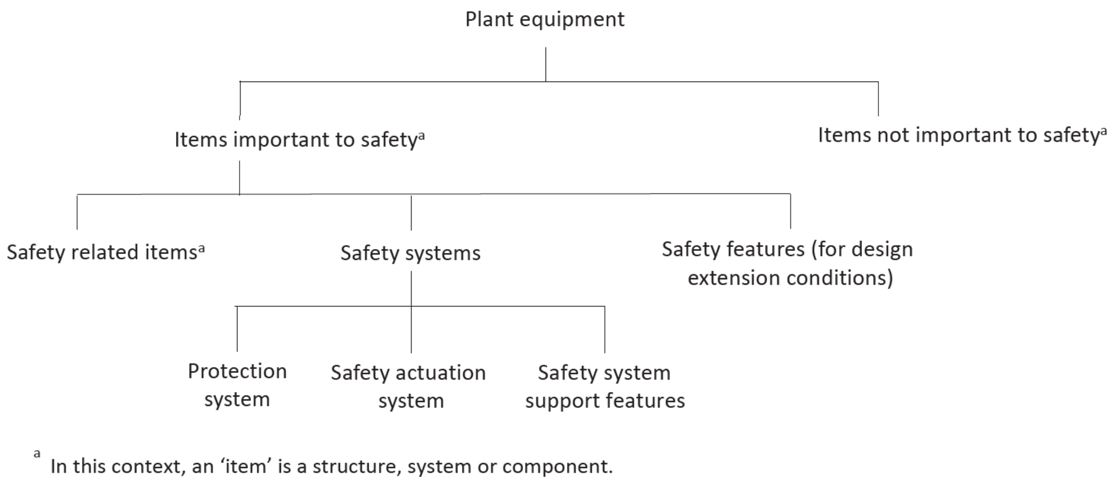
  [IAEA Safety Glossary (2018)]

- **operational states**

  States defined under normal operation and anticipated operational occurrences.

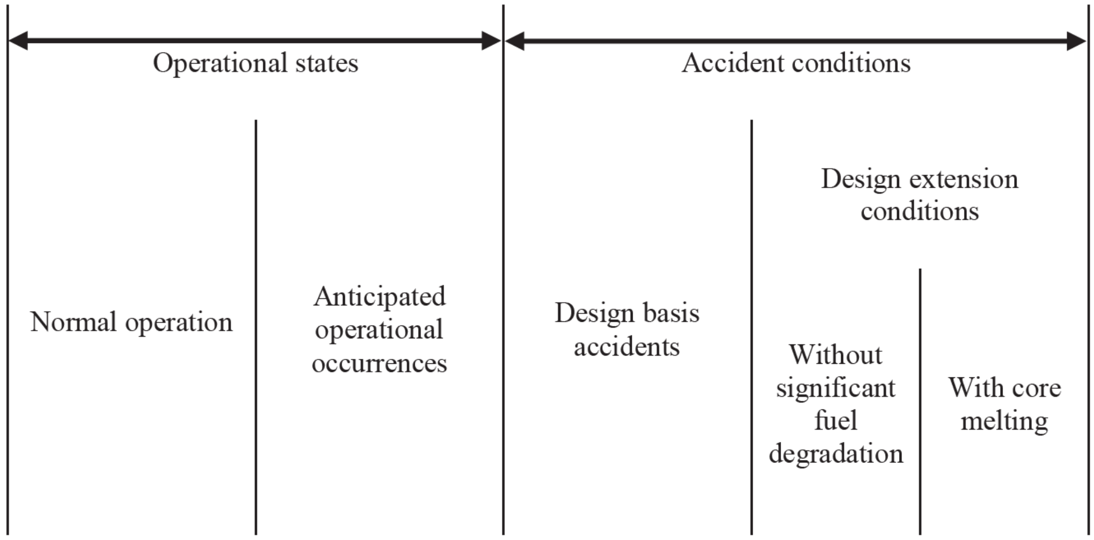
  [IAEA Safety Glossary (2018)]

- **plant equipment**



[Based on IAEA Safety Glossary (2018), "Safety related items" is replaced to "Safety relevant items".]

- **plant states (considered in design)**



[IAEA Safety Glossary (2018)]

- **primary coolant system**

  The coolant system used to remove heat from the reactor core and to transfer the heat to the coolant in the secondary coolant system.

- **protection system**

  System that monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

  [IAEA Safety Glossary (2018)]

- **reactor coolant boundary**

  The reactor coolant boundary is defined as the barrier of components which contains the primary coolant. The breakage of this boundary induces a primary coolant leak. The reactor coolant boundary forms a barrier against radioactive materials release together with the reactor cover gas boundary.

- **reactor coolant system and associated systems**

  All systems used to remove heat from the reactor core and transfer that heat to the ultimate heat sink. The reactor coolant system and associated systems include the primary coolant system, the secondary coolant system, the decay heat removal system, the cleanup facilities, and the power conversion system with associated coolant system.

- **reactor cover gas boundary**

  The reactor cover gas boundary is defined as the barrier of components which contains the reactor cover gas. The breakage of this boundary induces a reactor cover gas leak. The reactor cover gas boundary forms a barrier against radioactive materials release together with the reactor coolant boundary.

- **safe state**

  Plant state, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and maintained stable for a long time.

  [IAEA Safety Glossary (2018)]

- **safety limit leak rate**

  Containment leak rate for safety assessment. Evaluate radiation exposure with this leak rate.

  [Based on IAEA NS-G 1.10 (2004)]

- **safety actuation system**

  The collection of equipment required to accomplish the necessary safety actions when initiated by the protection system.

  [IAEA Safety Glossary (2018)]

- **safety feature (for design extension conditions)**

  Item that is designed to perform a safety function or that has a safety function for design extension conditions.

  [IAEA Safety Glossary (2018)]

- **safety group**

  The assembly of equipment designated to perform all actions required for a particular initiating event to ensure that the limits specified in the design basis for anticipated operational occurrences and design basis accidents are not exceeded.

  [IAEA Safety Glossary (2018)]

- **safety relevant item**

  An item important to safety that is not part of safety systems.

  [Based on “safety related item” from IAEA Safety Glossary (2018)]

- **safety relevant system**

  A system important to safety that is not part of safety systems.

  [Based on "safety related system" from IAEA Safety Glossary (2018)]

- **safety system**

  A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the reactor core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

  [IAEA Safety Glossary (2018)]

- **safety system settings**

  Settings for levels at which safety systems are automatically actuated in the event of anticipated operational occurrences or design basis accidents, to prevent safety limits from being exceeded.

  [IAEA Safety Glossary (2018)]

- **safety system support features**

  The collection of equipment that provides services such as cooling, lubrication and energy supply required by the protection system and the safety actuation systems.

  [IAEA Safety Glossary (2018)]

- **secondary coolant system (or intermediate coolant system)**

  The coolant system used to transfer heat from the coolant in the primary coolant system to the working fluid in the turbine system such as a water/steam system via a heat exchanger.

- **sodium-concrete reaction**

  A chemical reaction due to the direct contact between sodium and concrete, which generates hydrogen gas that may cause overpressure in a containment.

- **sodium fire**

  A fire caused by sodium combustion. Sodium spontaneously catches fire when exposed to air at the operating temperature of an SFR.

- **sodium water reaction**

  A chemical reaction caused by the direct contact between sodium and water/steam.

- **steam generator**

  A heat exchanger to transfer heat from a sodium system to a water/steam system.

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