Advanced sodium-cooled fast reactor development regarding GIF safety design criteria

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Abstract. Design studies on a next generation sodium-cooled fast reactor (SFR) considering the safety design criteria (SDC) developed in the generation IV international forum (GIF) was summarized. To meet SDC including the lessons learned from TEPCO's Fukushima Dai-ichi nuclear power plants (1F) accident, the heat removal function was enhanced to avoid loss of the heat removal function even if any internal events exceeding design basis or severe external events happen. Several design options for alternative decay heat removal systems have been investigated. An auxiliary core cooling system using air as ultimate heat sink has been selected as an additional cooling system regarding system reliability and diversification. Even though the advanced SFR already adopts seismic isolation system, main component designs have been improved considering revised earthquake conditions. For other external events, design measures for various external events are taken into account. Reactor building design has been improved and important safety components are diversified and located separately to improve independency. Design studies and evaluations on the advanced sodium-cooled fast reactor have contributed to the development of safety design guidelines (SDG) which is under discussion in the GIF framework.

Key Words: decay heat removal, sodium-cooled reactor, loss of heat removal system

1. Introduction

In the framework of generation IV international forum (GIF), Safety Design Criteria (SDC) and Safety Design Guidelines (SDG) for the generation IV SFRs have been developing in the circumstance of worldwide deployment of SFRs [1]. The SDC and SDG, which incorporate the lessons learned from TEPCO's Fukushima Dai-ichi nuclear power plants (1F) accident, require modifications to enhance design measures against severe accidents and to provide design measures against severe external events by utilizing inherent or passive safety features based on general SFR characteristics. This paper describes safety design improvements on a sodium-cooled fast reactor (SFR) which has been proposed as an advanced concept meeting GIF design goals. To meet the GIF design goals, the advanced SFR includes several key technologies such as high-burnup core, safety enhancement, compact reactor vessel, two-loop cooling system using high-chromium steel, integrated intermediate heat exchanger (IHX)/pump component, reliable steam generator (SG), natural-circulation decay heat removal systems (DHRS), simplified fuel handling system (FHS), containment vessel (CV) made of concrete that is reinforced with steel plates, and advanced seismic isolation system [2]. However, more safety improvements are required in order to meet the SDC and SDG. Those safety improvements could contribute to the SDC and SDG by showing design solutions for further safety enhancement.

2. Core and Safety Design

2.1. Core

The core performance requirements and the design conditions for the advanced SFR for the demonstration and the commercial phase were reviewed and modified as shown in Table I. Using a remarkable correlation between fuel compositions and core characteristics, the design envelopes that include any fuel composition appearing in an FR fuel cycle were established and their consistency was checked by comparing with some representative FR-deployment scenario simulations [3]. According to the revised requirements and conditions, the core design has been modified [4]. By optimizing core arrangements and employing a flux adjuster as shown in Fig.1, the core design attains horizontally flat power distributions and small power deformations for motions of the primary control rods moving achieving a high burnup with a total average discharged burnup of 83 GWd/t.

Table I Core performance requirements and Design conditions [3]

Item	Commercial Reactor	Demonstration Reactor
<General>		
Power output	1500 MWe	750 MWe
Coolant temp.	550 / 395 °C	Same
Flow rate	18000 kg/s	9000 kg/s
<Performance>		
Total burnup (breeding / break-even)	>60 / >80 GWd/t	>60 GWd/t
Core burnup	150 GWd/t	—
Operation cycle length	Longer than demo	>13 months
Breeding ratio		
(low-breeding)	1.1-1.0	1.1
(high-breeding)	1.2	—
MA content	< 3 wt.%	Same
<Safety>		
void reactivity	< 6 $	Same
Core height	< 100 cm	Same
Specific heat	> 40 kW/kg	Same
Fuel assembly	FAIDUS	Same

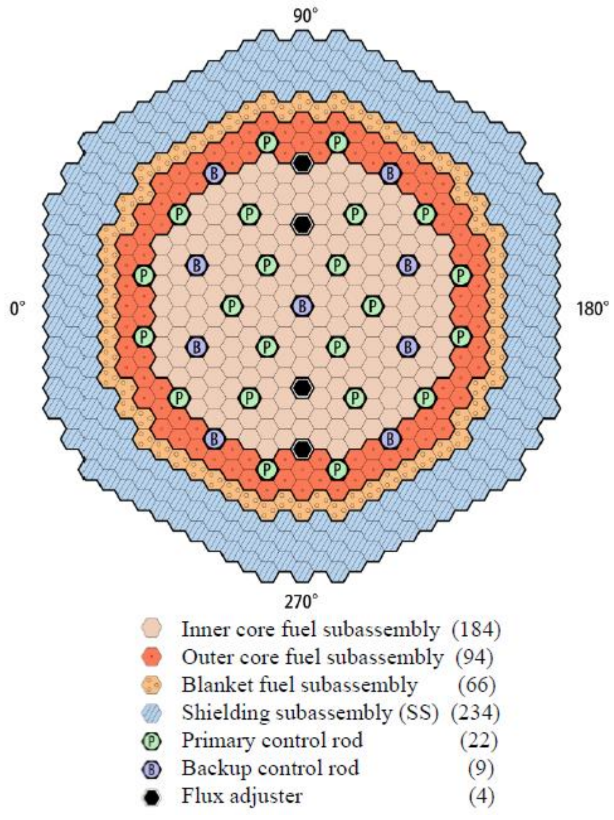


FIG. 1 Core configurations of the demonstration reactor [4]

For radial shielding, the original design adopted zirconium hydride to reduce core diameter [2]. Regarding lesson’s learned from the 1F accident, the shielding material is changed from zirconium hydride to stainless steel to exclude hydrogen production in severe cases. Due to this change, the column of the radial shielding increases from 2 to 3 enlarging the core diameter. The efficiency of the stainless steel shielding has been confirmed by analyses [4]. For the reactor vessel, there is no impact on the reactor vessel structure, since the reactor vessel diameter is already enlarged by maintenance or fabrication points of view.

2.2. Safety

For anticipate transient without scram (ATWS), design measures installed in the original design were already effective and sufficient to meet the SDC. For prevention, the original design adopted self-actuated shutdown system (SASS) [2]. In the recent study related with SDC and SDG, the SASS design was modified to improved response time and the improved response time was confirmed by a 3 dimensional thermal hydraulics analysis. Using the improved response time, the performance of SASS for full and partial load operations has been confirmed [8].

For CDA mitigation, the approach is prevention of severe power burst events with recriticality and stable cooling of core materials in the reactor vessel achieving in-vessel retention. In developing the CDA scenario, the core degradation sequences were conveniently divided into four phases: the initiating, early discharge, material relocation, and heat removal phases. For the initiating phase, the core accepts design limits on coolant void reactivity, core specific power, and core height as described in safety requirements of Table I. For the early discharge, the advanced SFR core adopts the modified fuel assembly with inner duct structure (FAIDUS) [2]. Recently, a methodology for SAS4A core modeling was improved to obtain data of power profile and reactivity profile consistent with core design, which are very important to analyze fuel failure position and reactivity insertion in the initiating phase. Various types of ULOF and UTOP transients from the full power operation state and the low power operation state with cores of EOEC and BOEC were analyzed. The results showed that the present core design could prevent severe power burst even considering uncertainty of sodium void worth as shown in Fig. 2 [9].

For the material relocation phase, molten fuel discharge from the control rod guide tubes (CRGT) is under investigation involving EAGLE-3 in-pile experiments [10]. For the heat removal phase, there is a core catcher at the bottom of the reactor vessel and decay heat could be removed by full natural convection decay heat removal system (DHRS).

2.3. Seismic conditions

After the 1F accident, the design seismic condition of SFR is reviewed. As the results of reviewing, the design countermeasures for external hazards are carried out, then, the acceleration level of the floor response spectra to be used for design seismic conditions has been increased. The reasons are design seismic condition is larger and the weight of reactor building is increased according to the improved external hazards. Especially, the vertical response acceleration at the primary pump motor floor is exceeded to design criteria with the original design. The revised seismic conditions are shown in Fig. 3 [11] as shown 2012 soft and hard rocks. Considering majority of domestic LWR conditions for both soft and hard rocks, the design seismic condition for safety shutdown (SS) has been selected to 0.8 of the 2012 condition. As for the severe earthquake beyond SS condition, 1.8SS has been selected.

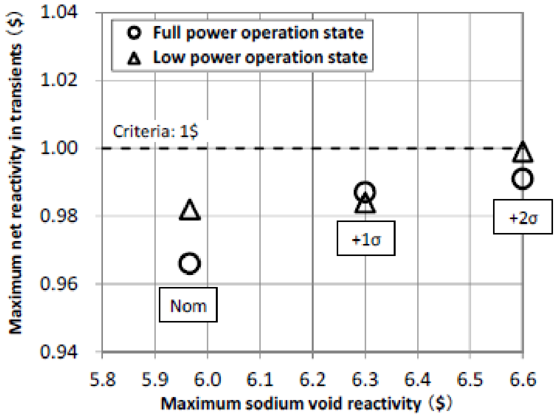


FIG. 2 *maximum net reactivity in ULOF [9]*

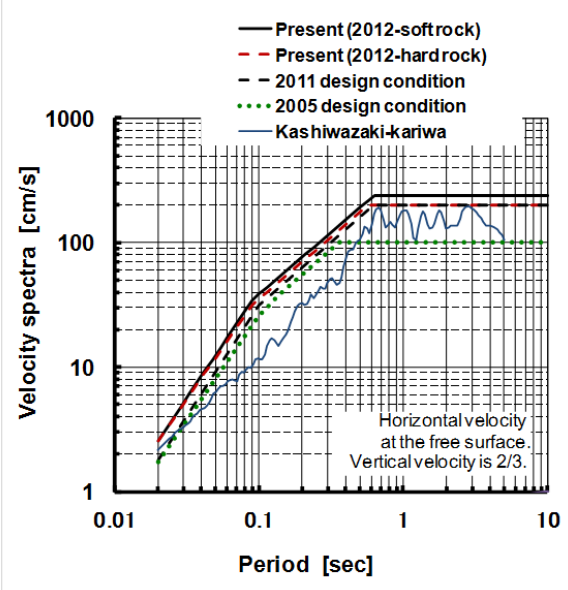


FIG. 3 Design Seismic Condition in 2012 [11]

3. Plant System Design

3.1. Reactor vessel

During evaluation on the key technologies [2], the reactor vessel is installed a sodium annulus to protect the main vessel from moving sodium surface during startup and shutdown operation as shown in Fig. 5. With the sodium annulus with stagnant sodium level, the thermal stress on the main vessel could be reduced. In the recent study, the detail of the sodium annulus has been designed and reactor vessel structures have been modified. From maintenance point of view, in-vessel structures have been improved providing access for visual or volumetric tests at least one side of coolant or cover gas boundary. For decay heat removal systems, auxiliary core cooling system (ACS) has been newly installed as an alternative DHRS. The detail of the ACS will be described in section 3.4. Accommodating those design improvements, the inner diameter of the reactor vessel is enlarged from 9.46m [12] to 11.83m.

For seismic design, buckling has been evaluated based on the revised SS earthquake condition discussed in section 2.3. The floor response spectrum at the reactor vessel support is shown in Fig. 4. Based on seismic analysis, the reactor vessel could maintain its integrity with thickness of 45mm even in the severe earthquake including 1.8SS condition. For the guard vessel, integrity with sodium in the annulus between the reactor and guard vessels has been evaluated. Based on seismic analysis, the guard vessel with 25mm thickness could stand the 1.8SS seismic condition.

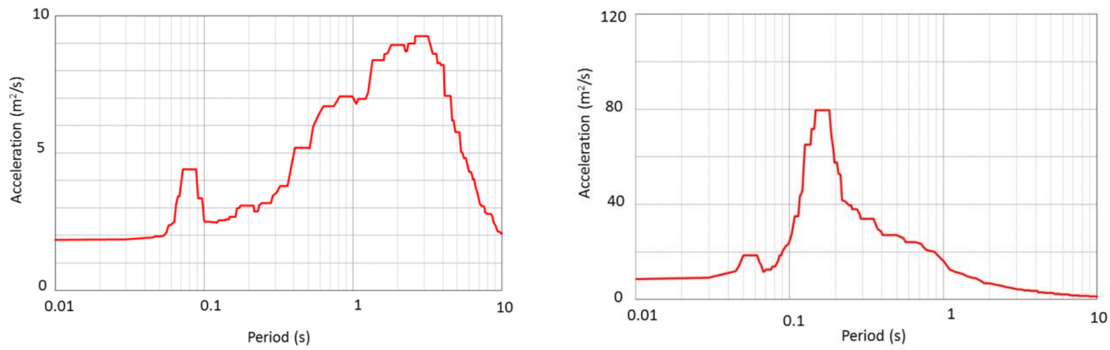
For CDA, molten fuel discharge from CRGTs and relocation to the core catcher are taken into account in the design. The core support structure has protections against molten fuel jets and distance between the core support and the core catcher is large enough to form debris. Since structural integrity of the core catcher was concerned due to its exposure to sodium flow with high flow rates, the outlet nozzle of CL-piping was modified for preventing such the sodium flow from facing the core catcher.

3.2. Primary cooling system

For the integrated pump/IHX component, maintenance and repair capability of the IHX has also been improved. In the original design, the heat exchange tubes for the primary reactor auxiliary cooling system (PRACS) are built-in in the upper plenum of the IHX. In the improved design, the built-in type was modified to plug-in type modules. With the plug-in type heat exchanger, checking and repair are available after removing the PRACS heat exchangers, and access to the IHX upper plenum was also improved [6].

To satisfy safety requirements, the integrity to withstand the severe earthquake, the reliability of the guard vessel in the primary coolant leak, and the reliability of expansion joints in the sodium-water reaction have been evaluated. In addition, important evaluations have been implemented on thermal transient, structural sympathetic vibration with pump rotation and wear-out of IHX tubes to validate design consistency [13]. To meet those requirements regarding fabrication capability also, the integrated pump/IHX has been modified as shown in Fig. 6.

For primary piping inspection, survey study on inspection devices and improvement of inspection access were investigated. For the inspection device access, the thermal insulation is moved from inside to outside of the guard piping. Partition structure and bellows on the guard pipe design were modified taking into account the revised thermal insulation and structural integrity [5]. With the revised design, structural integrity has been confirmed. From the view point of fatigue due to flow-induce vibration, Ithe pipe stresses considering design factors such as the stress concentration factor were less than the design fatigue limit. Therefore, this evaluation confirmed the integrity of the primary hot-leg piping in the demonstration reactor [14].



(horizontal) (vertical)

FIG. 4 Floor response spectrum at the reactor vessel support

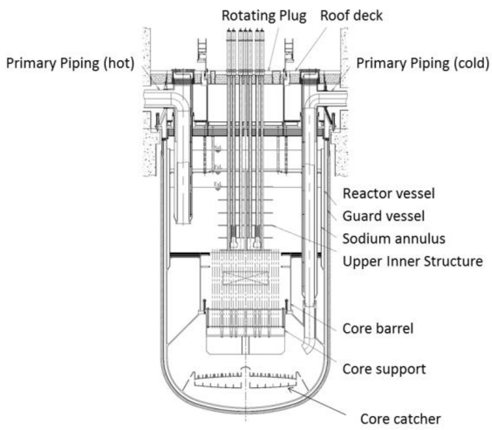


FIG. 5 Reactor vessel

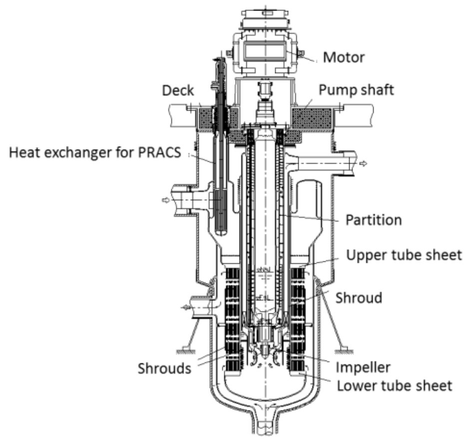


FIG. 6 Integrated Pump/IHX

3.3. Secondary cooling system

For steam generator (SG), the original design adopts the double-wall tube reliable SG for both safety and property protection. Periodical inspections on both inner and outer tubes are required to keep reliable sodium-water boundaries [2]. Taking into account limited R&D resources and reduction of R&D risks, alternative SG concepts: coated wall tube, protective wall tube, and nickel based alloy tube SGs were investigated as a development step for the double-wall tube SG [15]. Since 2011, a protective wall tube SG which equips the outer tubes with mechanically contacted inner tubes (Fig. 7) has been investigated. While the selected protective wall tube is basically similar to the double-wall tube, the thickness of the inner tube is thicker than that of the original double-wall SG to be able to withstand target wastage in a sodium-water reaction by itself [6]. Since requirements on fabrication and inspection are relaxed taking into account reduction of R&D risks, the outer tube only acts as protective layer against such target wastage because it is not secured by ISI.

Tube failure propagation has been calculated to assess property protection performance on outer tubes. The evaluation results showed that the failure propagation could be limited to only one-tube propagation by wastage thanks to the outer tube and that the total leakage rate is limited to one double-ended guillotine scale. As a severe condition, failures of both inlet and outlet release valves in the steam-water side are assumed. In this case, the number of failed tubes is 38 and maximum water leak rate is approximately 130kg/s. Quasi steady pressure has been evaluated as shown in Fig. 8. Pressure increase of the secondary coolant system by quasi steady pressure is estimated about 3.0MPa with a calculation using water leak rate based on behavior of realistic tube failure propagation. The reason why behavior of realistic tube failure propagation has been assumed is that tubes are unlikely to be broken at the completely same time by sodium-water reaction. The structural integrity of the boundary of the primary and the secondary coolant system is presumed to be ensured against the above pressure increase [16].

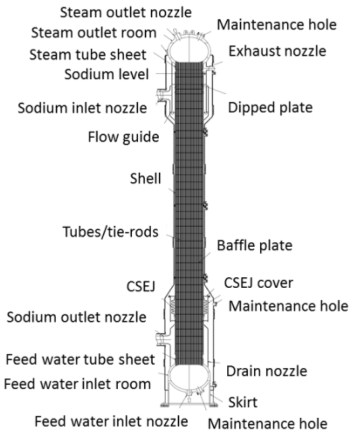


FIG.7 SG

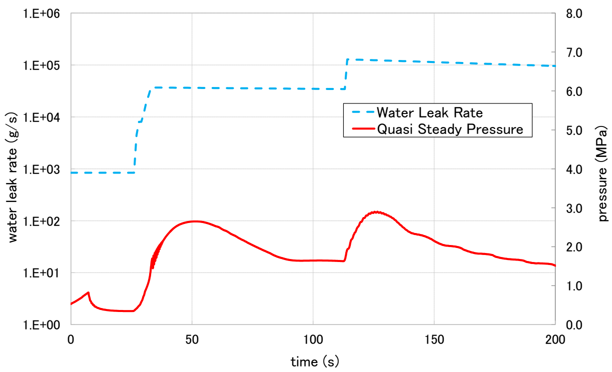


FIG.8 Quasi steady pressure in DEC [16]

3.4. Decay heat removal system

The original decay heat removal system (DHRS) of the advanced SFR in 2010 version consisted of a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS) adopting full natural convection system as shown in Fig. 9 [7]. Involving lessons learned from the 1F accident, requirements on design base DHRS have been modified. In that modification, safety requirements on design extended conditions have been clarified and sodium temperature criteria have been changed taking into account design margin even for design extended conditions. With the new requirements, capacities of DRACS and PRACS have been up dated. For design base conditions, 1.5 circuit operation is required to maintain coolant temperature lower than 600deg-C. However, for design extended conditions, only one circuit is sufficient to remove decay heat with sodium temperature at 650deg-C. With the improved DHRS specification, a long-term station blackout (SBO) transient has been evaluated. In that evaluation, DRACS could remove decay heat during 10 days without control in natural circulation mode. For measures against loss of heat removal system (LOHRS), recovery of original design base DHRSs and auxiliary core-cooling system (ACS) using air as ultimate heat sink (Fig. 10) has been selected. System configuration of ACS with forced circulation of air has been designed. From the viewpoint of diversity, the ACS adopts forced circulation for the air. With that forced circulation system, the location of the air cooler has flexibility and the ACS is free from a high stack for natural circulation capacity. With the new DHRS configuration including ACS, designs of sea water cooling systems and emergency power supply have been updated [17].

3.5.Balance of plant

The advanced SFR has adopted a simple fuel handling system with advanced technologies [2]. An in-vessel fuel handling system consists of a combination of an UIS with a slit and a pantograph type fuel handling machine dramatically reduces the reactor vessel diameter. An ex-vessel storage tank (EVST) provides buffer storage before spent fuel subassemblies stored in the water pool. The EVST could reduce plant outage time due to refueling and periodical inspection and also reduce decay heat of spent fuel. The EVST has enough capacity for full core evacuation in case of emergency [18]. Cooling system of the EVST has also been improved as similar to DHRS. The EVST cooling systems have dipped type heat exchangers inside the EVST as shown in Fig. 11 and there are 4 independent system considering failure and maintenance. Additionally, an alternative cooling system like ACS is also installed as a diversified cooling system.

Taking into account those modification on DHRS and EVST cooling system, emergency power supply has been modified also [19, 17] as shown in Fig. 12. Battery capacity is increased from 2 hours to 24 hours. And an alternative emergency power supply system composed of alternative DGs and lead-acid battery is newly introduced to maintain safety grade components against the long-term SBO. Taking into account experiences on the 1F accident, the alternative emergency power supply system is designed to supply the power for 10 days (240 hours) for reactor cooling, fuel storage cooling, PAM instrumentation and emergency lighting.

For the reactor building, various external events have been evaluated and the reactor building design accommodates those external events. Reactor building tolerance against earth quake, tsunami, strong wind, snow and fire has been evaluated. Based on those evaluations, the advanced SFR reactor building has been improved [20]. Additionally, the reactor building layout is also modified to improve separation of redundant safety components [21].

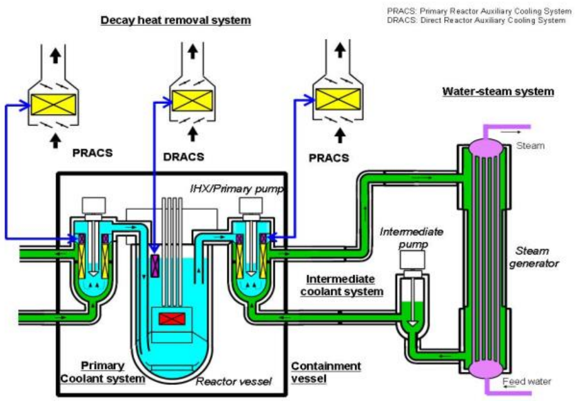


FIG.9 DHRS [7]

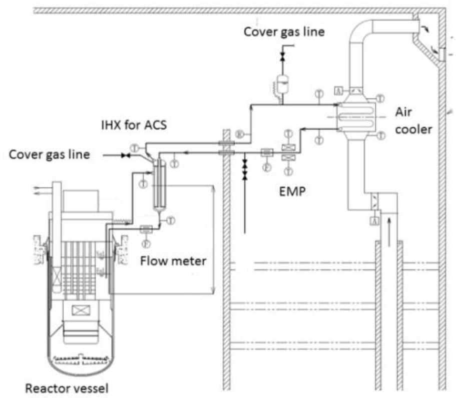


FIG.10 Alternative DHRS [17]

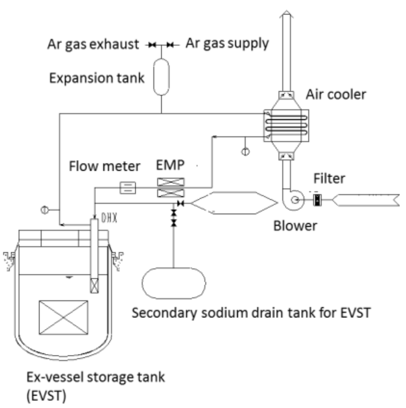


FIG.11 EVST cooling

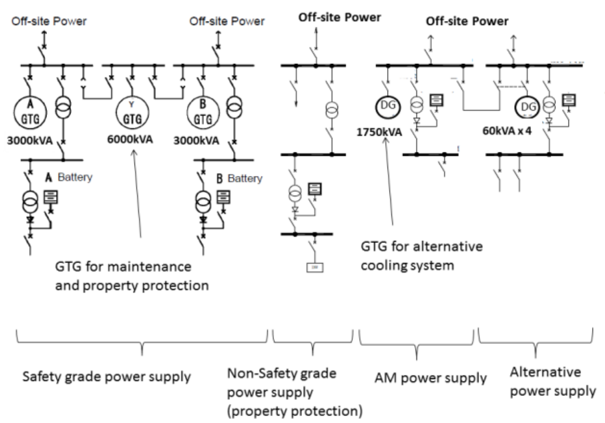


FIG.12 Emergency power supply [17]

4. Impact of Safety Enhancement

Taking into account safety and maintenance improvements to meet the SDC and SDG, mass amount of major components has been evaluated. The total mass amount of nuclear steam supply system (NSSS) is increased from 3707 to 5273 tonne. The major reasons are improvement of seismic and maintenance capability features of the reactor vessel and integrated pump/IHX. For the reactor building volume, the CV volume has increased from 21000 to 46000m^3 mainly due to the installation of ACS and increase of the reactor vessel diameter. As a result of new systems to meet the SDC and SDG, including ACS, the alternative EVST cooling system and improved emergency power sources, the reactor building volume has increased from 257000 to 403000m^3.

TABLE I: Mass amount of the demonstration reactor (tonne)

Item	2010 version	2015 version
Reactor structure	1190	1774
Primary cooling system	859	1683
Secondary cooling system	1658	1816
NSSS total	3707	5273

5. Conclusions

Design studies on a advanced sodium-cooled fast reactor (SFR) considering the safety design criteria (SDC) developed in the generation IV international forum (GIF) was summarized. To meet SDC including the lessons learned from the 1F accident, the heat removal function was enhanced to avoid loss of the heat removal function even if any internal events exceeding design basis or severe external events happen. Several design options for alternative decay heat removal systems have been investigated. An auxiliary core cooling system using air as ultimate heat sink has been selected as an additional cooling system regarding system reliability and diversification. Even though the advanced SFR already adopts seismic isolation system, main component designs have been improved considering revised earthquake conditions. For other external events, design measures for various external events are taken into account. Reactor building design has been improved and important safety components are diversified and located separately to improve independency. Design studies and evaluations on the advanced sodium-cooled reactor have contributed to the development of safety design guidelines (SDG) which is under discussion in the GIF framework.

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