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Enhanced safety characteristics of SMART100 adopting passive safety systems☆

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ABSTRACT

SMART100 is a small sized integral type pressurized water reactor (PWR) with a rated thermal power of (Chung et al., 2015) 365 MW, which adopts various inherent and passive design features to enhance safety. Most of the primary circuit components, such as the core, reactor coolant pumps, steam generators, and a steam pressurizer are contained in a single leak-tight reactor pressure vessel. Due to the integral reactor design, the large pipes connecting the major components are removed. Thus, the possibility of a large break loss of coolant accident (LBLOCA) is inherently eliminated and the natural circulation capability is improved during the transient. Also, SMART100 has inherent design characteristics of the large primary coolant inventory per unit thermal power, low core power density, and increased secondary system design pressure. Due to these design characteristics, SMART100 can enhance the mitigation capability to a wide range of initiating events. In addition to these inherent safety design features, the safety goals of SMART100 are enhanced by the passive safety systems such as the passive residual heat removal system (PRHRS) (Bae et al., 2001) and passive safety injection system (PSIS). To confirm the enhanced safety of SMART100, deterministic safety analyses were performed for the safety related design basis events (SRDBEs). The results of the analyses using an evaluation model of the TASS/SMR-S code and conservative initial/boundary conditions and assumptions show that the acceptance criteria for the fuel integrity, system integrity, and radiation doses specified in the safety review guidelines are well met. Therefore, the passive safety systems of SMART100 adequately mitigate the consequences of all SRDBEs and maintain the plant in a safe shutdown condition without any AC power or operator action for at least 72 h.

1. Introduction

The worldwide technology developments for the prevention of global warming and the replacement of fossil fuels for the reduction of air pollutants are continuously required and the need for nuclear power with inherently low green-house gas emissions is being reexamined. The Middle East and other countries with abundant energy resources are also interested in conserving fossil resources and are actively considering the introduction of nuclear power for diversification of energy sources. Among the power plants operating in the world, electric power of less than 300 MW accounts for 96.5%, and small electrical power supply is trending. Particularly, it is hoped that small nuclear power plants will be introduced by the countries where small size aging fossil power plants have to be replaced or where it is difficult to construct large nuclear power plants due to their geographical and financial conditions. Recently, various advanced types of SMRs (small and medium sized reactors) are under development worldwide for non-electric applications of nuclear energy. Since large nuclear power plants are not economically viable for non-electric applications, these SMRs can be applied for the peaceful use of nuclear energy in the areas of district heating, seawater desalination, industrial process heat generation, and ship propulsion (Kim et al., 2013).

Korea Atomic Energy Research Institute (KAERI) has been developing the integral type of PWR, SMART (System-integrated Modular Advanced ReacTor), from 1997 for small-scale power generation and seawater desalination. The standard design of SMART with the core thermal power of 330 MW adopting proven technologies and innovative design features was launched at KAERI in 2009. Together with the Korea Electric Power Corporation (KEPCO) consortium, KAERI submitted a standard safety analysis report (SSAR) (KEPCO & KAERI, 2010) for the application of standard design approval (SDA) in December 2010 (Kim

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Nomenclature

AC	Alternating Current
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrence
CCWS	Component Cooling Water System
CDL	CPRSS Discharge Line
CHRS	CPRSS Heat Removal System
CHX	CPRSS Heat Exchanger
CMT	Core Makeup Tank
CMTAS	Core Makeup Tank Actuation Signal
CPRSS	Containment Pressure and Radioactivity Suppression System
CRA	Control Rod Assembly
CRL	CPRSS Return Line
CSL	CPRSS Steam Line
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
DC	Direct Current
DNBR	Departure from Nucleate Boiling Ratio
ECT	Emergency Cooldown Tank
EDG	Emergency Diesel Generator
FIV	Feedwater Isolation Valve
FLB	Feedwater Line Break
FMHA	Flow Mixing Header Assembly
IRWST	In-containment Refueling Water Storage Tank
LCA	Lower Containment Area
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MDNBR	Minimum DNBR
MSLB	Main Steam Line Break
MSIV	Main Steam Isolation Valve
MTC	Moderator Temperature Coefficient
PA	Postulated Accident
PBL	Pressure Balance Line
PLCSMF	Pressurizer Level Control System Malfunction
PRHRAS	Passive Residual Heat Removal Actuation Signal
PRHRS	Passive Residual Heat Removal System
PSAR	Preliminary Safety Analysis Report
PSIS	Passive Safety Injection System
PSV	Pressurizer Safety Valve
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REA	Rod Ejection Accident
RPVA	Reactor Pressure Vessel Assembly
RRT	Radioactive-material Removal Tank
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break LOCA
SG	Steam Generator
SDA	Standard Design Approval
SGTR	Steam Generator Tube Rupture
SIL	Safety Injection Line
SIT	Safety Injection Tank
SITAS	Safety Injection Tank Actuation Signal
SMR	Small and Medium size Reactor
SRDBE	Safety Related Design Basis Event
SRG	Safety Review Guidelines
SRP	Standard Review Plan
SSAR	Standard Safety Analysis Report
TLOF	Total Loss of Reactor Coolant Flow

Table 1
Differences in major design features.

	SMART SDA	SMART100
Plant Power	330 MWt (~100 MWe)	365 MWt (~110 MWe)
Safety Systems	Active & Passive	Fully passive
PRHRS	4 trains (36 h)	4 trains** (72 h)
Safety Injection System	4 trains* (Active SI pump)	4 trains** (Passive CMT, SIT)
Shutdown Cooling System	2 trains SCS (Safety)	2 trains CCWS (Non-safety)
Containment Protection	2 trains CSS (Active)	CPRSS (Passive)
EDG	2 100% Active, Safety	Non-safety DGs
Single Failure of Safety System	1 train failure	No train failure
Operator Response Time	30 m	72 h
Plant Cooldown using Safety System	Cold shutdown condition(90 °C)	Safe shutdown condition (215 °C)

* Electrically independent 2 trains and mechanically independent 4 trains.
** Electrically and mechanically independent 4 trains and eliminate the single train failure by 2 valves in parallel lines for each train.

et al., 2013). Through a thorough licensing review process by a Korean nuclear regulatory body, SMART obtained SDA from the nuclear safety and security commission (NSSC) in July 2012 that made SMART the first licensed advanced integral reactor in the world (Kim et al., 2014).

To cope with the increased demands for safety after Fukushima Daiichi accident in March 2011, KAERI carried out safety enhancement research and development (R&D) adopting the fully passive safety systems into the standard design of SMART from March 2012 to February 2016. Various validation tests for the SMART passive safety systems were included in this R&D. Through these validation tests, the performances of passive safety injection systems were confirmed and the code validation calculations were done using the test data.

In March 2015, South Korea and Saudi Arabia signed a memorandum of understanding to build a SMART partnership between both countries for the construction of SMART in Saudi Arabia. It has pioneered the first overseas export of small-sized nuclear power plants under development, and it is able to preoccupy the small nuclear power plant export market and establish a foundation for joint venture with Saudi Arabia in the Middle East and North Africa. From December 2015 to November 2018, South Korea and Saudi Arabia jointly completed the SMART pre-project engineering to prepare a preliminary safety analysis report (PSAR) for the construction license application of SMART Units 1&2 (hereinafter referred to as "SMART100") (KAERI & K.A.CARE, 2018).

Korea Hydro & Nuclear Power Co., Ltd. (KHNP)/KAERI/King Abdullah City for Atomic and Renewable Energy (K.A.CARE) submitted an SSAR of the SMART100 design (KHNP et al., 2019) as an attachment of application for standard design approval to the NSSC.

The design of SMART100 was upgraded from the standard design of SMART (SMART SDA) by increasing the core thermal power from 330 MW to 365 MW, adopting the fully passive safety systems, and optimizing the balance of plant design. Table 1 shows the differences in major design features of SMART SDA and SMART100. As shown in this Table, the design of SMART100 was changed as follows: The performance of PRHRS was extended from 36 h to 72 h without operator intervention. The PRHRS and PSIS are composed of 4 electrically and mechanically independent trains. To prevent the single train failure of the PRHRS and PSIS, each train is composed of two parallel pipings and two valves are installed in each parallel piping. The passive safety injection system composed of core makeup tank (CMT) and safety injection tank (SIT) replaces the former active safety injection system. The containment protection is performed by the passive containment

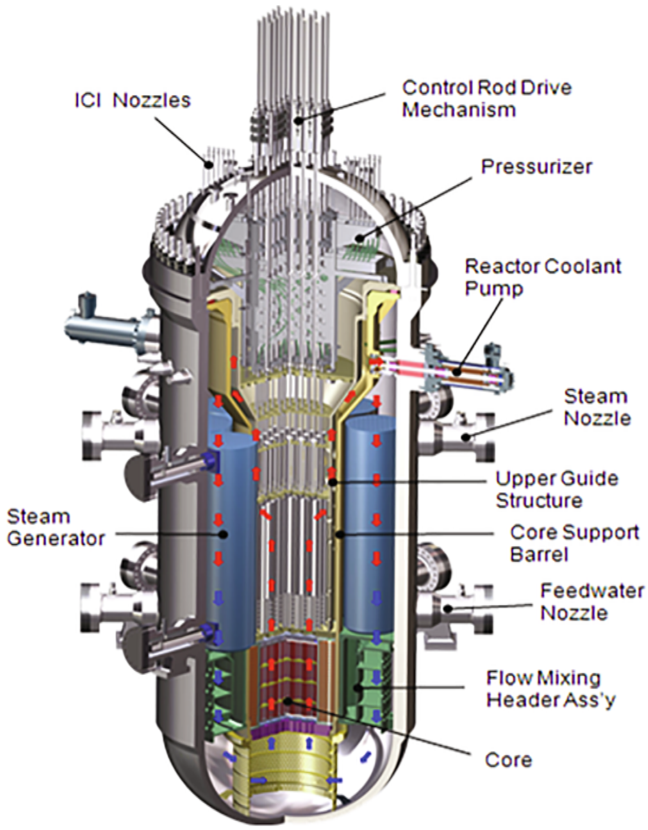

Fig. 1. SMART100 reactor pressure vessel assembly.

pressure and radioactivity suppression system (CPRSS) instead of the former active containment spray system (CSS). The active shutdown cooling system (SCS) is changed from the safety system to the non-safety component cooling water system (CCWS). There is no emergency diesel generator (EDG). All safety systems can be operated not depending on the AC power for 72 h. Four trains of safety-grade emergency batteries provide necessary DC power for valve actuation and post-accident monitoring. The passive safety system can maintain the SMART100 plant in a safe shutdown condition following the design basis accidents without AC power or operator action for at least 72 h (USNRC, 1994).

The SMART100 adopts the design characteristics containing most of the primary components, such as a core, four canned motor reactor coolant pumps (RCPs), eight helically coiled once-through steam generators (SGs), and a pressurizer (PZR) in a single leak-tight reactor pressure vessel assembly (RPVA) (Bae et al., 2001), as shown in Fig. 1.

The reactor coolant flows upward through the core, upper guide structure, inside core support barrel and RCP suction region. Also, the pumped reactor coolant flows downward through the SG shell side, flow mixing header assembly (FMHA), flow skirt, lower plenum, and then into the core. The secondary feedwater supplied to the SG bottom region flows upward through the SG tube side by removing the heat generated in reactor coolant system (RCS). The superheated steam exits at the top of the SG region.

Fig. 2 shows a schematic of SMART100 adopting fully passive safety systems (KHNP et al., 2019). The SMART100 nuclear steam supply system consists of the RCS forming a reactor coolant pressure boundary, secondary system, chemical and volume control system (CVCS), CCWS, PRHRS, PSIS, automatic depressurization system (ADS), CPRSS, and so on. The PRHRS, PSIS, ADS, and CPRSS are passive safety systems, and the others are non-safety systems. The passive safety systems maintain the SMART100 plant in a safe shutdown condition following a design basis event without AC power or operator action for at least 72 h.

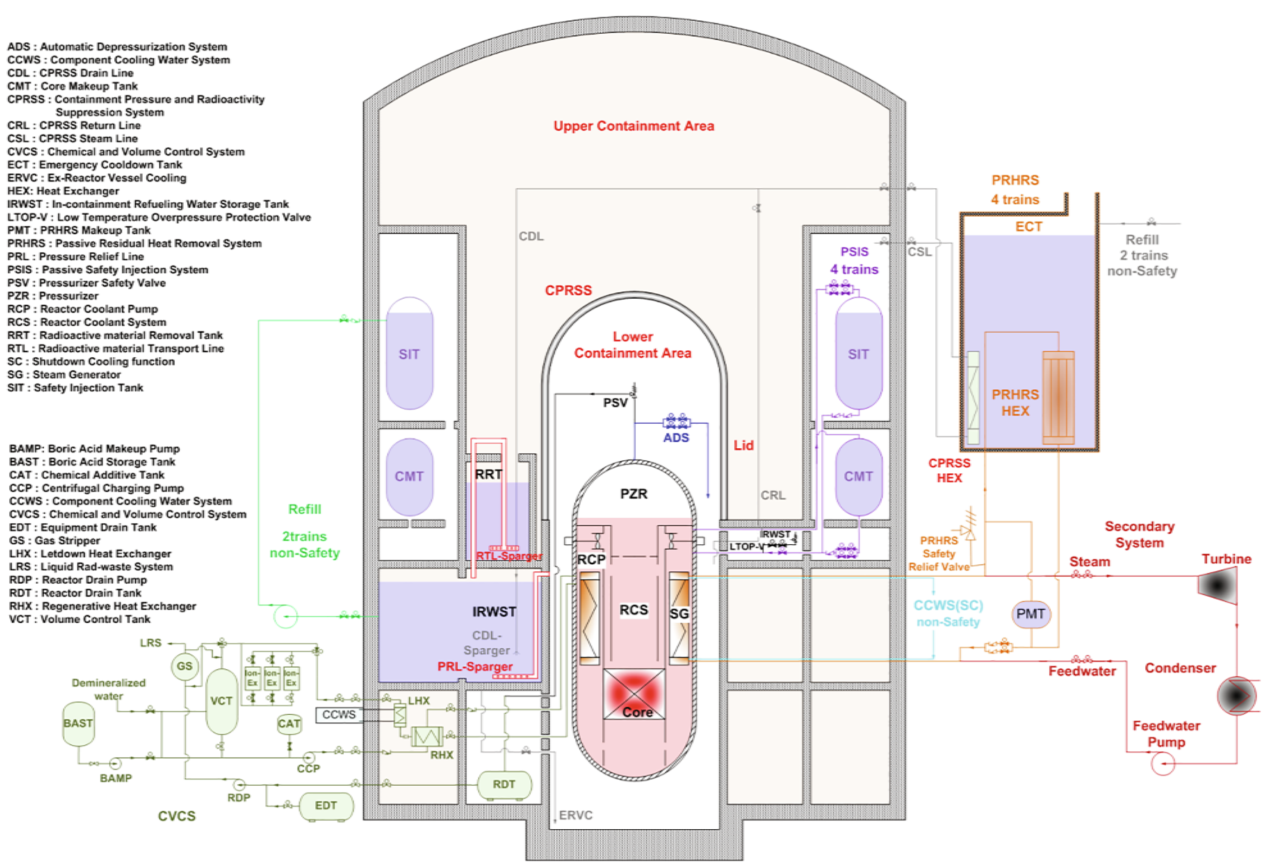

Fig. 2. Passive safety systems of SMART100.

The RCS transfers core heat to the secondary system through the SGs and plays a role of a barrier that prevents the release of reactor coolant and radioactive materials to the reactor containment. The forced circulation flow of the reactor coolant is formed by four RCPs installed at the upper side of the reactor vessel. The RCS and its supporting systems are designed with sufficient core cooling margin for protecting the reactor core from damage during all normal operation and anticipated operational occurrence (AOO). The reactor overpressure protection is achieved by the PZR safety valves (PSVs).

The PRHRS is connected to the main steam line and feedwater line outside containment, which removes the core decay heat in emergency situations where normal steam extraction or feedwater supply is unavailable after reactor trip (Bae et al., 2007). The PRHRS consists of four electrically and mechanically independent trains, and each train is composed of one emergency cooldown tank (ECT), one PRHRS heat exchanger, one PRHRS makeup tank, related valves, connecting pipes and instruments. Both a pair of check valves and a pair of isolation valves are installed on parallel lines in each train of the PRHRS to eliminate the single train failure. The PRHRS cools the RCS by the natural circulation flow developed by the elevation difference between the SG and PRHRS heat exchanger and the density difference. The safety function of the PRHRS is maintained continuously for a long-term period when an ECT is refilled periodically by a PRHRS ECT refilling system. Two external connection lines at ground level are provided so that two ECT makeup tank can be replenished from external water sources and each ECT makeup tank are connected to two ECTs. The opening set pressure of the PRHRS safety relief valve is the same as the design pressure of the RCS. Thus, the radioactive material release through the PRHRS safety relief valve is prevented in the SGTR accident.

The PSIS is connected to the upper side of reactor vessel, which provides emergency core cooling following the postulated design basis accidents. The PSIS consists of four electrically and mechanically independent trains, and each train is composed of one CMT, one SIT, one safety injection line (SIL), one pressure balance line (PBL), related valves, and instruments. In addition, each train consists of two parallel flow paths to prevent a system function loss from a single failure of isolation valves required to operate in an accident, and two isolation valves are installed on each parallel flow path in series. The CMT is fully filled with highly concentrated borated water during normal operation and is isolated from the RCS by the isolation valves and check valves in the SIL (Chun et al., 2014). The borated water in the CMT is injected into the RCS when the isolation valves in the SIL opened by the CMT actuation signal (CMTAS) such as the low pressurizer pressure signal (Chun et al., 2014). The SIT is fully filled with highly concentrated borated water and air at atmospheric pressure during normal operation and is isolated from the RCS by the isolation valves in the PBL and check valves in the SIL. The borated water in the SIT is injected into the RCS when the isolation valves in the PBL opened by the SIT actuation signal (SITAS) such as the low-low pressurizer pressure signal. The valves isolating individual trains of the PSIS receive emergency power from onsite or offsite power sources or DC batteries. An emergency battery system consists of four independent power systems. The relief valve, which is installed at the SIL, provides a low temperature overpressure protection function of the RCS. The PSIS keeps the reactor in a safe shutdown by periodically refilling the SIT using the non-safety refill system from in-containment refueling water storage tank (IRWST) after 72 h following a LOCA.

The ADS consists of two independent trains. The valves and pipes in each train are placed in parallel considering a single failure, and two valves are installed in series on each parallel line to ensure the isolation in the normal operation. The ADS valves are opened when the CMT water level reaches low level set-point. The ADS rapidly depressurizes the RCS to activate SIT earlier for the LOCA. Also it can be manually operated for a total loss of secondary heat removal accident for feed and bleed function with PSIS.

The CPRSS is composed of CPRSS lid, pressure relief lines (PRLs) and PRL-spargers, an IRWST, radioactive material transport lines (RTLs) and RTL-spargers, radioactive material removal tank (RRT), CPRSS Heat Removal System (CHRS) as a subsystem of the CPRSS, and instruments. The CHRS consists of four mechanically independent trains, and each train is composed of one CPRSS heat exchanger (CHX), a CPRSS steam line (CSL), a CPRSS discharge line (CDL) and CDL-spargers, a CPRSS return line (CRL), one ECT, and instruments and valves. The ECT is shared with the PRHRS. In the CPRSS, the lower containment area (LCA) is connected to the IRWST via the PRL and the PRL-sparger, the IRWST is connected to the RRT via the RTL and the RTL-sparger, and the RRT is connected to the upper containment area (UCA) via the RRT vent located at the top of the RRT. Also the SIT compartment in the LCA is connected to the IRWST via the CSL, CHX, and the CDL during 72 h following the accident and then is connected to the gas area in the LCA below the bottom of the CHX via the CSL, CHX, and the CRL after 72 h. The steam and the fission products released into the LCA after the LOCA and main steam line break (MSLB) are discharged to the IRWST through the PRL and PRL-spargers by the pressure difference between the LCA and IRWST. The containment pressure is suppressed by the steam condensation in the CHX and IRWST. The steam and non-condensable gas mixture and the fission products in the IRWST gas area are introduced to the RRT through the RTL-sparger. The steam and fission products are condensed and dissolved in the RRT. And the non-condensable gas is discharged into the UCA through the RRT vent located at the top of the RRT. Thus, the CPRSS suppresses the pressure and temperature in the containment area following accidents such as LOCA and MSLB, and removes the radioactive fission products from the containment area.

The shutdown cooling function cools down the RCS from the shutdown cooling entry temperature to the refueling temperature within 96 h after reactor shutdown and maintains the RCS at the refueling temperature for a long-term period. The CCWS supplies the component cooling water to four SGs for shutdown cooling function.

The CVCS performs cleanup operation for keeping water quality and purity of the reactor coolant intact. The system also compensates the reactor coolant leakage from the RCS, and provides continuous measurement methods for boron concentration and radioactivity level of fission products during normal operation of the plant. The system supplies makeup water to auxiliary equipment, and provides appropriate volume control methods.

The following sections describe the inherent and passive safety characteristics of the SMART100 design and present the safety analysis results for the SRDBEs.

2. Safety characteristics of SMART100

SMART100 adopts various safety enhancement design features to lower the core damage frequency and radiological consequences compared with the commercial PWRs. Due to the integral arrangement of the major primary components in a single reactor pressure vessel, the large RCS pipes are removed and the RCS pressure loss is decreased. Thus, the possibility of an LBLOCA is inherently eliminated, and the natural circulation capability is improved during the transients and accidents. The large primary coolant inventory per unit thermal power compared with the commercial PWRs increases the accident mitigation capability during the LOCA and feedwater line break (FLB), and it causes a slow RCS cooling due to a large thermal inertia during MSLB. The design pressures of the secondary system and PRHRS are the same as that of the primary system. Thus, the possibility of an SGTR and secondary system pipe break is very low. The core power density of SMART100 is about one-third of the commercial reactors. Thus, the fuel heatup is not severe, which in turn causes an increase in the core thermal margin. SMART100 has a small SG secondary side water inventory by adopting a helically coiled once-through SG in which secondary coolant flows inside the tubes. Thus, the possibility of a post-trip re-criticality of the MSLB is reduced. SMART100 adopts a canned motor RCP which eliminates the small break loss of coolant accident (SBLOCA) caused by an RCP seal leakage. SMART100 also adopts the innovative FMHA which mixes the coolant sufficiently in the case of an asymmetric cooling accident such as MSLB and FLB. Thus, the possibility of a core local power increase is very low.

In addition to these inherent safety design features, the safety goals of (Bae et al., 2001) SMART100 are enhanced by the passive safety systems such as the PRHRS and PSIS.

The PRHRS removes the residual heat from the core and sensible heat in the RCS if the normal RCS cooling via secondary system is unavailable after reactor trip. The PRHRS is designed to cool the RCS to the safe shutdown condition within 36 h after the accident initiation and maintains the safe shutdown condition for at least another 36 h. Therefore, the safety function is performed for at least 72 h without AC power or any corrective action by the operator. The passive residual heat removal actuation signal (PRHRAS) is generated by a low main steam line pressure signal, high main steam line pressure signal, low feedwater flow signal, high feedwater flow signal, low PZR level signal, high PZR level signal, high LCA pressure signal, high PZR pressure signal, or low SG inlet temperature increase signal. When the PRHRAS is generated, the main steam isolation valves (MSIVs) and feedwater isolation valves (FIVs) begin to close and the PRHRS outlet isolation valve begins to open.

The PSIS injects the borated water into the RCS by gravity (Chun et al., 2014) head to prevent core uncovery in case of an SBLOCA and to increase shutdown margin following an MSLB. The cooling water in the CMT provides makeup and boration functions to the RCS during the early stage of SBLOCA and non-LOCA such as MSLB. The CMTAS is generated by a high LCA pressure signal, low PZR pressure signal, low-low SG inlet temperature increase signal, or PRHRAS. Then, the isolation valves in the SIL open to inject the borated water from the CMT into the RCS. The SITAS is generated by a low-low PZR pressure signal, or low-low SG inlet temperature increase signal. Then, the isolation valves on the PBL opens and the cooling water in the SIT is injected into the RCS by gravity when steam from the RCS is injected into the SIT through the PBL, and the internal pressure of the RCS and the SIT reaches the equilibrium state (Chun et al., 2014).

3. Safety analysis for the major design basis events

The safety of the SMART100 adopting the inherent safety design features and passive safety systems is assessed for the SRDBEs (Bae et al., 2001). The SRDBEs are classified in accordance with the Regulatory Guide 1.70 (USNRC, 1978), standard review plan (SRP, NUREG-0800) (USNRC, 2007), and standard review guidelines (SRG) (KINS, 2014a) set by the regulatory body. The SRDBEs define the transients and accidents postulated in the SMART100 safety analysis to classify all the unplanned occurrences that shall be accommodated by the SMART100 design and mitigated by the actuation of the reactor protection system and engineered safety features.

The initiating design basis events of SMART100 are categorized by frequency of occurrence and by type. According to the SRP, the initiating events are classified into two kinds, AOO or postulated accident (PA). The initiating events are also categorized as seven different types depending upon the resulting effects on the SMART100 plant after such an event occurs. Although SMART100 has a design and operational characteristics of an integral type reactor, it has almost similar design basis events to as well as different from those of the commercial loop-type PWR plants. Some events are excluded or included due to the design characteristics different from the conventional loop type PWR. Considering the SMART100 specific design features, the events of an inadvertent opening of a steam generator relief or safety valve and an LBLOCA are eliminated from SSAR Sections 15.1 and 15.6, respectively. Instead, the events of an inadvertent opening of a turbine bypass valve, the improper operation of a PRHRS, and the inadvertent opening of a PRHRS safety relief valve in SSAR section 15.1 and the inadvertent operation of the automatic depressurization system in SSAR section 15.6 are additionally accounted for.

Table 2
Representative events for each event category.

Event category	Representative event
15.1 Increase in Heat Removal by the Secondary System	MSLB
15.2 Decrease in Heat Removal by the Secondary System	FLB
15.3 Decrease in Reactor Coolant System Flow Rate	TLOF
15.4 Reactivity and Power Distribution Anomalies	REA
15.5 Increase in Reactor Coolant Inventory	PLCSMF
15.6 Decrease in Reactor Coolant Inventory	SBLOCA

In order to confirm the enhanced safety of SMART100, deterministic safety analyses were performed for the SRDBEs selected for its specific design and operation characteristics. As shown in Table 2, the safety analysis results for the representative events in each section of SSAR 15.1 to 15.6 such as MSLB, FLB, total loss of reactor coolant flow (TLOF), control rod assembly ejection accident (REA), pressurizer level control system malfunction (PLCSMF), and SBLOCA are presented in this paper.

3.1. Safety analysis methods

The computer code used for the safety analysis is TASS/SMR-S (Kim, 2017), which has been developed at KAERI for the performance and safety analyses (Bae et al., 2001) of SMART100. This code calculates the system thermal–hydraulic response, fuel rod departure from nucleate boiling ratio (DNBR), and fuel rod temperature under a full range of operating conditions. The basic code structure adopts a one-dimensional geometry. A node and flow-path network models the system responses. The node encloses the control volumes, which represent the fluid mass and energy. The flow-path connecting the nodes represents the fluid momentum and has no volume. It uses the fundamental conservation equations of liquid mass, mixture mass, non-condensable gas mass, mixture momentum, gas energy, and mixture energy for the non-equilibrium two-phase flow (Chung et al., 2015a, 2015b). The difference between the gas velocity and the liquid velocity is calculated using the drift-flux model. A number of SMART100-specific models reflecting the design characteristics such as the helically coiled SG, the steam PZR, and the heat exchanger in the PRHRS are addressed in the code (Chung et al., 2012). The core model includes the core power and core heat transfer.

The core power is calculated by a point kinetics model, which is used to describe the time dependent response of the core power to the reactivity feedbacks. The fission power input to the fuel is calculated from the reactor kinetic equations with six delayed neutron groups. The decay power is based on the fission product inventory, which would result from a long-term steady state operation at a specified initial power level.

The TASS/SMR-S code has been verified using the various conceptual or analytical verification problems and validated using the various separate effect tests (SETs) and integral effect tests (IETs) data (Chung et al., 2016).

A deterministic safety analysis method is used. By using the conservative model and conservative initial/boundary conditions and assumptions, conservative analysis results are calculated. The initial core power and feedwater flow rate are assumed as 103% of the nominal values considering the measurement uncertainties (Bae et al., 2001). In the analyses of each event, sensitivity analyses are performed to select the limiting initial condition from the viewpoint of safety criteria among various initial condition ranges of the core inlet fluid temperature, PZR pressure, RCS coolant flow rate, PZR water level, and axial offset. Depending on the cooldown or heatup transient characteristics, the moderator and fuel temperature coefficients are selected as a combination of the least or most negative ones for conservative results. The scram reactivity insertion curve is for the limiting bottom-skewed axial power shape, and the minimum shutdown rod worth with the most

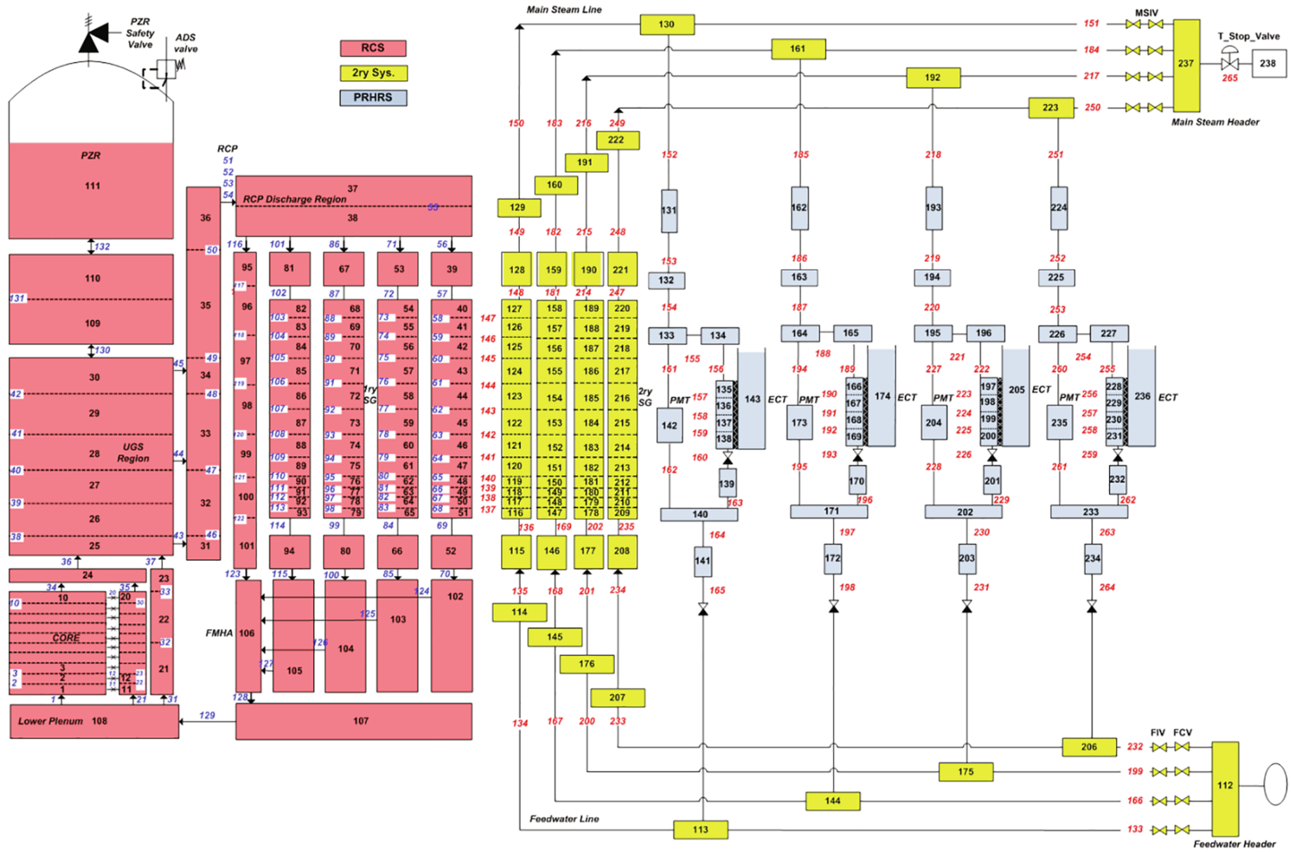

Fig. 3. TASS/SMR-S Nodalization for SMART100 System.

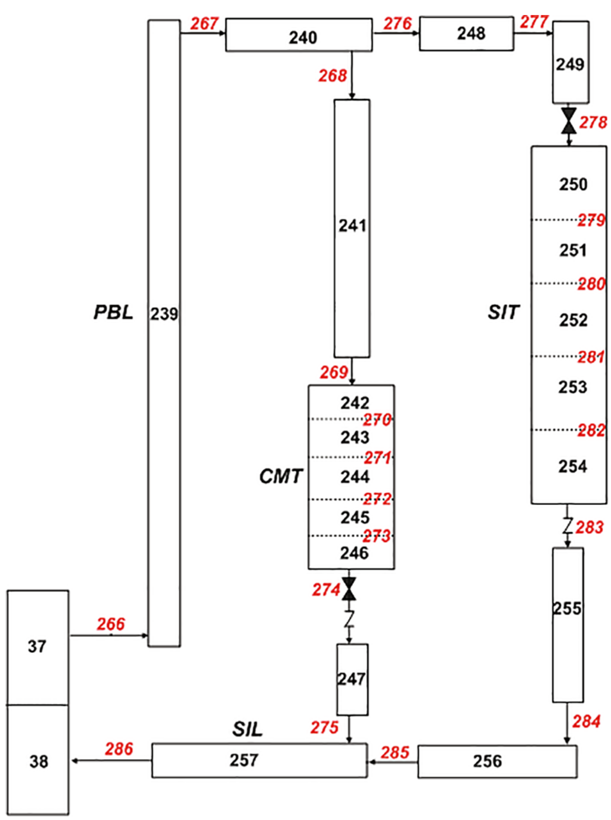

Fig. 4. TASS/SMR-S Nodalization for PSIS (1 of 4 trains).

reactive rod stuck out is considered (Bae et al., 2001). The decay heat curve used is a conservative ANS-71 decay heat curve (ANS, 1971) with a 1.2 multiplication factor. For the conservative analysis from the inventory loss point of view, the break flow is maximized by assuming the reactor containment area temperature and pressure to be constant as the initial atmospheric conditions throughout the accident (Chun et al., 2014). Following the passive safety system performance requirement, the operator action is not considered until 72 h after the event (USNRC, 1994).

Fig. 3 shows the TASS/SMR-S nodalization for the RCS, secondary system, and PRHRS of SMART100. The orange colored nodes represent the RCS. The core, upper plenum, PZR, RCP suction and discharge, shell side of SG, FMHA, and lower plenum regions are modeled, respectively. The yellow colored nodes and grey colored nodes represent the secondary system and PRHRS, respectively. Four trains of feedwater line, tube side of SG, main steam line, and PRHRS are modeled independently. Also, all of the heat structures of the system are modeled (Bae et al., 2001).

Fig. 4 shows the TASS/SMR-S nodalization for the one of four trains of PSIS. Each train composed of PBL, CMT, SIT, and SIL is connected to the RCP discharge regions (nodes 37, 38 in Fig. 3).

3.2. *Safety analysis results*

Among the events analyzed in this paper, the total loss of reactor coolant flow and pressurizer level control system malfunction events are AOOs, and the other events such as a main steam line break accident, feedwater line break accident, control rod assembly ejection accident, and small break loss-of-coolant accident are PAs. The acceptance criteria adopted for the safety analysis for SMART100 are as follows:

Acceptance Criteria for AOOs (USNRC, 2007):

i. The pressures in the RCS and main steam system should be maintained below 110% of the design pressures.
ii. The fuel cladding integrity shall be maintained by ensuring that the minimum DNBR (MDNBR) remains above the 95/95 DNBR limit.

Acceptance Criteria for Postulated Accidents (USNRC, 2007):

The following are the specific criteria for PAs: Individual sections of the SRP specify additional criteria pertaining to specific postulated accidents.

i. The pressures in the RCS and main steam system should be maintained below the acceptable design limits, considering potential brittle, as well as ductile, failures.
ii. A coolable core geometry should be maintained.
iii. The release of radioactive material shall not result in offsite doses in excess of the acceptance criteria specified in SRP.
iv. For the REA and SBLOCA, additional safety criteria specified in the SRG 15.4.8 (KINS, 2014b) and 10CFR50.46 (USNRC, 1974) should be met, respectively.

3.2.1. Main steam line break

A main steam line break accident, which occurs as a result of thermal stress or cracking in the main steam line, is a limiting accident for an increase in heat removal by the secondary system (Chung et al., 2012). A pipe break in the main steam system causes an excessive increase in the steam flow rate and a rapid decrease in the secondary system pressure, which causes an increase in heat removal by the secondary system and decrease in RCS coolant temperature and pressure. The positive reactivity is inserted due to the negative moderator temperature coefficient (MTC), which causes an increase in core power and heat flux, and thus the DNBR degradation. The main parameters of concern related with safety for this accident are pre-trip fuel degradation, offsite doses and a post-trip return to power.

The reactor trip may be occurred by one of the several available reactor trip signals such as a low RCP speed, low main steam line pressure, or variable overpower depending on the assumptions considered in each event case. The boron injection from the CMT and SIT causes the core reactivity to decrease. The PRHRAS is generated by a low PZR level after the reactor trip signal generation. The core decay heat and residual heat of RCS is removed by the natural circulation of PRHRS operation. The reactivity increase by the decreased RCS temperature is significantly less than the shutdown rod worth, and thus a post-trip return to power condition is not reached. An asymmetric cooldown is occurred by a one section steam line break. However, the coolant is fully mixed flowing through the FMHA and flow skirt. Thus, there is a low possibility of a local core power increase.

Transient behaviors of the RCS pressure and MDNBR following a large main steam line break inside containment during full power operation with a loss of offsite power concurrent with the initiation of event are shown in Figs. 5 and 6. The RCS pressure decreases continuously by the increase in heat removal by the secondary system. A decrease in the reactor coolant temperature at the core inlet in combination with assumed most negative MTC causes an increase in the core power (Chung et al., 2003). The DNBR decreases slowly as the core power increases at the beginning of the transient. As the RCPs coast down by the assumed loss of offsite power (LOOP), the DNBR decreases abruptly by a mismatch between the core power and core mass flow (Chung et al., 2003). The DNBR rises abruptly when the reactor trip occurs and increases continuously as the natural circulation in the RCS and PRHRS is fully established. The MDNBR is well above the specified acceptable fuel design limit (SAFDL) of DNBR. The offsite doses resulted from the MSLB are compliant with the acceptance criteria specified in Section 15.1.5 of SRP.

3.2.2. Feedwater line break

A feedwater line break accident, which occurs as a result of thermal stress or cracking in the feedwater pipe, is a limiting accident for a decrease in the heat removal by the secondary system (Chung et al., 2012).

A pipe break in the feedwater system causes an immediate decrease in feedwater flow rate, which causes a decrease in heat removal by the secondary system and an increase in RCS coolant temperature and pressure. A negative reactivity is inserted due to assumed least negative MTC, which causes a core power decrease. The main parameters of concern related with safety for this accident are the maximum system pressure and offsite doses.

The RCS temperature and pressure increase when the heat removal by the SG decreases following an FLB accident inside and outside the reactor containment. The reactor trip signal and PRHRAS are generated by the high PZR pressure. By the PRHRAS, the MSIVs and FIVs begin to close and the PRHRS outlet isolation valves begin to open, which isolate the SGs from the turbine and connect the SGs to the PRHRS. The PZR safety valve is opened when the PZR pressure reaches its opening setpoint, and closed when the PZR pressure dropped to its closing setpoint.

Transient behaviors of the RCS pressure and MDNBR following an FLB are shown in Figs. 5 and 6. As shown in these Figures, the maximum RCS pressure is less than 110% of the design values, and the MDNBR is above the DNBR SAFDL, respectively. Due to the no fuel failure and the early isolation of secondary system by the PRHRAS, the offsite doses caused by the FLB are compliant with the acceptance criteria specified in section 15.0.1 of SRP.

3.2.3. Total loss of reactor coolant flow

A total loss of reactor coolant flow is a typical event for a decrease in the reactor coolant flow rate and caused by a complete loss of electrical power supply to all the RCPs in operation. This event results in a complete loss of forced circulation of primary coolant flow, and thus produces the largest degradation in the DNBR margin compared with a partial loss of forced reactor coolant flow events (Bae et al., 2001). The simultaneous loss of electrical power to all four RCPs causes an immediate decrease in core flow rate, which causes a decrease in core heat transfer and an increase in RCS coolant temperature and pressure. A negative reactivity insertion occurs due to a negative MTC, which causes core power and heat flux decreases. The main parameters of concern related with safety for this event are the MDNBR and maximum system pressure.

Simultaneously with the LOOP, the turbine trips, RCPs start to coast down, and the feedwater pump stops (Bae et al., 2001). The reactor trip signal is generated by a low RCP speed. The RCP coast down time is short by the design feature of the canned motor RCP. The coolant flow decrease rate is greater than the core power decrease rate, and thus the MDNBR is reached in a few seconds after the event. The PRHRAS is generated by a low feedwater flow rate, and the RCS is cooled down slowly by the natural circulation of the PRHRS (Chun et al., 2014).

Transient behaviors of the system pressure and MDNBR following a complete loss of electrical power to all four RCPs are shown in Figs. 5 and 6. As shown in these Figures, the maximum RCS pressure is less than 110% of the design pressure, and the MDNBR is above the DNBR SAFDL, respectively. Therefore, the acceptance criteria regarding the system integrity and fuel cladding integrity are met.

3.2.4. CVCS malfunction - PLCS malfunction

A CVCS malfunction caused by a PLCSMF with a LOOP is a limiting AOO for an increase in RCS inventory. This event is initiated when the PZR level controller fails at low level, and a low level signal is transmitted to the controller. In this situation, the PLCS controller closes the letdown orifice isolation valve and opens the charging flow control valve to their minimum and maximum positions, respectively. Thus, the maximum rate of coolant addition to the RCS occurs, which causes the increases in the RCS inventory and the PZR water level and pressure. The

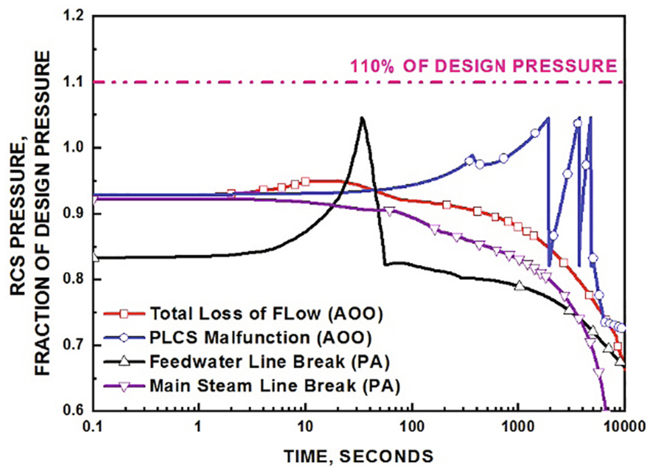

Fig. 5. Normalized RCS pressures for major SRDBEs.

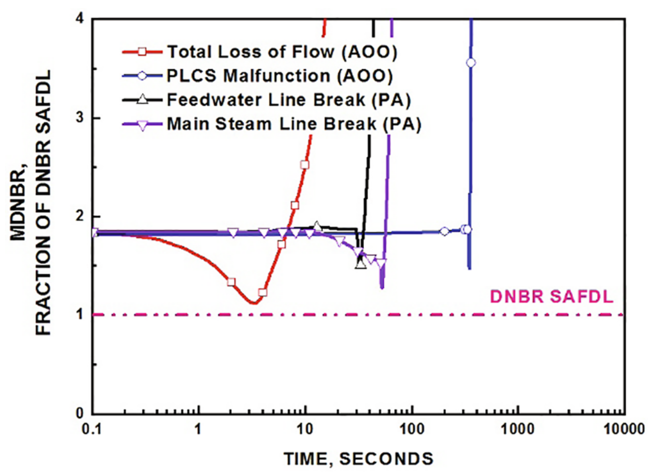

Fig. 6. Normalized MDNBRs for major SRDBEs.

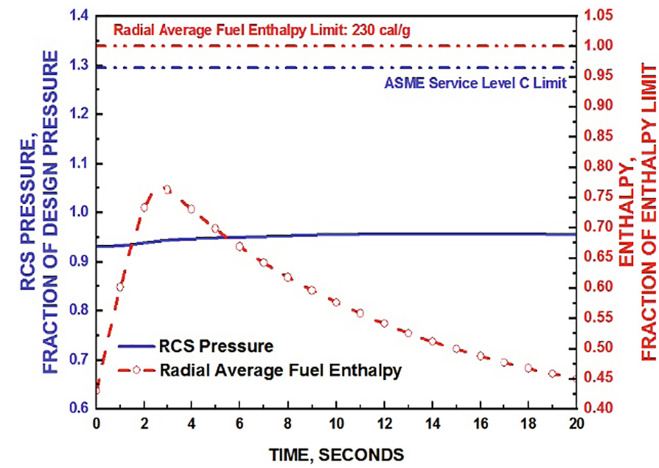

Fig. 7. Normalized RCS pressure and radial average fuel enthalpy for REA.

main parameters of concern related with safety for this event are the maximum system pressure and MDNBR.

The reactor trip signal is generated by a high PZR pressure signal. Simultaneously with the reactor trip, turbine trip and LOOP are assumed to occur. Upon a LOOP, RCPs start to coast down, and feedwater pumps stop. The PRHRAS is generated by a low feedwater flow rate and the RCS is cooled down by the natural circulation of the PRHRS (Chun et al., 2014).

Transient behaviors of the RCS pressure and MDNBR following a PLCSMF with a LOOP at turbine trip are presented in Figs. 5 and 6. The PZR pressure keeps increasing even after the reactor trip by the high PZR pressure signal and reaches PSV opening setpoint. After that time on, the PZR pressure shows decreasing/increasing characteristics due to the repeated opening and closing of PSVs until the charging flow is terminated by the CVCS isolation by the high PZR water level signal. The MDNBR reaches at the minimum value when the RCP stops by the LOOP with reactor trip. As shown in these Figures, the maximum RCS pressure is less than 110% of the design pressure, and the MDNBR is above the DNBR SAFDL, respectively. Therefore, the acceptance criteria regarding the system integrity and fuel cladding integrity are met.

3.2.5. Control rod assembly ejection

A control rod assembly ejection accident is a limiting accident for reactivity and power distribution anomalies. The circumferential rupture of the control rod drive mechanism housing or nozzle causes an instantaneous ejection of a control rod assembly within 0.05 s. This causes an instantaneous addition of positive reactivity and a rapid increase in core power and fuel temperature. The reactor trip signal is generated by the variable over power. By the negative reactivity insertion due to the Doppler feedback, the core power and heat flux decrease. Core boiling occurs due to the increased core heat flux, which in turn causes an increase in RCS coolant temperature and pressure. The main parameters of concern related with safety for this accident are the fuel enthalpy, maximum system pressure, and offsite doses. Simultaneously with the reactor trip, the turbine trips, RCPs start to coast down, and feedwater pumps stop. The core power decreases rapidly by the control rod insertion. The PRHRAS is generated by a low feedwater flow rate and the RCS is cooled down by the natural circulation of the PRHRS (Chun et al., 2014).

Transient behaviors of the RCS pressure and radial average fuel enthalpy following an REA are presented in Fig. 7. As shown in this Figure, the maximum RCS pressure is well within the ASME Service Level C limit, and the radial average fuel enthalpy is well below the acceptance criterion specified in section 15.4.8 of SRG, respectively. Also, the conservatively calculated offsite doses caused by the REA are compliant with the acceptance criteria specified in section 15.4.8 of SRP.

3.2.6. Small break LOCA

A small break LOCA is a limiting accident for a decrease in RCS inventory. This accident is caused by the postulated break of pipes that are connected to the primary system of an integral reactor vessel. In SMART100, large break LOCA is inherently eliminated and only a small break LOCA of less than 2 in. of inner diameter is possible. The major pipe's penetration nozzles such as safety injection nozzles and CVCS nozzles are located at the same elevation as the RCP. These high penetration locations increase the amount of coolant left in the vessel after an SBLOCA (Chun et al., 2014). The inner diameters of all nozzles are less than 2 in.. The main parameters of concern related with safety for this accident are the reactor pressure vessel water level, maximum fuel cladding surface temperature, core coolable geometry, and offsite doses.

Upon the break, the RCS mass and energy are released into the containment area, which causes rapid decreases in the RCS inventory and pressure. As the RCS pressure decreases, reactor trip occurs by a low PZR pressure signal. The low PZR pressure signal generates a CMTAS and the CMT isolation valves at the safety injection line are opened. The highly borated water in the CMT is injected into the annulus in the reactor pressure vessel by gravity. Simultaneously with the reactor trip, the RCPs begin to coast down and the feedwater pumps stop running because it is assumed that the turbine trip and the LOOP occur simultaneously. When the PRHRAS is generated by the low feedwater flow, the PRHRS is connected to the secondary system and removes the core residual heat. As the RCS pressure decreases continuously, the SITAS is

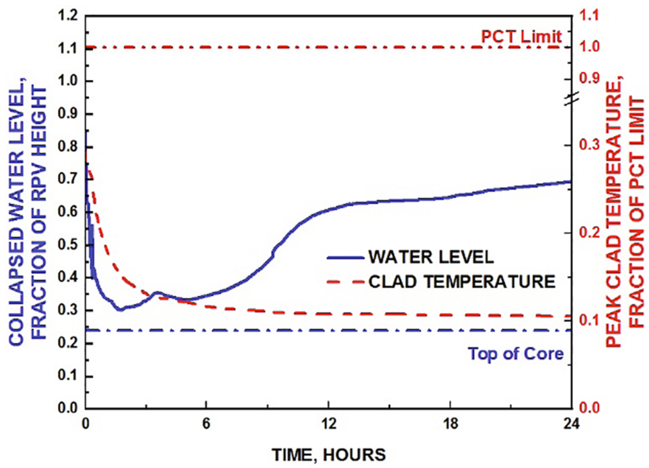

Fig. 8. Normalized reactor vessel water level and peak cladding temperature for SBLOCA.

generated by the low-low PZR pressure and the SIT isolation valves at the pressure balance line are opened (Chun et al., 2014) and the highly borated water in the SIT is injected into the annulus in the reactor pressure vessel by gravity. The RCS inventory loss is compensated by the passive safety injection flow from the CMTs and SITs.

Transient behaviors of the collapsed water level of core support barrel and hot spot fuel cladding surface temperature following a double ended guillotine break of two-inch safety injection line which is the limiting LOCA in SMART100 is presented in Fig. 8. When the safety injection flow exceeds the break flow, the RCS inventory is recovered. The minimum collapsed water level of the core support barrel during the accident is well above the top of core. Therefore, the core is filled with water for the entire period of time after the accident, and the core residual heat is continuously removed by the PRHRS and the low temperature safety injection water. During the entire transient, the fuel cladding surface temperature is continuously decreased, and it is expected that the fuel cladding oxidation and hydrogen generation rates are negligible. The rupture of fuel cladding does not occur, so the coolable geometry of the core can be maintained sufficiently. Therefore, the acceptance criteria for the emergency core cooling system performance are met. Also, the offsite doses caused by the SBLOCA are compliant with the acceptance criteria specified in section 15.0.1 of SRP.

4. Conclusions

The conservative deterministic safety analysis results for the SMART100′s safety related design basis events show that:

For AOOs, the maximum RCS and secondary system pressures do not exceed the acceptance criteria (110% of the design pressure), ensuring that the integrities of those systems are maintained. Also, the MDNBR remains above the DNBR SAFDL value, ensuring that the fuel cladding integrity is maintained.

For PAs, the maximum pressures of the RCS and secondary system are well below the acceptance criteria (110% of the design pressure), ensuring that the integrities of those systems are maintained. For REA, the peak RCS pressure is well within the ASME Service Level C limit and the maximum radial average fuel enthalpy is lower than the acceptance criterion of 230 cal/g, respectively. For SBLOCA, no core uncovery occurred throughout the transient, which results in no fuel heat-up. Thus, the cladding oxidation and hydrogen generation percentages are negligible, and the coolable core geometry is maintained. Also, the offsite doses caused by the PAs are compliant with the acceptance criteria specified in the SRP.

From the above conservative safety analysis results, the passive safety systems of SMART100 are evaluated to adequately mitigate the consequences of all SRDBEs and maintain the plant in a safe shutdown condition without any AC power or operator action for at least 72 h.

CRediT authorship contribution statement

Kyoo Hwan Bae: Conceptualization, Writing - original draft, Writing - review & editing. **See Darl Kim:** Methodology, Formal analysis. **YongJae Lee:** Methodology, Formal analysis. **Guy-Hyung Lee:** Methodology, Formal analysis. **SangJun Ahn:** Software. **Sung Won Lim:** Methodology, Formal analysis, Supervision. **Young-In Kim:** Project administration.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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References

ANS, 1971, Decay Energy Release Rates Following Shutdown of Uranium-fueled Thermal Reactors, ANS-5, ANS, U.S.A.
Bae, K.H., et al., 2001. Safety evaluation of the inherent and passive safety features of the smart design. Ann. Nucl. Energy 28 (4) (2013), 333–349.
Bae, K.H., et al., 2007. SBLOCA long term cooling procedure for the integral type PWR. Ann. Nucl. Energy 34 (2007), 333–338.
Chun, J.-H., et al., 2014. Safety evaluation of small-break LOCA with various locations and sizes for SMART adopting fully passive safety system using MARS code. Nucl. Eng. Des. 277 (2014), 138–145.
Chung, Y.-J., et al., 2003. Thermal hydraulic analysis of SMART for heat removal transients by a secondary system. Nucl. Eng. Des. 225 (2003), 257–270.
Chung, Y.-J., et al., 2012. Development and assessment of system analysis code, TASS/SMR for integral reactor, SMART. Nucl. Eng. Des. 244 (2012), 52–60.
Chung, Y.-J., et al., 2015a. TASS/SMR code improvement for small break LOCA applicability at an integral type reactor. SMART, Nucl. Eng. Design 295 (2015), 221–227.
Chung, Y.J., et al., 2015b. Strength assessment of SMART design against anticipated transient without scram. Prog. Nucl. Energy 85 (2015), 617–623.
Chung, Y.-J., et al., 2016. Assessment of TASS/SMR code for a loss of coolant flow transient using results of integral type test facility. Ann. Nucl. Energy 92 (2016), 1–7.
KAERI, CARE,, K.A., 2018. SMART Units 1& 2 PSAR. Daejeon, Korea.
KEPCO, KAERI, 2010. SMART Standard Safety Analysis Report. Daejeon, Korea.
KHNP, Kaeri, CARE, K.A., 2019. SMART100 Standard Safety Analysis Report. Daejeon, Korea.
Kim, H.-K., et al., 2013. Thermal-hydraulic analysis of SMART steam generator tube rupture using TASS/SMR-S code. Ann. Nucl. Energy 55 (2013), 331–340.
Kim, K.K., et al., 2014. The first licensed advanced integral reactor. J. Energy Power Eng. 8 (2014), 94–102.
Kim, H.R., 2017. TASS/SMR-S Code Technical Report. KAERI, Daejeon, Korea.
KINS, 2014b. Safety Review Guidelines for Light Water Reactors. Korea Institute of Nuclear Safety, KINS, Korea.
KINS, 2014, Safety Review Guidelines for Light Water Reactors, KINS/GE-N001, Rev. 4, Daejeon, Korea.
USNRC, 1974, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR 50.46, USNRC, U.S.A.
USNRC, 1978, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition), Regulatory Guide 1.70, Rev. 3, USNRC, U.S.A.
USNRC, 1994, SECY-94-084, Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, USNRC, U.S.A.
USNRC, 2007, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, USNRC, U.S.A.