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A Safety Analysis of a Steam Generator Module Pipe Break for the SMART-P

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Abstract : SMART-P is a promising advanced small and medium category nuclear power reactor. It is an integral type reactor with a sensible mixture of new innovative design features and proven technologies aimed at achieving a highly enhanced safety and improved economics. The enhancement of the safety and reliability is realized by incorporating inherent safety improving features and reliable passive safety systems. The improvement in the economics is achieved through a system simplification, and component modularization. Preliminary safety analyses on selected limiting accidents confirm that the inherent safety improving design characteristics and the safety system of SMART-P ensure the reactor's safety. SMART-P is an advanced integral pressurized water reactor. The purpose of this study is for the safety analysis of the steam generator module pipe break for the SMART-P. The integrity of the fuel rod is the major criteria of this analysis. As a result of this analysis, the safety of the RCS and the secondary system is guaranteed against the module pipe break of a steam generator of the SMART-P.

Key words: SMART-P, integral reactor, safety analysis, module pipe break

1. Introduction

Various advanced types of small and medium reactors are currently under development worldwide, and some of them are ready for construction. One beneficial advantage of a small and medium reactor is the easy implementation of advanced design concepts and technologies. Drastic safety enhancement can be achieved by adopting inherent safety characteristics and passive safety features. Economic improvement is pursued through a system simplification, modularization, and a reduction in the construction time.

SMART-P, a small sized integral type pressurized water reactor with the rated thermal power of 65.5 MWt is one of those advanced types of small and medium reactors. Design features contributing to the safety enhancement are basically the inherent safety improving features and passive safety features. The steam generator is one of the major reactor components in the SMART-P. The heat that is generated in the RCS is transferred to the secondary system in the steam generator.

The definition of a steam generator module pipe break in SMART-P means one module pipe break of a steam generator in the reactor vessel. This accident, a steam generator module pipe break, is one of the most severe cases from radioactive point of view in the SMART-P. From an analysis of this accident we can obtain a confidence in the safety of the SMART-P design.

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2. A brief Description About the SMART-P

2.1 Inherent Safety Characteristics

SMART-P contains all the major components in a single pressurized vessel. The reactor assembly of SMART-P contains major primary components such as a core, twelve SGs, a PZR, two MCPs, and twelve CEDMs in a single PRV. There is a SMART-P configuration in Fig. 1. The integral arrangement of the RCS removes the large size pipe connections between the major components, and thus fundamentally eliminates the possibility of LBLOCAs. This feature becomes a contributing factor for the safety enhancement of the SMART-P.

Canned motor MCPs remove the need for a MCP seal, and thus basically eliminate a potential SBLOCA associated with a seal failure. The reactor coolant forced by MCPs installed vertically at the top of the PRV flows upward through the core, and enters into the shell side of the SG from the top of the SG (Fig. 1).

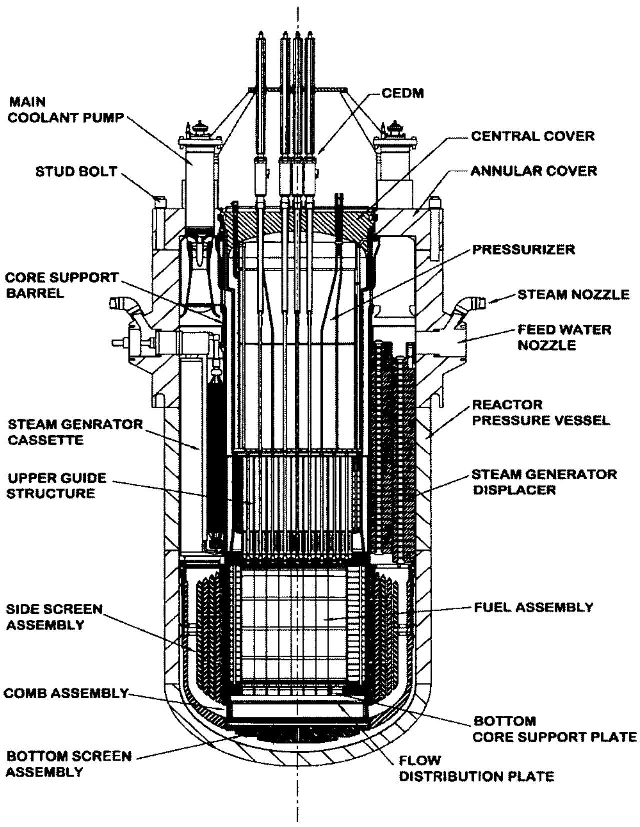

Fig. 1. SMART-P configuration.

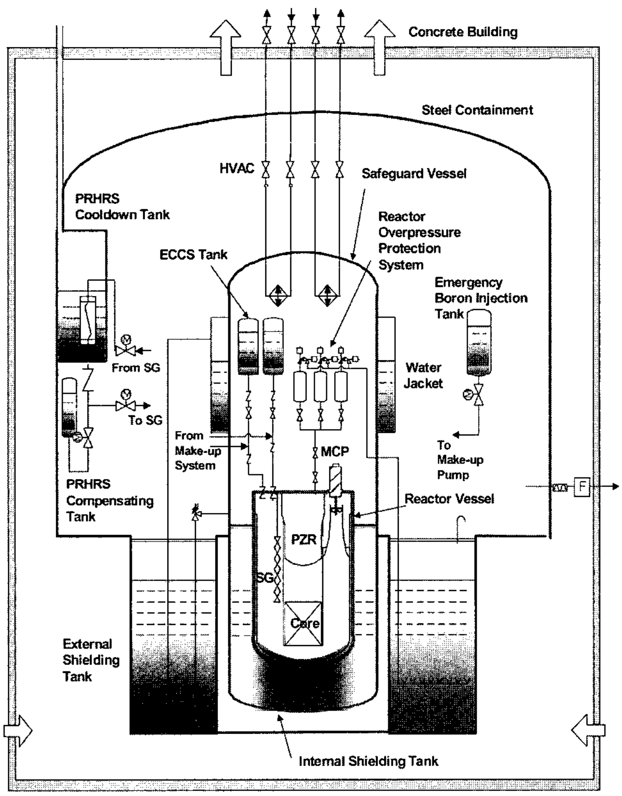

Fig. 2. Schematic diagram of the SMART-P engineered safety features.

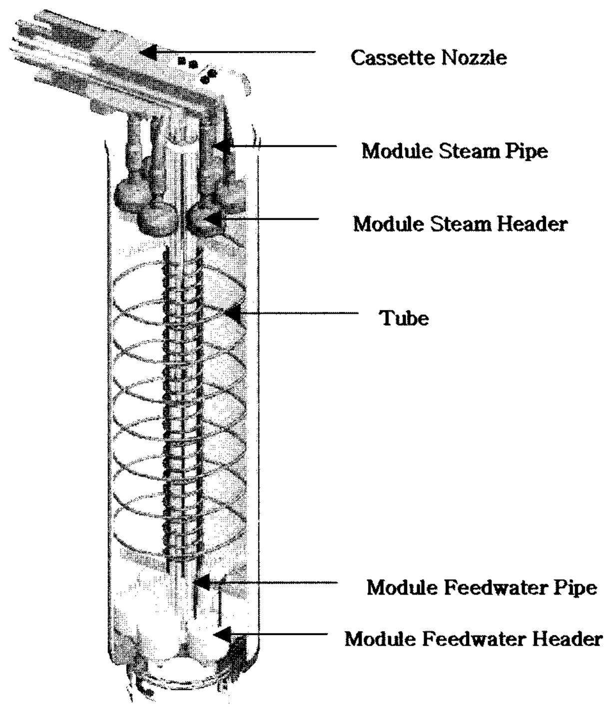

Fig. 3. Steam Generator.

Twelve SGs are located at the circumferential periphery with an equal spacing inside the PRV. The modular type once-through SGs are located relatively high above the core to provide a driving force for the natural circulation flow. The modular type once-through SG has an innovative design feature with helically coiled tubes to produce superheated steam at normal operating conditions. This design feature along with a low flow resistance enables the system to have the capability of a natural circulation operation with the maximum power of 25%. The secondary side feedwater comes into the helically coiled tube side from the bottom of the SGs, and flows upwards to remove the heat from the shell side and then exits the SG in a superheated steam condition (Fig. 3).

The in-vessel self-controlling pressurizer is located above the reactor water level. The system pressure is passively adjusted by a partial pressure of the steam and nitrogen gas in the pressurizer in accordance with a variation in the pressure and temperature of the primary coolant. In this way, the reactor always operates at its own operating pressures matched with the system's condition. A large volume of passive PZR can accommodate a wide range of pressure transients during system transients and accidents.

A soluble boron-free operation is an evolving design characteristic of the SMART-P core along with the low core power density design. The soluble boron-free operation greatly contributes to the simplification of the associated auxiliary systems, and to the reduction of the liquid waste production. The reactivity change due to the fuel burn-up is compensated for very fine-step maneuvering of the CEDM. A substantially large negative MTC due to the soluble boron-free core, offers benefits to an inherent power stability and resistance to transients. Further, advanced man-machine interface systems using digital techniques and equipment reduces the human error factors.

2.2 Passive Safety System

Besides the inherent safety characteristics of SMART-P, a further enhanced safety is accomplished by highly reliable engineered safety systems. The engineered safety systems which are designed to function passively on demand consist of a reactor shutdown system, PRHRS, ECCS, safeguard vessel, ROPS, and COPS. Fig. 2 shows the schematic diagram of the safety system.

The shutdown of SMART-P can be achieved by a function of one of two independent systems. The primary shutdown system is the control rods containing an Ag-In-Cd absorbing material. The shutdown signal de-energizes the CEDM and then the control rods drop into the reactor core by gravity and immediately stop the neutron chain. In the case of a failure of the primary shutdown system, the emergency boron injection system is provided as an active backup by a make-up pump. One train of the make-up system is sufficient to bring the reactor to a sub-critical condition.

The PRHRS removes the core decay heat and sensible heat by a natural circulation in the case of an emergency such as the unavailability of a feedwater supply or station blackout. Also, the PRHRS may be used in the case of a long-term cooling for repair or refueling. The PRHRS consists of 4 independent trains with a 50% capacity each. Two trains are sufficient enough to remove the decay heat.

Each train is connected to the feedwater and steam line pipes and consists of a heat exchanger submerged in an ECT, a compensating tank, check valves and isolation valves. The ECT is located high enough above the reactor vessel to remove the decay heat by a natural circulation when the secondary system loses its heat removal capability. The compensating tank makes up the initial inventory loss in the PRHRS. The heat exchanger removes the core decay heat and the sensible heat of the RCS, which are transferred to the PRHRS through the steam generators after a reactor shutdown. The heat exchanger is a straight tubular type with the secondary system water in the tube-inside and the ECT water in the tube-outside. The water in the ECT is heated, boiled and eventually evaporated into the atmosphere. The water inventory in the ECT can remove the decay heat for 72 hours without any operator action.

Under emergency cooldown conditions, the main feedwater and steam isolation valves are closed, and the PRHRS valves connecting the heat exchanger are opened in response to the signal generated by the control system. In this case, the steam that enters the heat exchanger from the stream generator cassette is condensed, and the condensed water returns to the steam generator by gravity.

The ECCS provides a coolant to the PRV during the early stage of a SBLOCA. The ECCS consists of two independent trains with a 100% capacity each. Each train includes a cylindrical water tank pressurized with nitrogen gas, isolation and check valves, and a pipe connected to the PRV.

When the RCS depressurizes below the pressure setpoint, the coolant in the emergency core cooling tank is injected into the upper annular cavity by a gas pressure. Also, the active makeup pump actuates when the upper annular cavity water level reaches a low level setpoint, which compensates for the loss of a reactor coolant inventory and keeps the coolant level well above the top of the core for a long term cooling period.

The safeguard vessel is a leak-tight pressure retaining steel-made vessel intended for the accommodation of all the RCS including the reactor assembly, the gas cylinders, and the associated valves and pipes. The primary function of the safeguard vessel is to confine the radioactive products within the vessel and thus to protect any reactor coolant leakage to the containment. The safeguard vessel also has a function of limiting the break flow following a SBLOCA by a pressure balance with the PRV and keeps the reactor core from being damaged with the operation of the PRHRS and ECCS. In the case of the postulated design basis accidents, the pressure in the safeguard vessel is reduced by a heat removal through the metal structure itself and the water jackets installed around the vessel, together with the operation of the air conditioning system.

3. The SG Features

SMART-P has twelve identical SG cassettes that are located on the annulus formed by the PRV and the core support barrel. Each SG cassette is of an once-through design with a number of helically coiled tubes. The primary reactor coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flow is evaporated in the tube and exits the SG cassette nozzle header at a 40°C superheated steam condition of 3.5

MPa (Fig. 3). In the case of a normal shutdown of the reactor, the SG is used as the heat exchanger for the PRHRS, which permits the independent operation of the PRHRS from the hydraulic condition of the RCS. The SG cassette consists of six modules in which 96 tubes are connected equivalently. In the case of a tube leak, each module can be plugged at up to 10% of the total heat transfer area. The nozzle feedwater header and steam header of the cassette are designed to be a single structure located on the lateral surface of the PRV to reduce the vessel penetrations.

The SG cassette tube bundle is manufactured by coiling it onto the central mandrel with subsequent tubes coiled over the previous ones. After coiling, the tube bundle is subjected to a heat treatment to eliminate the residual stresses. The design temperature and pressure of the SG cassette are 310°C and 14.7 MPa, respectively.

4. Analysis Method and Results

The definition of a SG module pipe break in SMART-P means one module pipe break of a steam generator in the reactor vessel. The penetration parts of the reactor vessel are the subsection pipes of the feed lines and the main steam lines. They consist of 12 pipes respectively and one pipe has 6 module pipes with a 13 mm diameter. These 6 module pipes have 96 helical tubes in a steam generator [1, 2]. The module pipes in a reactor vessel act as a protective barrier for any radioactivity propagation from the primary to the secondary system. If a SG module pipe break occurs, there is a complex thermal hydraulic effect as well as a leakage of the break flow to the secondary system.

4.1. The General Behavior of a SG Module Pipe Break Accident

One module pipe break at a steam generator is the initiating event. The fluid in the secondary system is mixed with that of the primary system which includes a radioactivity. Mixed fluid is sent to the turbine successively until a ceasing of the turbine. This radioactivity can be released to the environment by the air ejector at a condenser after sending it to the condenser. The air ejector is used for the release of the non-condensable gas to the air. Actually before long the reactor trip signal will be set by the radioactivity detectors on the steam lines. This signal indicates a high level leakage of radioactivity from the secondary system. For a conservation of the result, a reactor trip signal by a low PZR pressure is used after skipping the signal of a high level radioactivity in the secondary system in the beginning. The pressure of the RCS is decreased according to the leakage of the coolant to the secondary system continuously. The system is tripped by a low PZR pressure signal. After signaling a reactor trip the steam generators are isolated by the feedwater and main steam isolation valves. And the steam generators are connected to the PRHRS. The PRHRS removes the decay heat by a natural circulation.

In a view of the break flow, a smaller break area creates a larger integrated steam amount through the main steam lines. Because the small break area brings about a delay of the reactor trip time. From this view the maximum leakage of the SG module pipe break is similar or less conservative to the result of a SGTR [3, 4].

4.2 The Analysis method and Assumption

The assessment of a primary to secondary break in the LCO is performed using the TASS/SMR code. Basically TASS/SMR can model SMART-P with nodes and paths. The code can calculate the core power, pressure, flow, temperature and other values of the primary and secondary system for the various initiating conditions [5,6]. ABAQUS code is used for the calculation of the fuel temperature. One stuck CEDM having the largest reactivity and LOOP are assumed. As a result of a LOOP the MCP does not supply coolant to the RCS after the reactor trip. SG module pipe break accident is classified as one of the limiting condition accidents in the SRDBE for the SMART-P. The limiting criteria are;

The minimum CHFR must be greater than 1.3.

The primary system pressure must be less than 110% of the design value

The total radioactivity from an accident must be less than the permissible amount [7].

The ANS-73 curve is used for the calculation of the decay heat [8].

4.3 Results

The sensitivity study for the various initial conditions with the double-ended guillotine break is performed. The minimum CHFR is the criteria. As a result, the determined initial conditions are a low primary flow, high core power, high at the core entrance. The least negative reactivity coefficient and most negative moderator coefficient create more severe results. The reactor trip signal set by the low PZR pressure is 65 seconds after the initiating event.

According to the over-cooling due to a mismatching of the heat generation in the core and the heat removal in the SG, the coolant temperature at the SG inlet decreases from the beginning of the transient as shown

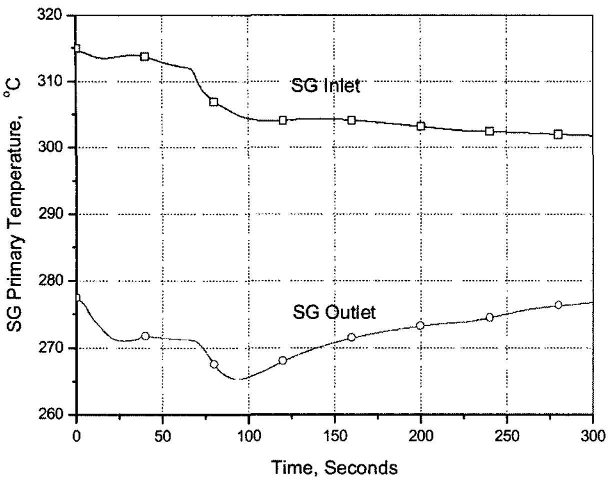

Fig. 4. The primary coolant temperature.

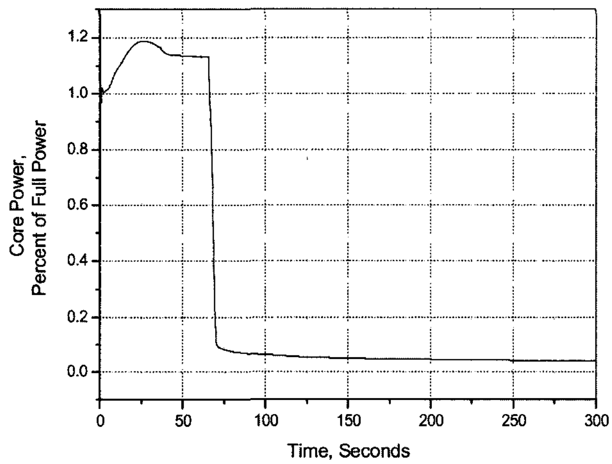

Fig. 5. The behavior of the core power.

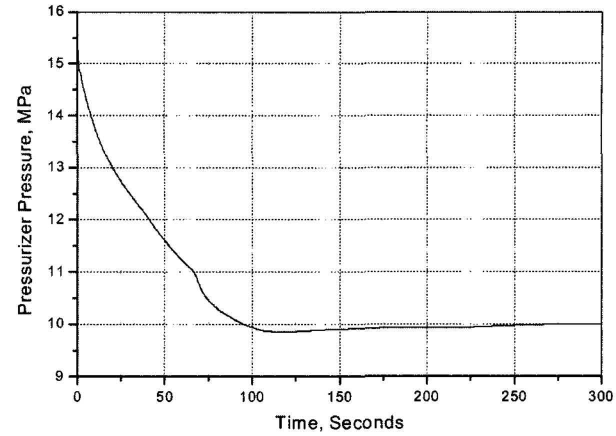

Fig. 6. The PZR pressure.

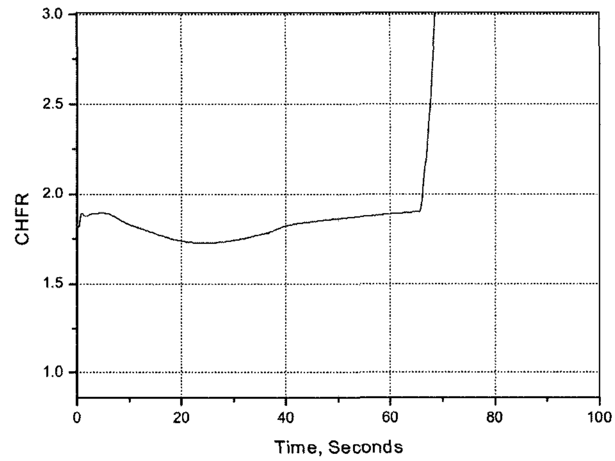

Fig. 7. The behavior of the CHFR.

in Fig. 4. Thus, the core power is increased by the negative reactivity characteristics of the coolant. The core power has a maximum value at 26 seconds of 119% (Fig. 5).

The RCS pressure is decreased continuously by a leakage of the coolant to the secondary system during the transient. The initial pressure, 15.52 MPa, is the maximum RCS pressure. The RCS pressure is stabilized by the PRHRS after the reactor trip (Fig. 6). The secondary pressure at the broken part of SG is increased by the incoming flow from the RCS at an initial time. By the initiation of the PRHRS the secondary pressure maintains a balance with the pressure of the RCS. The passive safety systems play important roles in mitigating the consequences of the events and ensure the SMART-P safety requirements. The passive residual heat removal system is an ultimate PRHRS for the design bases events. Fig. 7 shows the change of CHFR. The minimum CHFR is 1.73 at 24 seconds.

One of the safety goals, which can maintain a 72 hours grace period without any operator's action, should be achieved by the PRHRS capability. The reactor power ultimately decreased to the decay heat level as the PRHRS is operated. Due to a small water inventory in the helically coiled once-through SG and the very high negative shutdown rod worth of the SMART-P, the cool down of the RCS by the increased heat removal events is limited. There is no possibility of a return-to criticality in the core after the reactor trip as shown in Fig. 5. Therefore, the reactor can be maintained in a sub-critical state whatever happens in the secondary side.

5. Conclusion

In SMART-P, a system simplification and a reduction of pipes and valves are possible due to the implementation of advanced passive systems and of highly inherent safety characteristics. Modularization, component standardization, and on-shop fabrication and a direct site installation of components are additional characteristics that can contribute to the reduction of the construction cost.

In the case of a SG Module Pipe Break accident with LOOP for the SMART-P, the minimum CHFR is maintained at over 1.3 and the hottest fuel rod temperature is below 606°C during the transient. It means that the integrity of the fuel rod is guaranteed. The RCS and the secondary system pressures are maintained below 18.7 MPa, which is the system's design pressure. The natural circulation in the RCS and the PRHRS is well established during the transients and is enough to ensure a stable plant shutdown state after the reactor trip. Also, it was observed that the safety features of the SMART-P design carried out their functions well. A radioactivity analysis will be performed with an integrated break flow later.

Acknowledgement

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Nomenclature

CEDM	: Control Element Drive Mechanism
CHFR	: Critical Heat Flux Ratio
COPS	: Containment Overpressure Protection System
ECCS	: Emergency Core Cooling System
ECT	: Emergency Cooldown Tank
HVAC	: Heating, Ventilating and Air Conditioning system
KAERI	: Korea Atomic Energy Research Institute
LBLOCA	: Large Break Loss of Coolant Accident
LCO	: Limiting Conditions for Operation
LOOP	: Loss Of Offsite Power
MCP	: Main Coolant Pump
MDC	: Moderator Density Coefficient
MPa	: Mega Pascal
MTC	: Moderator Temperature Coefficient
MWt	: Mega Watt thermal
PRHRS	: Passive Residual Heat Removal System
PRV	: Pressurized Reactor Vessel
PZR	: Pressurizer
RCS	: Reactot Coolant System
ROPS	: Reactor Overpressure Protection System
SBLOCA	: Small Break Loss of Coolant Accident
SG	: Steam Generator
SGTR	: Steam Generator Tube Rupture
SMART-P	: System-integrated Modular Advanced Reac-Tor for a Pilot
SRDBE	: Safety Related Design Based Events
TASS/SMR	: Transient And Setpoint Simulation/Small and Medium Reactor

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