Proceedings of the Korean nuclear Autumn Meeting
Yongpyong, Korea, October 2002

Preliminary Safety Analysis on Transient of Heat Removal Increase by Secondary System for SMART

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Abstract

SMART was an integral type reactor of 330 MWt, which enhanced safety by adopting inherent safety design features. Thermal hydraulic characteristics on transient of heat removal increase by the secondary system for the SMART have been carried out by means of TASS/SMR code. The primary, secondary, and residual heat removal systems of the SMART were modeled properly. Then, a set of transients for the whole system was investigated. The results of the analyses using the conservative initial and boundary conditions showed that the safety features of the SMART design well carried out their functions and large moderator temperature coefficient due to the soluble boron free reactor affected on the transient behavior.

1. Introduction

Small and medium sized integral type reactors for the diverse utilization of a nuclear energy are getting much attention from the international nuclear community (Kupitz, 1997). Large capacity power reactors are not economically viable for non-electric applications. However, small and medium sized nuclear cogeneration reactors diversify the peaceful uses of nuclear energy in the areas of seawater desalination, district heating, heat-generation process and ship propulsion. Those reactors have been studied to develop an economically viable and safer advanced reactor. For examples, SMART (Chang et al., 2002, 1998), IRIS (Carelli et al., 2002), KLT-40 (Humphries and Davis, 1998) and MRX (Yamaji et al., 1995).

The basic design development of the SMART and its application system for desalination was completed in March, 2002 (Chang et al. 2002). The SMART plant aims to generate about 90 MWe of electricity using 90% of the total energy supplied and produce 40,000 m^3/day of potable water using the remaining energy. The SMART is a small modular pressurized water reactor, which is designed to reach a high safety level and passive safety features. Different from the loop type commercial reactors, the main characteristic of the SMART is the integral reactor concept containing all the major reactor coolant system components (for examples, canned motor pumps, once-through steam generator, pressurizer) located in the reactor vessel. Also, the design characteristics are a self-controlled pressurizer with non-condensable gas, and a large negative moderator temperature coefficient result from the soluble boron free core.

The SMART safety is based on inherent safety design features to eliminate the possibility of accident sequences from occurring, and to reduce the severity of their consequences such as natural circulation operation, large water inventory and double containment. The main safety objective of the SMART is to increase the degree of inherent safety features by eliminating or mitigating the initiating events and accidents via the advanced design. Examples of the strategy are: i) the adoption of an integral reactor to eliminate vessel penetration of a large sized pipe and thus elimination of the possibility of a large break loss-of-coolant-accident, ii) the adoption of helically coiled once-through steam generator with the primary fluid flowing on the outside of the tube bundle to mitigate the possibility of the multiple steam generator tube rupture accident, iii) the reactor design with a high degree of natural circulation up to 25% of nominal power, and iv) the adoption of double containments, which are called a containment and a safeguard vessel, to mitigate the consequences of small break loss-of-coolant-accidents.

In addition to these inherent safety features, the safety goals are enhanced through highly passive safety systems such as a passive residual heat removal system (RHRS), passive emergency core cooling system (ECCS), safeguard vessel, and over pressure protection systems.

2. Safety systems of the SMART

Fig. 1 shows a schematic diagram of the SMART safety systems. The SMART safety system along with the multiple safety barrier design such as fuel, reactor vessel, safeguard vessel, and containment prevents a core damage and minimize a radiation release to the environment during events and accidents.

The primary safety strategy in the SMART depends on two main items. The safeguard vessel, which is a steel-made leak-tight pressure vessel housing reactor vessel as shown in Fig. 1, is adopted to limit the blow down following a small break loss-of-coolant-accident, and to confine any radioactive release from the primary circuit under design bases events and accidents related to the loss of integrity of the primary system. Secondly, the residual heat removal system passively removes a core decay heat by a natural circulation for emergency situations when normal feed water supply and steam extraction are unavailable.

The residual heat removal system is designed with four independent identical trains and the operation of any two trains will be sufficient to remove the decay heat generated in the core. Each train is connected to the feed and steam lines and consists of a heat exchanger submerged in an emergency cooldown tank (ECT), a compensating tank, a check valve and isolation valves. The emergency cooldown tank is located high enough above the reactor vessel to remove the decay heat by natural circulation when the secondary system loses the capability for heat removal. The compensating tank, which is pressurized with N_2 gas, makes up the initial inventory loss in the residual heat removal system and steam generator regions. The check valve, installed on the pipe between the compensating tank and the heat exchanger, is used to protect the reverse flow to the heat exchanger.

The SMART has twelve modular helically coiled steam generator cassettes, paired together to four steam and feed lines. A passive residual heat removal system is connected to each steam and feed lines to guarantee redundancy. Following an event or accident, the main feed water and steam isolation valves (MFIV/MSIVs) are closed with a reactor trip signal, and the isolation valves on the residual heat removal system pipeli`ne are opened. Each feed water and steam line has redundant isolation valves in order to minimize the malfunction of feed/steam water isolation valves following steam generator tube rupture accident. A natural circulation loop is established and a decay heat can be removed from the reactor core. This system is simple from a design, however it still requires careful study and examinations from thermal hydraulic perspective such as a verification of the correct performance of the system under various accident conditions.

Since the SMART design inherently eliminates the large break loss-of-coolant-accident by elimination of a large sized pipe penetration to the reactor vessel, the emergency core cooling system is provided to protect the core damage during a small break loss-of-coolant-accident. The emergency core cooling system consists of independent two trains and the operation of one train is sufficient to provide its function. Each train includes a cylindrical water tank pressurized with N_2 gas, isolation valves and a check valve. When the primary system depressurizes below the emergency core cooling system tank pressure, the coolant in the tank is injected into the upper annular cavity of the pressurizer. The emergency core cooling system piping of each train shares the reactor vessel penetration with the make up system as well as boron injection system as shown in Fig. 1.

The safety features are completed by the definition of a limited number of passive safety systems to achieve a high degree of safety and a low core damage frequency (CDF), which is estimated about 8.5×10^{-7}. it must be pointed out that the deterministic analysis such as those discussed in this paper are iterated with a probabilistic safety analysis (PSA) carried out in parallel. Significant information coming from the preliminary PSA already led to modification of the systems. For example, diversity for isolation valves, feed and bleed function, and application of a containment spray system for the lower pressure makeup system.

3. Thermo-hydraulic analysis on transient of heat removal increase by the secondary system

Passive safety systems such as residual heat removal system, emergency core cooling system and reactor over-pressure protection system, play important roles to mitigate the consequences of event/accident and thus ensure the SMART safety requirement. The residual heat removal system is an ultimate passive decay heat removal system for design bases event/accident. The SMART safety goal of a 72 hours grace period without any operator action should be achieved by the residual heat removal system capability. The emergency core cooling system is a dedicated passive safety system to mitigate the consequences of a postulated small break loss-of-coolant-accident. The reactor overpressure protection system is used to mitigate the consequences of overpressure accidents.

3.1 Analysis model and initial/boundary conditions

A thermo hydraulic analysis for the SMART has been carried out by means of the TASS/SMR code for a full range of reactor operating conditions (Yoon, 2001). TASS/SMR code has been developed by Korea Atomic Energy Research Institute (KAERI). The system thermal-hydraulic response is modeled by a node and flow-path network. The nodes enclose control volumes which represent the fluid mass and energy. The flow-paths connecting the nodes represent the fluid momentum and have no volume. The thermal-hydraulic model is formulated with six one-dimensional conservations. The conservation variables are mixture mass with liquid and steam, liquid mass, non-condensable gas mass, mixture energy, steam energy, and mixture momentum. The mass and energy for the liquid and the steam including the gas are calculated for each node. Mass flow rate is calculated for each flow-path. The code incorporates slip effects in the flow-paths by means of empirical correlations. Nodes with homogeneous or fully mixed fluid are at equilibrium. The possible states for non-homogeneous or phase-separated nodes with separate two-phase mixture and steam regions are saturated liquid with saturated steam, sub-cooled liquid with saturated steam, saturated liquid with superheated steam, and sub-cooled liquid with superheated steam. A number of SMART specific models reflecting the SMART's design characteristics such as a helically coiled steam generator, pressurizer with non-condensable gas and heat exchanger in the residual heat removal system, have been addressed in the code. A detailed departure from nucleate boiling ratio (DNBR) analysis is performed using a MATRA code, which is a sub-channel analysis code (Yoo and Whang, 1998) and is calculated based on local coolant conditions calculated at every time step by means of results of TASS/SMR for inlet boundary conditions of the MATRA code. The SR-1 critical heat flux correlation is used for the DNBR calculation of the transient.

The plant configuration adopted for the TASS/SMR model is shown in Fig. 2. The core and the pressurizer are located at the lower and upper parts of the reactor vessel, respectively. Four canned motor pumps and twelve steam generator cassettes are symmetrically arranged along the annular region between the reactor vessel and the core support barrel. After removing the heat generated in the core, the coolant enters the suction header of the canned motor pump via the upper core region. Passing the canned motor pump, the coolant is distributed to the shell side of the steam generator cassette and transfers heat to the secondary coolant. The feed water flows upward in the helically coiled tube to remove the heat from the shell side primary coolant and exits the steam generator cassette as superheated steam.

TASS/SMR nodalization of the SMART is shown in Fig.3. To well predict the physical phenomena expected during event/accident, the system is modeled in detail. The core is modeled as one average channel having 57 fuel assemblies and one core bypass region. The fuel rods are modeled as 12 axial nodes and 5 radial nodes. Four canned motor pumps are modeled separately and the pressurizer component is divided by the upper annular cavity, intermediate cavity, and end cavity. Twelve steam generator cassettes are modeled as four sections and each section is divided to properly predict the heat transfer phenomena by 10 axial nodes. Four trains of the residual heat removal system connecting the main feed water and steam lines are modeled independently to simulate an asymmetric effect.

Conservative initial and boundary conditions as well as conservative assumptions are employed to evaluate the transient of an increase in heat removal by the secondary system of the SMART design. The major parameters used for event/accident analyses are shown in Table 1. The initial core power and feed water flow rate are assumed to be 98~102% of the nominal values considering the measurement uncertainty. The pressure of the pressurizer end cavity is chosen at a high value, 15.25 MPa, within a limiting condition for operation (LCO) and that of the steam generator outlet is 3.3 MPa. For the conservative results, the moderator density and Doppler reactivity values are selected as the least or most negative ones depending on the transient characteristics of the event/accident. The minimum shutdown rod worth with the most reactive rod assembly stuck out is assumed and a total rod drop time is 8 seconds. A conservative ANS-73 decay heat curve is used with a 1.2 multiplication factor.

3.2 Event and accident analysis results

3.2.1 Preliminary calculation

To evaluate validity, the results of the TASS/SMR are compared with those of MARS code using same initial and boundary conditions. The MARS code (Jeong et al., 1999) has been developed at KAERI by consolidating and restructuring the RELAP5/MOD3.2 (INEL, 1998) and CORBRA-TF (Thurgood et al. 1983), which has the capability of analyzing the one-dimensional and three-dimensional best estimated thermal-hydraulic system and the fuel responses of the light water reactor transients.

A transient of an increased main steam flow under nominal conditions is selected to compare the results and an initiating event for the transient is a 17% increase over the nominal full power steam flow rate. According to the coolant cooldown due to mismatching of the core power and the heat removal by the secondary side, the coolant temperature at the core inlet decreases from beginning of the transient as shown in Fig. 4. Thus, the core power is increased by the negative reactivity characteristics of the coolant. Fig. 5 shows the normalized core power. The TASS/SMR and MARS codes predict almost the same as an increase of the core power by the reactivity characteristics and a decrease to the decay heat power level by the reactor trip. Figs. 4 and 6 show the liquid temperature at the core inlet and outlet, and the pressure at the pressurizer, respectively. During the initial period of the transient by reactor trip, two codes predict the same behavior as the primary system is cooldown until the excessive heat removal. After reactor trip occurs, the MARS code predicts the pressurizer pressure higher than the TASS/SMR code. The MARS code can take an increase of steam temperature by the hot in-surge flow from the core to the pressurizer due to adoption of a thermal non-equilibrium model. On the other hand, the TASS/SMR code cannot take it because the code uses a thermal equilibrium model. In the TASS/SMR case, a steam temperature is constant at the pressurizer for the transient. As a result, the primary pressure of the TASS/SMR calculation is lower than that of the MARS and the results of the TASS/SMR is more conservative than that of the MARS in view of a departure from nucleate boiling ratio.

3.2.2 Increased main steam flow event

An inadvertent increased opening of the turbine control valve or the turbine bypass valve may cause an increase in the steam flow by the secondary system. Those may be caused by operator errors or malfunctions of the turbine system, and will result in no more than a 17% increase over the nominal full power steam flow rate. The most limiting single failure is determined to be the loss of the feed water control system with beginning of the transient. A loss of offsite power concurrent with a turbine trip following a reactor trip is considered as a basic assumption.

The opening of the turbine control valve increases the rate of heat removal by the steam generators causing cooldown of the primary system. Fig. 7 shows the core power behavior during the transient. Due to the negative moderator temperature coefficient, the core power increases from the initial value of 98% of rated core power, reaching a reactor trip value of 115%. The feed water control system supplies 117% of nominal feed water flow to steam generator cassettes such that excessive heat transfer from primary to secondary system is maintained. Following the generation of a turbine trip with the reactor trip and concurrent loss of offsite power, normal feed water flow is terminated and the flow in the residual heat removal system is initiated. Since the secondary system is isolated completely by the main feed water line isolation and steam line isolation valves, the residual heat removal system removes the heat stored in the core.

Figs. 8 and 9 show the primary coolant temperature and pressure, respectively. The primary coolant temperature and pressure continue to decrease to the point where a removed heat at the heat exchanger of the residual heat removal system is equal to the decay heat in the core. The secondary system pressure increases with closing of the main steam and feed water isolation valves and then decreases due to the cooldown caused by the flow through the residual heat removal system as shown in Fig. 10. Fig. 11 shows the heat rate removed in the residual heat removal system. Natural circulation is achieved in both the primary system and the residual heat removal system after the reactor is tripped. The extracted energy from the primary system is high at the beginning of the transient due to the large thermal difference between the extracted energy in the steam generator and the exhausted energy in the heat exchanger of the residual heat removal system. As the core power decreases the decay heat power level, a heat balance between the primary and residual heat removal systems is established and the systems maintain stable conditions. The residual heat removal system fulfills its mission, maintaining the rejected power well above the decay power.

A DNBR is one of major parameters. During the transient, the minimum DNBR occurs just after the reactor is tripped. When the turbine trip is initiated with the reactor trip, the loss of offsite power is occurred and the canned motor pumps begin to coast down. As the canned motor pump coastdown, the DNBR begin to decrease due to mismatch between the core power and the coolant flow, reaching a minimum value of 1.445 which is above a specified acceptable fuel design limit (SAFDL) and then rapidly increases as shown in Fig. 12.

3.2.3 Steam line break accident

A steam line break (SLB) accident is defined that a pipe ruptures in the main steam system, which may occur as a result of thermal stress or cracking in the steam line pipe. Degradation in fuel cladding performance may result from this accident. The steam line break accident is a limiting accident for the increase in the heat removal by the secondary system, and is performed to maximize potential for degradation in fuel performance. A break size of 0.022 m^2 in the containment is assumed for the accident analysis.

Figs 13 and 14 show the primary and secondary pressure. Following a steam line break in the presence of a strong moderator temperature coefficient, the steam generators begin to depressurize due to increased steam flow leaving the steam generators and to increase in feed water caused by the steam line break. The increase in effective heat rate causes the reactor coolant to enter the core inlet at a colder temperature as shown in Fig. 15. After a coolant passes through the steam generators, it enters the reactor vessel where some inter-section mixing occurs. A positive reactive insertion begins and core power increases at a rate related to the cooldown rate. The cooldown causes an increase in the core reactivity due to the negative moderator and Doppler reactivity coefficients, which result in a core power increase as shown in Fig. 16.

Simultaneous with the break, the core power increase rapidly, which results in a reactor trip by a high core power signal. The reactor trip causes a rapid decrease in the core power and the canned motor pumps begin to coast down. Then, the plant approaches hot shutdown conditions with the most reactive rod assembly stuck out. Consequently, the main feed water and steam isolation valves are closed and the residual heat removal system is connected to the feed water and steam lines. The cooldown of the reactor coolant continues after the reactor trip and the blowdown is terminated in all but a steam generator section connected with the broken steam pipe. The reactor power ultimately decreases to decay heat level as the residual heat removal system is operated. There is no possibility of return-to power during the accident due to a small water inventory in the helically once-through steam generator and large shutdown reactivity.

Fig. 17 shows the DNBR behavior during the initial period of the transient. The DNBR decreases slowly as the core power increase with the beginning of the transient. When the canned motor pumps start to coast down, the DNBR decreases rapidly by a mismatch between the core power and the core mass flow rate. The minimum DNBR of 1.41 is reached at 29 seconds, which is same as the specified acceptable fuel design limit for the SMART of 1.41. When the core power decreases to the decay heat level and the natural circulation in the primary and residual heat removal systems is fully established, the DNBR rises abruptly.

4. Conclusions

The inherent behavior of the SMART following an excessive steam release and a postulated steam line break are performed using the TASS/SMR and MATRA codes.

The results of the analyses using the conservative initial and boundary conditions and assumptions are as follows. The coastdown rate of the canned motor pumps is sufficient to prevent reaching the minimum departure from nucleate boiling ratio for the event and accident discussed. The natural circulation in the primary side is well established during the transient and it is enough to ensure a stable plant shutdown state with the reactor trip. The natural circulation of the residual heat removal system operates properly and is able to remove the decay heat from the reactor core. Also, it can be stated that the safety features of the SMART design well carry out their functions.

The flow in the residual heat removal system shows an oscillating behavior that need to be investigated in order to identify whether this is due to numerical instability or thermal hydraulic instability on a two-phase flow.

Acknowledgement

This study has been performed under a contract with the Korean Ministry of Science and Technology.

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Table 1 Initial and Boundary conditions

Parameter	Value
Core power, % of nominal power	98/102
Pressurizer pressure, MPa	15.25
Primary coolant flow rate, kg/s	1596.5
Core inlet liquid temperature, K	275
Steam generator outlet pressure, MPa	3.3
Feed water flow, % of nominal value	102
Feed water temperature, K	180
Shutdown rod worth, %$\Delta\rho$	-10
Shutdown rod drop time, s	8.0
Decay heat curve	ANS-73

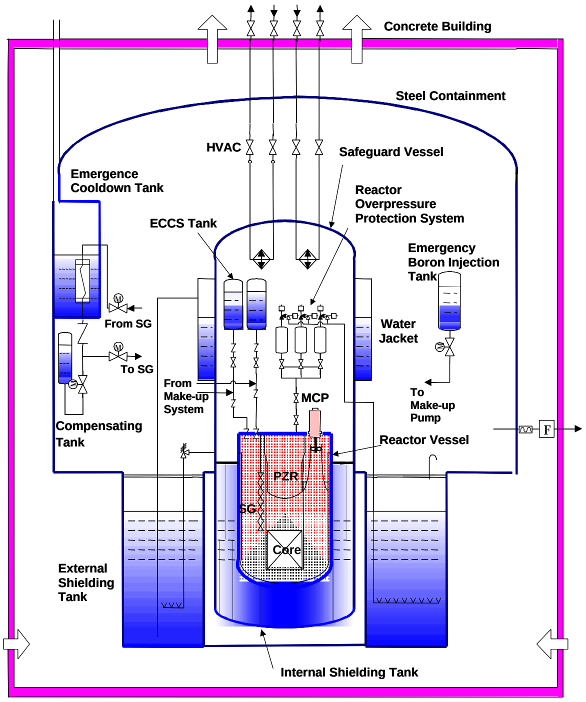


Fig. 1 Safety systems of the SMART

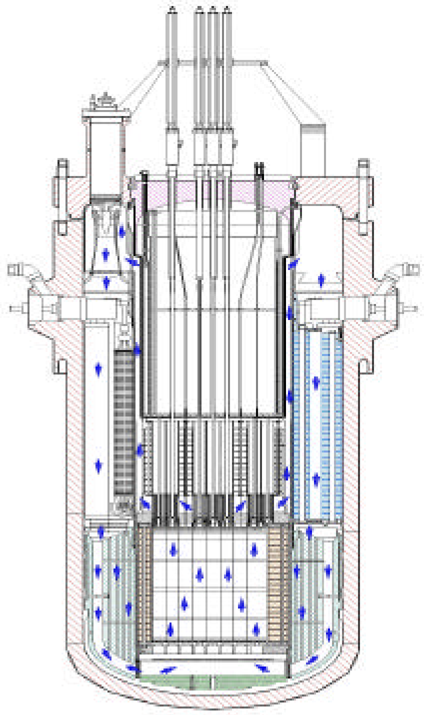


Fig. 2 Plant configuration for the SMART

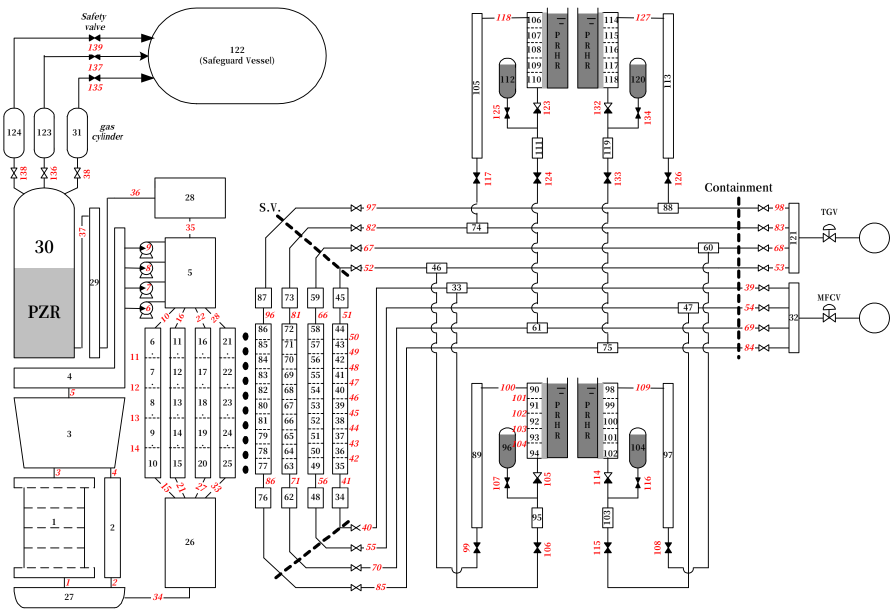


Fig. 3 TASS/SMR nodalization for the SMART

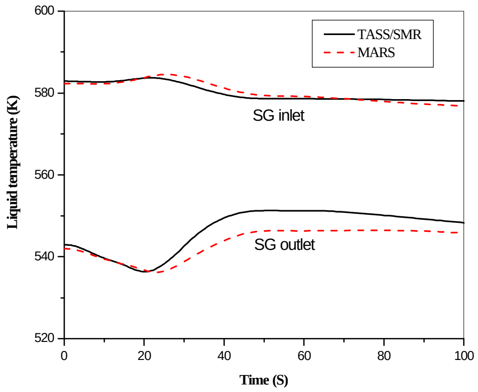


Fig. 4 Primary coolant temperature with TASS/SMR and MARS codes

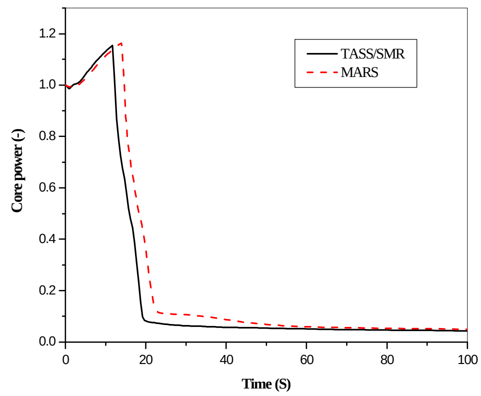


Fig. 5 Normalized core power with TASS/SMR and MARS codes

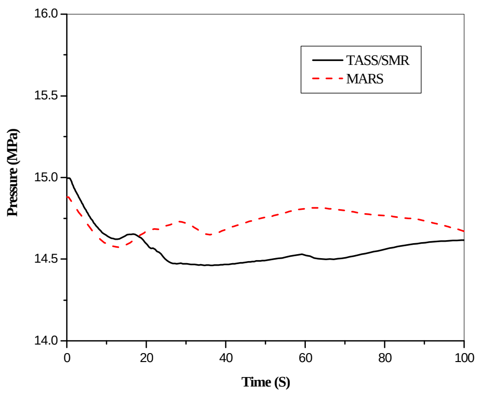


Fig. 6 Pressurizer pressure with TASS/SMR and MARS codes

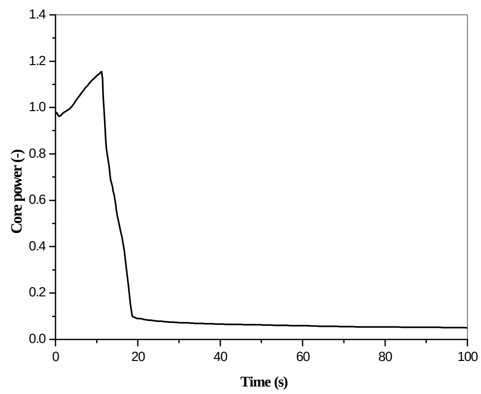


Fig. 7 Normalized core power for the increased steam flow event

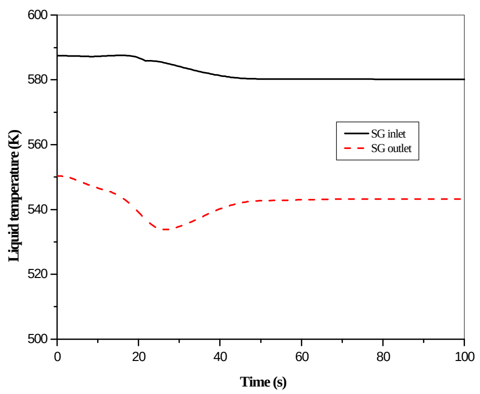


Fig. 8 Primary coolant temperature for the increased steam flow event

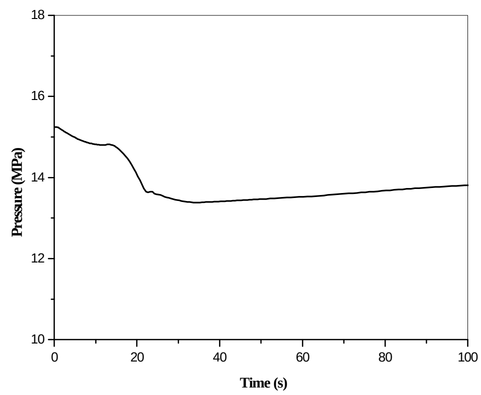


Fig. 9 Primary pressure for the increased steam flow event

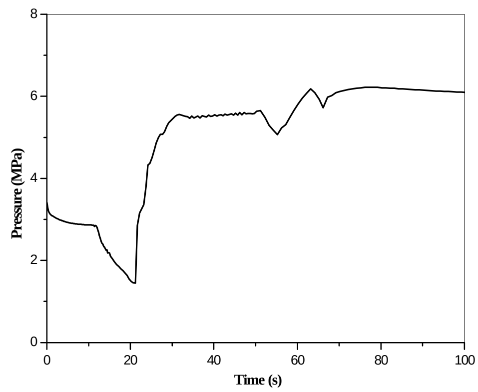


Fig. 10 Secondary pressure for the increased steam flow event

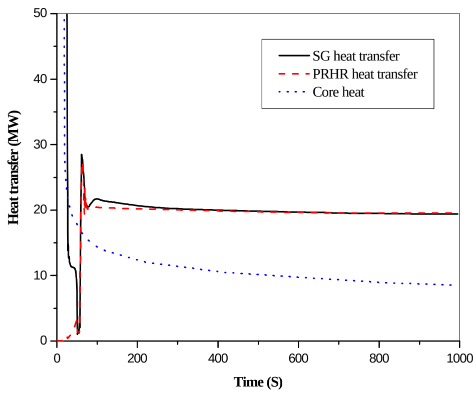


Fig. 11 Heat rate removed in RHRS for the increased steam flow event

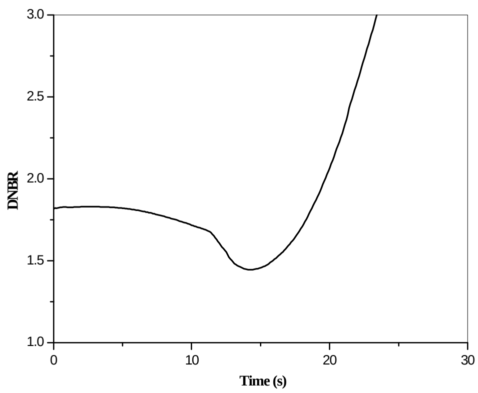


Fig. 12 DNBR for the increased steam flow event

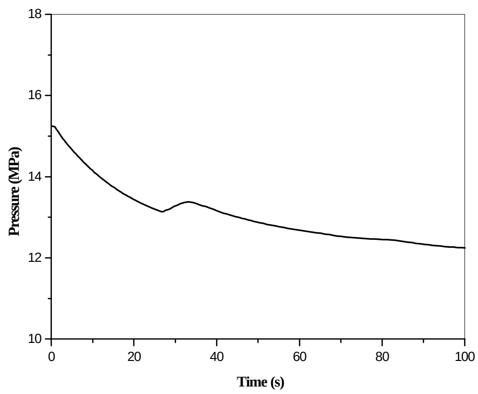


Fig. 13 Primary system pressure for the steam line break accident

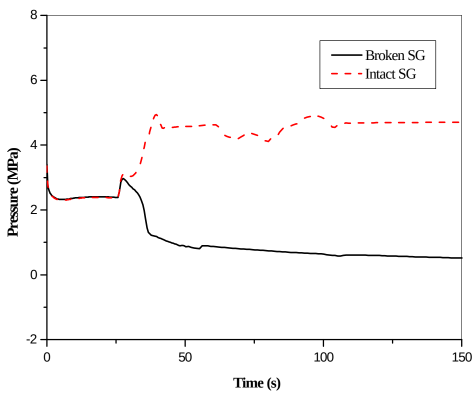


Fig. 14 Secondary system pressure for the steam line break accident

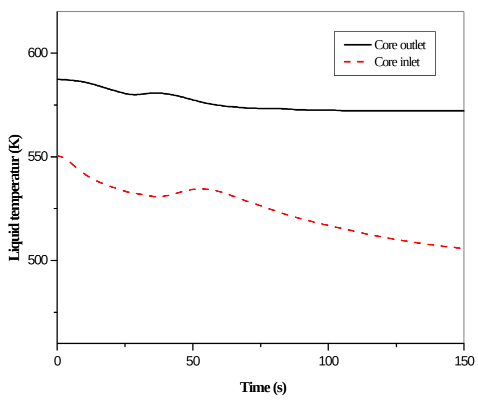
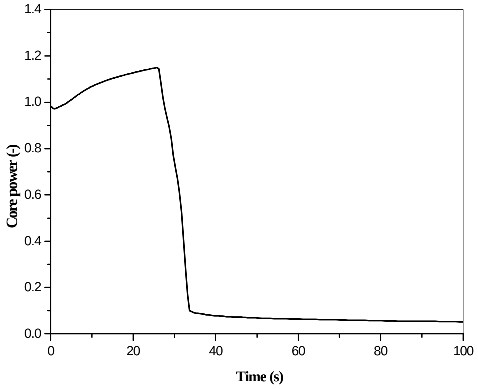


Fig. 15 Primary coolant temperature for the steam line break accident



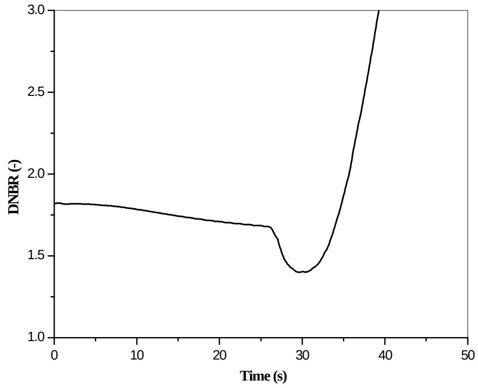


Fig. 16 Normalized core power for the steam line break accident

Fig. 17 DNBR for the steam line break accident