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This paper presents the function and performance of the safety systems and load following behavior of a Passive Safety Small Reactor for Distributed energy supply system (PSRD). The PSRD is an integrated-type PWR adopting natural circulation and self pressurization in the primary system, with the reactor thermal power between 100 and 300 MW, aimed at supplying electricity to small grids, district heating, etc. In design of the PSRD, high priority is placed on the enhancement of safety, as well as the improvement of economy, so that the reactor can be sited close to energy demand areas. The safety enhancement is ensured by drastic reduction of the possibility of loss-of-coolant accidents (LOCAs) by limiting the primary pipe penetration only for the safety valve line, and adoption of a fully passive safety system with a water-filled containment and so on. The key features of the system are described together with the analyses for load following by the RETRAN-02 and for transient of a LOCA by the RELAP5. The analysis results have shown that the reactor responds smoothly to a typical load change with help of a rather large negative moderator density reactivity coefficient. In the transient of a LOCA, the water-filled containment functions very well to terminate the break discharge with the result that the core is always flooded with the water, and to limit the maximum containment pressure below 1 MPa. To optimize the initial gas volume of containment, a simplified analysis tool has been developed and verified by comparison with the RELAP5 analysis.

KEYWORDS: Innovative Integral-type reactor, Natural Circulation Core Cooling, Passive Safety System, Water-filled containment, RELAP5, RETRAN–02, Load following, LOCA, Simplified Analysis Tool

I. Introduction

It is very important to increase utilization of nuclear energy in an untapped field besides the large-scale nuclear power plant for electricity generation, from viewpoint of addressing global warming and energy security.

Japan Atomic Energy Research Institute, JAERI, has been developing a Passive Safety Small Reactor for Distributed energy system (PSRD) with a reactor thermal power of 100 to 300 MW aimed at supplying electricity to small grids, district heating, etc., on the base of an advanced marine reactor MRX¹ that has been completed at the basic design stage.

The general concept of the PSRD has been introduced already in a report². The present paper describes the function and performance of the safety systems and equipments of the PSRD, focusing on transient behaviors of load following analyzed by The RETRAN-02 and a LOCA by RELAP5. A newly developed analysis tool for the containment design optimization is also described with the parametric study results.

II. PSRD System Description

Design features of the PSRD are as follows;

(1) Adoption of a simplified integral type reactor with natural circulation and self-pressurization in the primary loop, for economy and safety improvement,

(2) Reduction of the possibility of loss-of-coolant accidents (LOCAs) by limiting the primary pipe penetration only for the safety valve line, that is, even the volume control system is not connected to the primary loop during the normal operation period, for safety improvement,

(3) Adoption of a water-filled containment and the passive safety systems/equipments including hydraulic force actuation valves (HFAVs), for safety improvement

(4) Adoption of the in-vessel type control rod drive mechanism with a magnetic detach device that is sensible to the change in the temperature, which enables the reactor to be passively shut-down in response to the abnormal rise of the coolant temperature and/or the drop of the liquid level, for safety improvement

(5) Long core life cycle over 5 years with the load factor of 100%, for economy improvement.

1. Integral Type Reactor

The concept of PSRD is shown in Fig.1. Inside the reactor pressure vessel (RPV), the core locates in the lower part, the steam generators (SGs, two sets) in the middle part, the control rod drive mechanisms (INV-CRDMs) in the upper part. Around the core, the radiation shield is provided outside the core barrel.

The SG is of the once-through, helical coil tube type. The primary cooling water flows outside the tubes, and the secondary water and the steam flow inside the tubes. The SGs are hung from the main flange of the RPV. In refueling, the center flange together with the INV-CRDMs after de-latching the control rods is removed.

The primary cooling water flows up after passing through the core by single-phase natural circulation driving force, turns out the core barrel through the flow holes, which are positioned above the SGs, and flows down through the
SGs. The water level will vary during the normal operation between the top and the bottom of the flow holes.

In order to keep the natural circulation in the RPV under start-up or accidental conditions with a decreased liquid level, many holes of 50 mm diameter are placed on the core barrel at 8 different elevations. The bypass flow through the holes is approximately 5% of the core flow during the normal operational condition.

The volume control system and the purification system are not used during reactor power operation, in order to simplify the system and reduce possibility of a loss of coolant accident due to pipe rupture. These systems, however, will be used except for the reactor power operation, e.g., prior to open the RPV cover for refueling or prior to reactor startup after closing it. These lines of the system will be completely isolated during the reactor power operation. Pipes penetrating the RPV are limited to only the pipes of the steam, the feed water and the safety valves.

Major parameters of the PSRD with thermal reactor powers of 100 MW are presented in the Table 1. The thermal efficiency of the system depends on selection of a constant full power operation or a load following operation. For the former case, it can generate the electricity of 31 MW with the regenerative cycle of the turbine, and for the latter case, 27 MW without that system.

2. Reactor Core

The core of PSRD has been designed so as to achieve long life operation over five years without refueling or shuffling for enhancement of economic competitiveness. In the present design, the fuel assembly is based on that of current PWRs, of 17 ⅛ 17 type of fuel assembly with Zircaloy-4 cladding UO₂ pellets. Fuel pin pitch (13.9mm) of the PSRD core, however, is wider than that (12.6mm) of the current PWRs to ensure efficient burn-up by greater moderation.

An issue concerning the high burn-up of core is how to suppress a rather large reactivity at the BOL. To comply with this, the PSRD adopts the fuel rods doped with Gd₂O₃ as well as the control rods that can be inserted in all fuel assemblies.

The nuclear characteristics were evaluated by core analyses with SRAC95 developed by JAERI³, which contains the ASMBURN modular for assembly calculation and the COREBN module for core burn-up calculation. Detail of the 100MWt core was already reported⁴.

The reactor power is controlled by the automatic control system with the INV-CRDM that is installed inside the RPV, developed at JAERI⁵, and with help of a negative moderator density reactivity coefficient. The chemical shim for power control and the boron injection for emergency reactor shut down are not adopted in the PSRD.

3. RETRAN-02 Analysis for Load Follow

To clarify the characteristics of response to the load

<table>
<thead>
<tr>
<th>Table 1 Major parameters of PSRD design (100 MWt)</th>
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<tbody>
<tr>
<td>Reactor power (MWt)</td>
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<tr>
<td>Power plant output (MWe)</td>
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<tr>
<td>Type</td>
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<tr>
<td><strong>Reactor coolant</strong></td>
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<tr>
<td>Operation press. (MPa)</td>
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<td>Inlet/Outlet temp.</td>
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<td>Flow rate (kg/s)</td>
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<td><strong>Reactor core</strong></td>
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<td>Equivalent dia./height (m)</td>
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<td>U²³⁵ enrichment (Wt %)</td>
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<td>Fuel inventory (t)</td>
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<td><strong>Fuel</strong></td>
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<td>Outer diameter/pitch (mm)</td>
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<td>Burnable poison</td>
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<td>No. of fuel assemblies</td>
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<td><strong>Control rod and CRDM</strong></td>
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<td>Absorber</td>
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<td>No. of control rods</td>
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<td><strong>Steam generator</strong></td>
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<td>Type</td>
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<tr>
<td>Steam temp./press.(°C/MPa)</td>
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<td><strong>Reactor vessel</strong></td>
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<td>Inner dia./height(m)</td>
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<tr>
<td><strong>Containment</strong></td>
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<tr>
<td>Type</td>
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<tr>
<td>Design press. (MPa)</td>
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<td>Inner dia./height(m)</td>
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* For regenerative cycle of the steam turbine
change of the PSRD with 100MWt power, which adopts natural circulation and self-pressurization in the primary system, a typical transient was analyzed. Since the primary loop is a single-phase flow, the RETRAN-02 with the homogenous model was used.

In the analysis, the automatic reactor control system was not taken into account, that is, the reactor power was controlled by only a self-controllability. The RETRAN-02 node model, which has 45 Volumes, 46 Junctions and 29 Heat Slabs, is shown in Fig. 2.

The feed water flow rate was given to decrease from 100% of the normal flow rate to 50% in 200 sec., and increase again to 100% after 200 sec., as shown in Fig. 3(a), of which pattern is idealized as a day-load-change. The steam flow rate changed with a slight time lag in the same way. The feed water temperature was set to a constant through the transient, and the steam showed a small amount temperature rise when its flow decreased (Fig. 3(b)).

The reactor power is shown to respond to the SG secondary flow with the time lag of 50 to 100 sec., (Fig. 3(c)). The reactor power shows undershoot of about 10% against 50% due to excessive decrease of the core flow rate (Fig. 3 (d)) causing the core inlet temperature to rise too much (Fig. 3(f)). The reactor power, however, followed well to the load increase after it once reached to 50% at about 500 sec. The pressure of core (Fig. 3 (e)) changed similarly with that of the core outlet temperature (Fig. 3 (f)).

As a result, it can be said broadly that the reactor system with the natural circulation and the self pressurization has a capability of following to the load change even without the
automatic reactor control system. In design, the reactor automatic control system to maintain the core outlet temperature or the pressure to be a constant will be adopted using the control rod movement. The burden of the reactor automatic control system, however, will be greatly reduced.

4. Water-filled Containment and Passive Safety System

The inside of the containment is filled with water, i.e., water-filled containment. There is the nitrogen gas in the upper space and the fresh water below the gas inside the containment. The RPV and the emergency decay heat removal system (EDRS) and a containment water-cooling system (CWCS) are submerged in the water. The water-filled containment has a function of safety engineered system as well as one of enclosing the area for prevention of radioactive material release to surrounding. The water inside the containment has also a role of radiation shielding instead of the concrete shield.

A thermal insulation is necessary to prevent the heat loss from the RPV into the water. The RPV is covered by the thermal insulation of the stainless steel felt. A water-tight shell (WTS) keeps the thermal insulation from being wet by water. The heat loss from the RPV with this insulation is estimated less than 1% of the rated power.

The engineered safety system of the PSRD, the passive safety system, consists of the water-filled containment, the EDRS with hydraulic force actuating valves and the SGs, as shown in Fig. 4. The main functions of this system are to maintain core flooding in cases of accidents including a LOCA and to remove the core decay heat.

Core flooding can be maintained passively by pressure balance of the containment and RPV in an early transient period of a LOCA, and with help of EDRS in a later transient period. Thus, the core flooding in a LOCA can be attained passively without ECC pumps or an accumulator.

Decay heat can be removed with the EDRS as follows. When an accident e.g., a LOCA happens, the reactor will be shut-downed, the feed water pumps stop and the isolation valves for steam and feed water lines close. Hydraulic force actuation valves shown in Fig. 5, force of which are supplied through pipes from the feed water pump outlets, open passively due to pump stop, to flow the fluid in the EDRS to the SGs. The core decay heat is transferred to water of the containment through the SG and EDRS, by natural circulation heat transfer mode. Heat is transferred from water inside the containment to that outside it through the CWCS.

The pressure increase of the containment in case of a LOCA can be suppressed by steam condensation phenomenon in the water-filled containment. The design pressure of the containment, therefore, can be greatly lowered. The pressure increase and the water level of the core in case of a LOCA depend on the initial water level or the gas volume of the containment. The relationships between the initial water level of containment and the balance pressure will be described in the following analysis.

The water tight shell (WTS) has a U-bent pipe connecting to the water inside the containment. The pipe has a very small hole at the top of U-bent, which allows to breathe between the space inside the WTS and the upper gas space of the containment according to temperature change due to the states of reactor operations such as the full power, or the cold shut-down. This pipe allows the design pressure of WTS to be very low.

In addition to the engineered safety system, the PSRD adopts a passive reactor shut down system as the safety system. De-latching the driving shaft connecting the control rod by the INV-CRDM can scram the reactor after receiving a scram signal: This is the active reactor shut down. The reactor can be also shutdown passively by inserting the control rod into the core for response to core temperature rise - Passive Actuation Scram Device for High Temperature; PASD-HT- and/or drop of liquid level inside the RPV-.
Passive Actuation Scram Device for Low Water level; PASD-LW- of which the detail description is presented in a reference\(^2\).

The PSRD can adopt optionally the system of RPV outer-cooling type of In-vessel Retention (IVR) for case of a severe core damage accident or core melt. This is that when the temperature of the RPV wall rises to a certain value due to a severe accident, thermal expansion of the arrow that is attached to the bottom of the wall will break a part of the WTS and introduce the containment water into the space between the RPV and the WTS for outer-cooling of the RPV as shown in Fig. 4.

### III. RELAP5 Analysis

In order to confirm the basic mechanisms to maintain the core coolability, a LOCA caused by an unintentional opening of the safety valve was analyzed with the RELAP5 code\(^7\).

#### 1. Analysis Condition

Figure 6 shows a noding diagram for the analysis. All the components in the containment are modeled with 83 hydraulic nodes, and 18 heat structures of the code. The upper part of the containment is modeled with two vertical pipe components each consisting of twelve nodes and facing the heat exchanger (HEX) either for the EDRS or CWCS. The nodes in the two components are horizontally connected each other at each center elevation using the multiple junction component to simulate the two-dimensional natural circulation flow. The CWCS is modeled simply by using the table component giving the heat removal rate as a function of the containment temperature. The heat sink temperature is 308K.

The analyzed accident is an unintentional opening of the SV having a throat diameter of 50mm and located at the top of the RPV. This causes the primary coolant to be discharged directory into the containment water. The core scram and SG isolation are assumed to occur concurrently with the opening of the valve at time zero. The HFAVs in the EDRS is assumed to open concurrently with the isolation of SG.

#### 2. Analysis Results

Figure 7 shows the system pressure responses. The RPV pressure decreased rapidly due to the vapor discharge from the SV, which increased the containment pressure by increasing the liquid level and thus reducing the gas volume in the containment. Both pressures became approximately the same at 0.7 MPa at 2340 sec. After that, the pressures decreased slowly due to the heat removal by the CWCS. As the containment temperature decreased, the pressure became higher in the containment than in the RPV due mainly to the gas volume expansion, which terminated the coolant discharge and caused the flow reversal at the valve. This behavior is indicated in Fig. 8, where the discharged mass from the SV began to decrease at ~15000 s due to the flow reversal. The RPV liquid level took the minimum at this timing and then increased. At the time of the minimum liquid inventory, the liquid level was 2 m higher than the core top elevation, and one third of the SG was exposed to the steam environment.

The decay heat was transferred to the final heat sink through the natural circulations inside the RPV, the EDRS connected to the SG secondary side, and the containment. The natural circulation in the RPV continued throughout the transient even after the liquid level became lower than the SG entrance elevation due to the presence of the previously-mentioned holes on the core shroud. The natural circulation prevents the thermal stratification in the RPV, and thus enhances the heat transfer at the SG. The decay heat was transferred to the SG tubes by condensation for the region above the mixture level, and free convection for the
rest.

Natural circulation in the SG secondary side was established in the loop consists of the SG, the EDRS HEX, and connecting pipes, immediately after closing the isolation valves in the main steam lines and feedwater lines, and resultant opening of the HFAVs in the EDRS. The heat transfer mode was primarily saturation boiling inside the SG and condensation inside the EDRS HEX tubes.

Stable single-phase natural circulation was calculated in the containment with the typical fluid velocity of the order of 0.1 to 1 cm/s. Because of the presence of the noncondensable gas in the upper part, the containment fluid was always subcooled. The heat transfer modes on the outer surfaces of HEXs for the EDRS and CWCS were, therefore, single-phase free convection. The temperature increase or decrease along the HEXs was less than 2 K during a long-term cooling phase. Since the RELAP5 code is based on the one-dimensional momentum transfer conservation, large uncertainties exist in the calculation of the multi-dimensional natural circulation cooling performance in the containment. This uncertainty will be discussed later.

These results have confirmed that effectiveness of the designed features to passively maintain the RPV liquid inventory, remove the decay heat through the EDRS and CWCS, and limit the containment pressure.

IV. Fast Running Analysis Tool For Design Optimization

Although the RELAP5 code can provide detailed information on the safety system performance, the calculation requires relatively long time, which is not appropriate to be used in the design process. For the case mentioned in the previous section, it required ~10000 seconds on the Window PC with a 500 MHz CPU. Furthermore, the robustness of the code calculation is not sufficient for this kind of low pressure transient, that is, the calculation was often stopped by the physical property error. A simplified lumped parameter model was, therefore, developed for the parametric study for the design optimization. The model is also helpful to capture the essences of the physical characteristics of this reactor.

1. Model Description

The RELAP5 code calculation results have indicated that the temperature difference between the core inlet and outlet was less than 2 K during a transient, which suggests the RPV may be modeled with a lumped parameter system. The lumped-parameter mass and energy conservation equations for the RPV are expressed as follows:

\[
d\left(\alpha L_{RPV} \rho L_{RPV} + \alpha G_{RPV} \rho G_{RPV}\right) \frac{dt}{dV_{RPV}} = -W_{SV}, \quad \text{and}
\]

\[
d\left(\alpha L_{RPV} \rho L_{RPV} e L_{RPV} + \alpha G_{RPV} \rho G_{RPV} e G_{RPV}\right) \frac{dt}{dV_{RPV}} = \frac{Q_{core} - W_{SV} h_{SV} - H_{SG}\left(T_{RPV} - T_{SG}\right)}{V_{RPV}}
\]

where \(\alpha\) is the phase volumetric fraction, \(\rho\) density, \(e\) internal energy, \(h\) enthalpy, \(W_{SV}\) flowrate through the safety valve, \(V\) fluid volume, \(T\) temperature, \(Q_{core}\) core power, \(H\) the product of heat transfer area and heat transfer coefficient.

Because of assumed fluid saturation, the time-derivative of the physical property can be expressed by the product of the pressure-derivative of the property and the time-derivative of the pressure. The pressure and void fraction can be, therefore, explicitly obtained from the above two equations.

Since the fluid subcooling is significantly large during the normal full-power condition, a simple but effective technique was used to take into account the initial thermal energy accurately. In this technique, the volume of the RPV \(V_{RPV}\) is changed so that it corresponds only to the saturated region. That is, the volume \(V_{RPV}\) corresponds to the region initially having the core outlet temperature when \(P_{RPV} > P_{SAT}\), where \(P_{RPV}\) is a RPV pressure and \(P_{SAT}\) is the saturation pressure corresponding to the initial core inlet temperature. When \(P_{RPV} < P_{SAT}\), \(V_{RPV}\) changes to the whole RPV volume.

For the system consists of the SG secondary side, EDRS, and the connecting pipes, the two conservation equations are expressed as follows:

\[
d\left(\alpha L_{SG} \rho L_{SG} + \alpha G_{SG} \rho G_{SG}\right) \frac{dt}{dV_{SG}} = 0, \quad \text{and}
\]

\[
d\left(\alpha L_{SG} \rho L_{SG} e L_{SG} + \alpha G_{SG} \rho G_{SG} e G_{SG}\right) \frac{dt}{dV_{SG}} = \frac{H_{SG}\left(T_{RPV} - T_{SG}\right) - H_{EDRS}\left(T_{SG} - T_{com}\right)}{V_{SG}}
\]

(2)

Similarly, the time evolution of pressure and void fraction can be obtained explicitly from the two equations. The results of RELAP5 was used to determine the initial void fraction in the SG secondary side.
Because of the presence of the cover gas, the containment fluid is always subcooled, thus the mass and energy conservation equations are required only for the liquid region in the containment expressed as follows:

\[
\frac{d(\rho_{\text{L,CON}} V_{\text{L,CON}})}{dt} = W_{SV}, \text{ and}
\]

\[
\frac{d(\rho_{\text{L,CON}} e_{\text{L,CON}} V_{\text{L,CON}})}{dt} = W_{SV} h_{SV} + H_{\text{EDRS}} (T_{\text{SG}} - T_{\text{CON}}) - H_{\text{PCCS}} (T_{\text{CON}} - T_{\text{atm}})
\]

(3)

where only the region above the safety valve elevation is taken into account as the containment volume, since the lower part of the containment are stagnant and does not affect the transient behavior. The time derivations of liquid temperature and liquid volume are solved from the above two equations.

The containment pressure is determined by the ideal gas law and the saturation pressure corresponding to the containment temperature as follows:

\[
P_{\text{CON}} = \frac{M_{\text{Air}} R_{\text{gas}} T_{\text{CON}}}{V_{\text{G,CON}}} + P_{\text{atm}} (T_{\text{CON}})
\]

(4)

where \( M_{\text{Air}} \) is the air mass and \( R_{\text{gas}} \) is the gas constant.

The SV flow rate was obtained by taking the minimum of the flows calculated with the sonic flow model and the orifice flow model.

For the calculation of the heat transfer at the SG and HEXs for the EDRS and CWCS during a transient, the steady state relationship was assumed among the heat flux, coolant temperatures, and wall temperatures. Since the heat transfer correlations tend to have strong dependency on the wall temperature, the steady state relationship was calculated iteratively.

The used heat transfer correlations are the Jens-Lottes nuclear boiling correlation\(^9\) for the SG inner side, the Langmuir correlation\(^9\) for the SG outer side, a set of film condensation correlations for the EDRS HEX inner surface, and a set of free convection correlations for the EDRS HEX outer surface. Here, the Langmuir correlation is for a horizontal tube free convection and probably underestimates the heat transfer coefficient for the present case with condensation on a part of the surface. However, due to the large SG surface area, the heat resistance is still small. The used correlation sets for the condensation and free convection are those recommended in a handbook\(^10\). That for condensation includes the Nusselt equation, and that for free convection includes Churchill and Chu\(^11\). The heat transfer performance at the CWCS HEX was modeled with the table as was done for the RELAP5 calculation. The effects of the selection of these correlations are discussed later.

Above-mentioned conservation equations are solved explicitly using a Euler method. One case requires less than 30 seconds, which is several hundreds times faster than the RELAP5 calculation.

2. Comparison with RELAP5

The pressures in the RPV and containment calculated by the present model are compared with the RELAP5 in Fig. 7. The difference after 10000 s is caused mainly by the difference in the heat transfer coefficient for the outer surface of the EDRS HEX.

The figure 8 shows the comparison of the water mass discharged from the SV. The agreement is good for the critical flow condition, while the flow rate is larger by the present model for a non-choking flow after 2000 s. This difference is also caused mainly by the smaller heat transfer coefficient for the outer surface of the EDRS HEX.

Since the present model does not provide the fluid flow information, large uncertainties arise for the prediction of the heat transfer rate at SG and EDRS. However, since the SG is sized to transfer the full power, and condensation that has generally large heat transfer coefficients occurs on the inner side of the HEX tubes of the EDRS, the heat resistance exists mostly on the outer surface of the HEX tubes of the EDRS.

The heat transfer correlation used in the present model for the free convection is that for the vertical equal temperature wall in stagnant bulk fluid. In the actual case, the flow condition is affected by the containment geometry, and the detail is not known at present. The effects of the magnitude of the free convection heat transfer were, therefore, investigated parametrically changing the multiplier factor for the correlation value. As shown in the comparison plots of Figs. 9 and 10, good agreements with the RELAP5 were obtained in the case with the multiplier factor of 5.

Since the multiplier factor affects mainly the long-term cooling behavior, the maximum pressure was not significantly affected as shown in Fig. 11. On the other hand, more significant difference was observed for the minimum liquid fraction as shown in Fig. 12. With lower value for the
free convection heat transfer coefficient, the duration of fluid discharge prolongs and thus, the minimum core inventory decreases. Note that the minimum liquid fractions was always higher than the volume fraction below the core top elevation of 0.3 of the present design, meaning the core coolability was maintained in all the cases.

These results indicate the significance of the prediction of the free convection heat transfer in the containment to assess the performance of the safety features. Since this is an analysis of the single-phase NC flow, the accurate solution can be obtained using a CFD code. The results also indicate that the present model helps to capture the essential characteristics of this reactor design, which suggests that the present model can be used for the system optimization calculation. Since the RELAP5 cannot produce the reliable information on the multi-dimensional behavior, and the smaller multiplier factor provides the conservative results, the multiplier factor of 1 will be used in the following parametric study.

3. Parametric Calculations

The most important parameters for the LOCA analysis of this reactor design are the containment pressure and the RPV minimum liquid inventory. Generally, these two parameters have trade-off relations, that is, the reduction of the containment pressure requires the large gas volume, which decreases the minimum liquid inventory in the RPV, and vice versa. This relation was calculated by the present model as shown in Figs. 13 and 14. The figure 13 shows that when the gas volume is below 30 m$^3$, the increases of the gas volume significantly reduces the maximum containment pressure. However, for the volume above 30 m$^3$, the increase of the volume does not change the pressure significantly.

The minimum RPV liquid fraction decreases with increasing the containment gas volume as shown in Fig.14. Since the volumetric fraction below the core top elevation in
the RPV is 0.3, the core cooling is maintained in all the calculated conditions. These results indicate that the selected value for \(35 \text{ m}^3\) appears to be appropriate to reduce the maximum pressure, suppressing the containment volume and maintaining the core coolability.

The size and number of the safety valve are also important design parameters for this reactor concept, since the SV is a sole location that allows the coolant discharge from the RPV for the design basis. The effects of the SV capacity are summarized in Figs. 15 and 16. As shown in the figures, with decreasing the valve diameter, the minimum liquid inventory increases and the maximum containment pressure decreases. Note that effects appear only for the valve diameter of less than about 4 cm. Although the maximum containment pressure can be reduced to 0.4 MPa by using a 10mm diameter SV, this requires 25 valves to have the same capacity, which is not practical. Thus, the present selection of 50 mm valve seems appropriate.

These results indicate the appropriateness of the base case condition to keep the pressure in the containment at low level and sufficient water inside the RPV. The results also indicate the usefulness of the present model for the design optimization.

V. Conclusion
A highly passive, simplified compact reactor, PSRD with thermal power of 100 MW, aimed at supplying energy for electricity, district heating etc., has been designed. Characteristics of the reactor system with the passive safety system were revealed through the analyses as follows:
- The reactor system with natural circulation core cooling and self pressurization can respond smoothly to a typical load change with help of a rather large negative moderator density reactivity coefficient.
- The functions of the simplified passive safety system with the water filled containment - maintenance of the RPV coolant inventory, removal of the decay heat and suppression of the containment pressure rise – have been confirmed through a LOCA analysis by the RELAP5.
- A simple analytical tool was developed that can be used for the design optimization, of which applicability was validated by the RELAP5.
- Most significant uncertainty to evaluate the safety performance arises from the prediction of the free convection heat transfer for EDRS and similarly CWCS. Since this prediction is only affected by a single-phase liquid NC, more accurate prediction can be obtained from the analysis using a CFD code.
- The maximum containment pressure is not significantly affected significantly by the uncertainty of the free convection heat transfer coefficient.

Nomenclature

CFD: Computational Fluid Dynamics
CWCS: Containment Water Cooling System
EDRS: Emergency Decay Heat Removal System
HEX: Heat Exchanger
HFAV: Hydraulic-Force Actuation Valve
LOCA: Loss-Of-Coolant Accident
RPV: Reactor Pressure Vessel
SG: Steam Generator
SV: Safety Valve
WTS: Water Tight Shell

References
2) Ishida, T., Sawada, K., Yonomo, T., et al., Proceeding of INONE-11, 36470, Tokyo, 2003
6) Mcfauden, J.H., et al., NKP1850 CCM, EPRI