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A Study on the Conceptual Design of a 1,500 MWe Passive PWR with Annular Fuel

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Abstract

In this study, the preliminary conceptual design of a 1500 MWe pressurized water reactor (PWR) with annular fuel has been performed. This design is derived from the AP1000 which is a 1000 MWe PWR with two-loop. However, the present design is a 1500 MWe PWR with three-loop, passive safety features and extensive plant simplifications to enhance the construction, operation, and maintenance. The preliminary design parameters of this reactor have been determined through simple relation to those of AP1000 for reactor, reactor coolant system, and passive safety injection system. Using the MATRA code, we analyze the core designs for two alternatives on fuel assembly types: solid fuel and annular fuel. The performance of reactor cooling systems is evaluated through the accident of the cold leg break in the core makeup tank loop by using MARS2.1 code. This study presents the developmental strategy, preliminary design parameters and safety analysis results.

I. Introduction

Korea is constructing several Korean Standard Nuclear Power Plants (1,000 MWe) and developing the APR1400, that is the new name of the Korean Next Generation Reactor (1,400 MWe), to start commercial operation after 2010. Generally power uprated plant provides the advantages from the view point of the plant economy and site area. In this regard, the concept of a 1500 MWe passive reactor is considered. This design is derived from the AP1000 which is a two-loop, 1000 MWe PWR. However, the present design is a 1500 MWe PWR with three-loop.

The present study contains the conceptual design for a 3-loop 1,500 MWe passive PWR with annular fuel by incorporating the passive safety features of AP1000 and the verification of the

system performance and safety of this reactor by code analysis. The core analysis in case of using annular fuel is performed to examine the potential for safety margin, which would accommodate a substantial increase of core power density while simultaneously providing larger thermal margins than current typical PWRs using solid fuel. That is caused by fuel design to have an annular geometry that allows internal and external coolant flow and heat removal.
The general developmental procedure for the 3-loop 1,500 MWe passive PWR is shown in Figure 1.

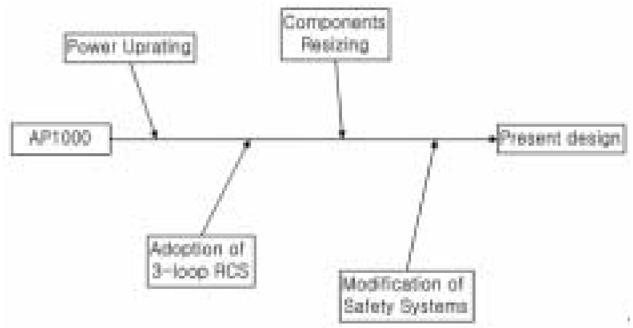

Fig. 1 Developmental strategy

II. Preliminary Conceptual Design

The reactor in the present study is designed to operate at thermal output of 4,700 MWt by using the same thermal efficiency, 31.92% as AP1000. In the core design the extra assemblies and/or an increase in the linear power density will be needed to enable the core power to be increased from 3400 MWth to 4678 MWth.

The reactor coolant system consists of three transfer circuits, each with a steam generator, two reactor coolant pumps, a single hot leg and two cold legs for circulating reactor coolant between the reactor and the steam generators. In addition, the system includes a pressurizer, automatic depressurization system, interconnecting piping, valves, and instrumentation necessary for operational control and safeguards actuation. All system equipments are located in the reactor containment.

The pressurizer is basically similar to that of AP1000, but internal volume is increased in proportion to power uprating. The calculated internal volume of the pressurizer in the present design is 2800 ft^3. This raised volume is used to provide more operational stability by absorbing pressure fluctuation as much as possible.

The steam generator of the present design can be the same component to that of AP1000. In the design of AP1000 one steam generator approximately takes charge of 545MWe. The present reactor has three steam generators and total power is 1500MWe. And therefore, the steam generator used in AP1000 is sufficient to cover the corresponding power for the present design. The AP1000 uses a Model Delta-125 steam generator.[1]

In the present design, reactor coolant pump can be also the same components to those of AP1000. The coolant pump of AP1000 is a single stage, hermetically sealed, high-inertia, and centrifugal canned-motor pump.[1] A reactor coolant pump is directly connected to each of two outlet nozzles on the steam generator channel head

The passive safety features use the same design approach and arrangement as the AP1000. The higher core thermal power of the present design requires passive core cooling system to remove more heat from the reactor coolant systems than in the AP1000. The passive core cooling system of this design is basically similar to that of AP1000, but internal volume is increased in proportion to power uprating. Table3-1 shows the overall plant design parameters.

Table 1 Overall plant design parameters

Parameters	KSNP	AP1000	Present design
NSSS thermal output [MWth]	2825	3415	4700
Gross electric output [MWe]	~1000	1090	1500
Number of S/Gs	2	2	3
Nominal RCS pressure [MPa]	15.5	15.5	15.5
Number of RCPs	4	4	6
Plant efficiency [%]	37.2	31.92	31.92
Total volume of pressurizer [ft^3]	1800	2100	2800

III. Results and Discussion

III.A. Core analysis using MATRA code

MATRA (Yoo and Hwang, 1998) code is used for safety analysis of core design in the present study, which is a multi-channel analyzer for steady states and transients in rod arrays. The MATRA has been developed to be run on an IBM PC or HP WS based on the existing CDC CYBER mainframe version of COBRA-IV-I. MATRA code calculates the local thermal hydraulic conditions, such as flow, enthalpy, pressure, void fraction in each flow channel, and the MDNBR in the hot sub-channel using the proper CHF correlations.[3]

In core design, the critical heat flux criterion is used to avoid the deterioration of heat transfer. The MDNBR is defined as the ratio of the deterioration heat flux to the maximum heat flux. It should be above 1.30. Main work in this chapter is to analyze the core designs by calculating the MDNBR for two alternatives on fuel assembly types: solid fuel type and annular fuel type.
In case of core design with solid fuel type, the core consists of 209 fuel assemblies of 17×17 arrays and the linear heat rate is 5.898 kW/ft. The fuel assemblies are set with an equivalent diameter of 3.51 m and reactor vessel inner diameter of 4.56 m. There are some assumptions in using MATRA code to calculate MDNBR. W-3 correlation is used for Critical heat flux (CHF) correlation. Grid loss is 0.6. Axial power distribution is the chopped cosine shape with peak to average value of 1.55 for axial power profile.[4] Radial power distribution is calculated by CASMO-3 code, which is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies or simple pin cells.

Figures 2 through 4 show the calculated results for the case of solid fuel type. Figure 2 shows the radial power distribution. Figure 3 and 4 indicate the changes of MDNBR in cases of core thermal power change and core inlet temperature change, respectively. The value of MDNBR is 1.54 at 100 % of core thermal power and 290 ^{o}C of core inlet temperature. However, MDBNR is below than 1.30 at 118 % overpower for transient.

Annular fuel assembly type is recently proposed at MIT. The fuel departs from the traditional solid rod design by employing the annular elements with both internal and external cooling of each fuel rod. The annular rods have larger diameter than the current fuel rods, thus using a significantly smaller number of pins in a regular size of a PWR assembly.[4] In case of core design annular fuel assembly type, the core contains 157 fuel assemblies of 13×13 arrays and the linear heat rate is 14.4 kW/ft. The fuel assemblies are set with an equivalent diameter of 2.93 m and reactor vessel inner diameter of 3.99 m.

Figures 5 through 8 show the calculated results for the case of annular fuel type. Figure 5 shows MDNBR to be calculated in cases of outer radius change in fuel region in annular fuel assembly type and change in number of assemblies. This indicates that the minimum number of assemblies is 157 with 7.41 mm of fuel outer radius to meet the MDBNR criterion.
Figure 6 and 7 show the configurations of mass flux in inner channel and outer channel, respectively. Figure 8 show the calculated MDNBR at 118 % power. The value of MDNBR is 1.33 at 118 % of core thermal power and 290 ^{o}C of core inlet temperature, which meets the criterion of 1.30.

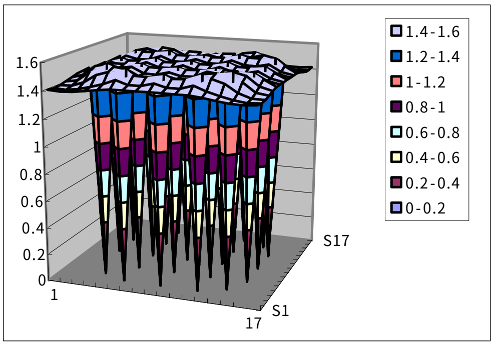

Fig. 2 Radial power distribution

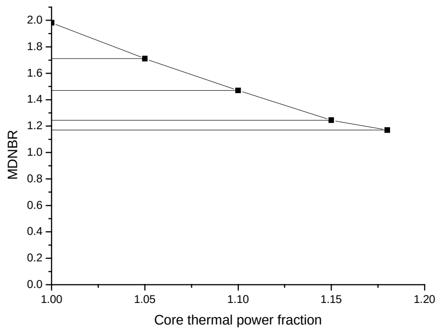

Fig. 3 MDNBR for the core thermal power

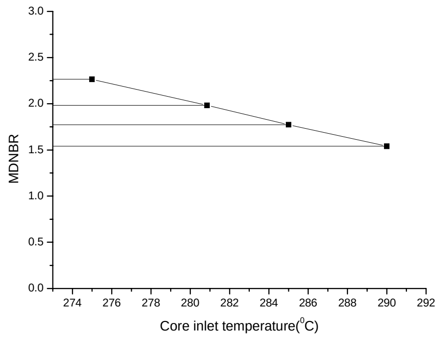

Fig. 4 MDNBR for the core inlet temperature

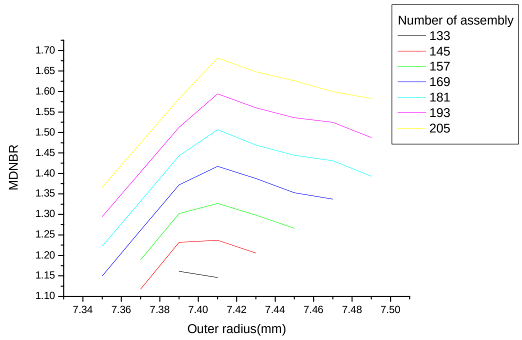

Fig. 5 MDNBR for the number of Assembly

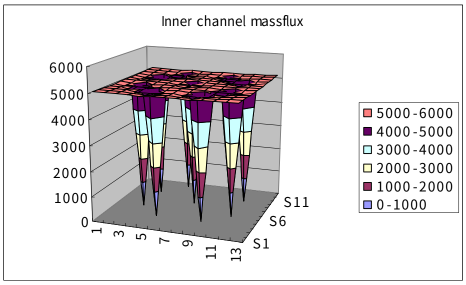

Fig. 6 Inner channel mass flux configuration

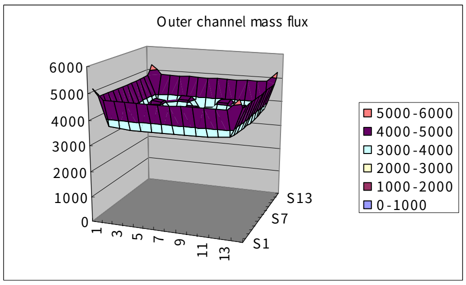

Fig. 7 Outer channel mass flux configuration

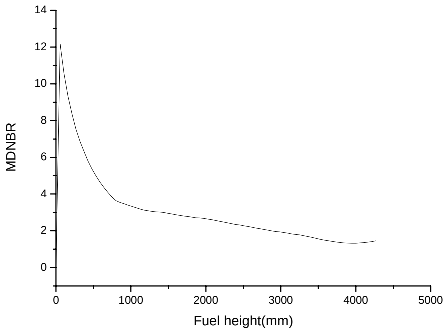

Fig. 8 MDNBR at 118 % Power

The major differences in the core design of annular fuel type compared to that of the solid fuel type are the decrease of 52 fuel assemblies and an increase in the average linear power. More differences between the annular fuel and solid fuel type are shown in Table 2. In summary, the core design with annular fuel type has the improved safety margins in comparison with current PWR fuels, which is the appreciable increase of DNBR margin due to the reduction of surface heat flux allowing a substantial power density increase by increased discharge burnup.

Table 2 Core design differences- solid and annular fuel type

Parameters	Solid-fuel	Annular-fuel
Fuel assembly type	17*17	13*13
Number of fuel assembly	209	157
Average linear power [kW/ft]	5.898	14.40
Effective core height [ft]	14	14
Core equivalent diameter [m]	3.51	2.93

III.B. System analysis using MARS code

The preliminary safety analysis about a small break loss of coolant accident (SBLOCA) of pressure balance line break between cold leg and CMT is carried out by MARS2.1 code to verify if the safety of this conceptual reactor is maintained. MARS2.1 code is a multidimensional thermal-hydraulic system code, which has developed by combining and restructuring the

RELAP5/MOD3.2.1.2 and COBRA-TF codes at KAERI for a multi-dimensional and multi-purpose realistic thermal-hydraulic system analysis of light water reactor transients.[2]

The passive safety design approaches are to depressurize the reactor coolant system if the break or leak is greater than the makeup capability of the makeup system or if the non-safety makeup system fails to perform. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core.

The passive core cooling system includes core makeup tanks, accumulators, a large IRWST, and the PRHR heat exchanger. The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. Gravity head of the colder water in the core makeup tanks provides the injection of the core makeup tanks. The core makeup tanks are located above the reactor coolant loops and each of them is equipped with a pressure balancing line from a cold leg to the top of the tank. The pressurized accumulators provide additional borated water to the reactor coolant system in the events of a LOCA. The IRWST provides additional water for long-term core cooling. As the reactor system depressurizes and mass is lost to the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state.[1]

The calculated results are shown in Figures 9 through 15. Major output parameters such as collapsed water level of core and fuel cladding temperature are shown in these figures. The pressure of primary side is illustrated in Figure 9. It indicates that when the accident is initiated, RCS pressures are decreased. Figures 10 and 11 show the collapsed liquid level of the pressurizer and the core, respectively. The core and pressurizer flow rates are dependent on the loop flow rates. Figures 12 and 13 show mass flow rate and void fraction of core, respectively. The core mass flow rate begins to drop rapidly. Figure 14 shows the injection flow of CMT, accumulator, and IRWST, which indicates that CMTs provide high-pressure safety injection in the event of a small-break LOCA and the pressurized accumulators provide additional borated water to the reactor coolant system.

Throughout the calculated results, the core never uncovers and the peak cladding temperature occurs at the inception of the event. The calculated peak cladding temperature behavior is illustrated in Figure 15. The estimated peak cladding temperature at this break is 635 K.
The accident of the cold leg break in the core makeup tank loop shows that the passive safety systems in this design are sufficient to mitigate the small break LOCA.

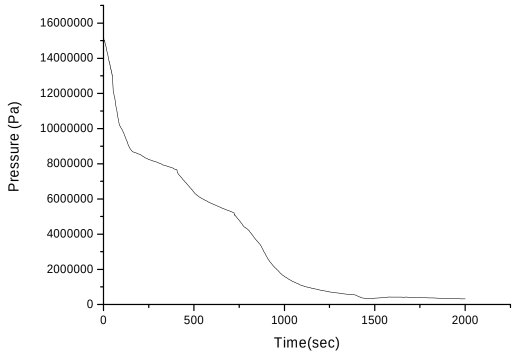

Fig. 9 Pressure of RCS

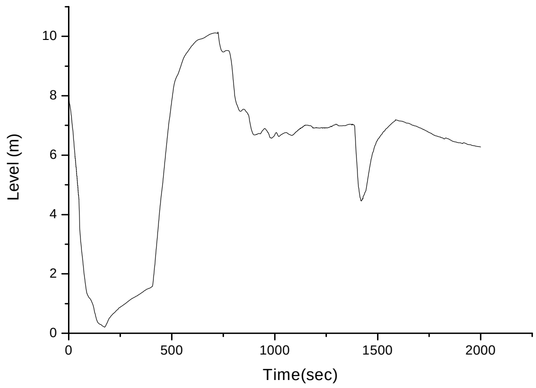

Fig. 10 Water level of pressurizer

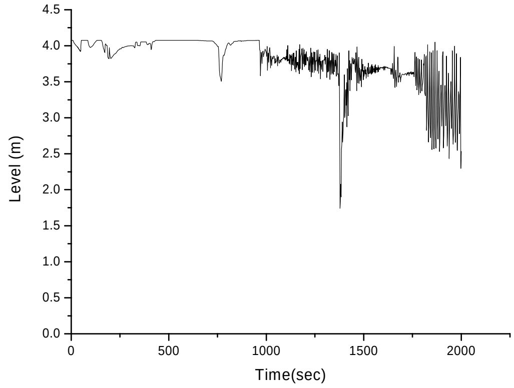

Fig. 11 Water level of core

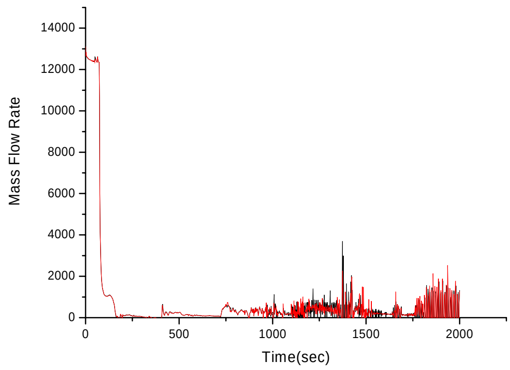

Fig. 12 Mass flow rate of core

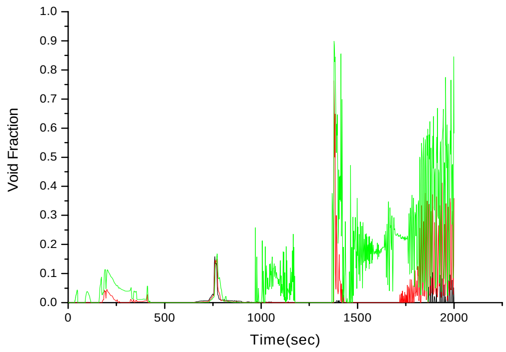

Fig. 13 Void fraction of core

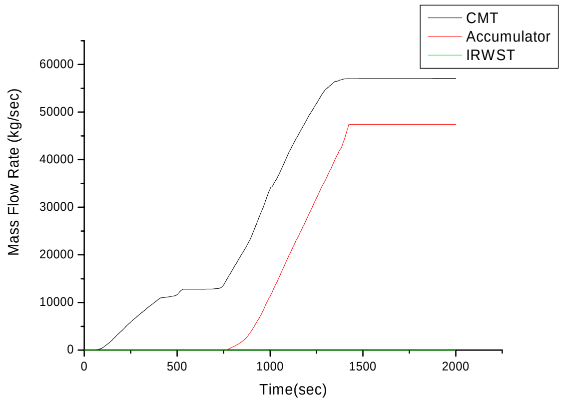

Fig. 14 Injection flow of CMT, accumulator, and IRWST

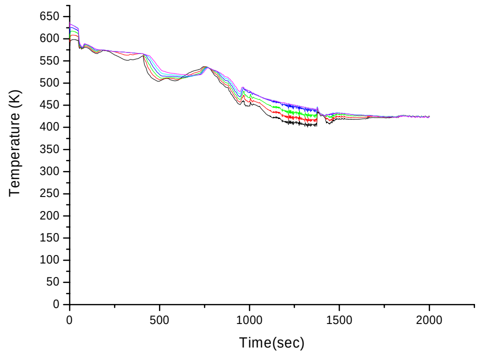

Fig. 15 Hot rod peak cladding temperature

IV. Conclusions

In this study, the design concept of a 1500 MWe PWR has been proposed. From previous results, the following conclusions can be summarized.The preliminary design of reactor is accomplished based on the parameters of AP1000 and reactor coolant system is designed just by adopting an added loop.

Using the MATRA code we analyze the core designs for two alternatives on fuel assembly types: solid fuel and annular fuel. The core design with annular fuel type has the improved safety margins in comparison with current PWR fuels, which is the appreciable increase of DNBR margin due to the reduction of surface heat flux allowing a substantial power density increase by increased discharge burnup.

The accident analysis for performance of passive safety systems is performed by using MARS2.1 code. This plant design includes passive safety features to prevent or minimize core uncovery during the small-break LOCA. The accident of the cold leg break in the core makeup tank loop shows that the passive safety systems in this design are sufficient to mitigate SBLOCA.

References

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