PRELIMINARY DESIGN CONCEPTS OF AN ADVANCED INTEGRAL REACTOR

KAP S. MOON, DOO J. LEE, KEUNG K. KIM,
MOON H. CHANG, SI H. KIM
Korea Atomic Energy Research Institute,
Taejon, Republic of Korea

Abstract

An integral reactor on the basis of PWR technology is being conceptually developed at KAERI. Advanced technologies such as intrinsic and passive safety features are implemented in establising the design concepts of the reactor to enhance the safety and performance. Research and development including laboratory-scale tests are concurrently underway for confirming the technical adoption of those concepts to the reactor design. The power output of the reactor will be in the range of 100MWe to 600MWe which is relatively small compared to the existing loop type reactors. The detailed analysis to assure the design concepts is in progress.

1. Introduction

The nuclear reactors currently under development in the worldwide nuclear societies are largely categorized into two different concepts with respect to the configurations of major primary components ; namely, loop type and integral type. Most of power reactors that are currently in operation and under development have loop type configurations which enable large-scale power output and thus provide economical power generation. On the other hand, integral reactors receive a wide and strong attention due to its characteristics capable of enhancing the reactor safety and performance through the removal of pipes connecting major primary components, even for a certain power limit due to the limited reactor vessel size which can be manufactured and transportable. The relatively small scale in the power output of integral reactors compared to the loop type reactors, however, draws a special concern for the various utilization of the reactor as an energy source, as well as power generation especially for the small-sized grid system.

Small and medium reactors with integral configurations of major primary components are actively being developed in many countries. The design concepts of those reactor vary with the purposes of application. Since the second half of 1995, Korea Atomic Energy Research Institute (KAERI) has been putting efforts to research and develop new and elemental technologies for the implementation to the advanced reactors. In parallel with those efforts, an advanced integral PWR by implementing those technologies and also passive safety features is under conceptual development. The electrial power output of the reactor will be in the range of 100MWe to 600MWe depending on the purpose of utilization such as power generation, energy supply for the seawater desalination and others. As far as the electricity generation concerned, this range of power output is considered as suitable for energy supply to the industrial complexes, remotely located islands, and specially isolated areas. The reactor core is conceptually designed with no soluble boron and hexagonal fuel assemblies to enhance the operational flexibility and to improve the fuel utilization. The reactor safety systems primarily function in a passive manner when required.

This paper describes the conceptual design features of the advanced integral reactor under development at KAERI, and also important R&D subjects concurrently in progress in order to prove and confirm the technical feasibility of design concepts.

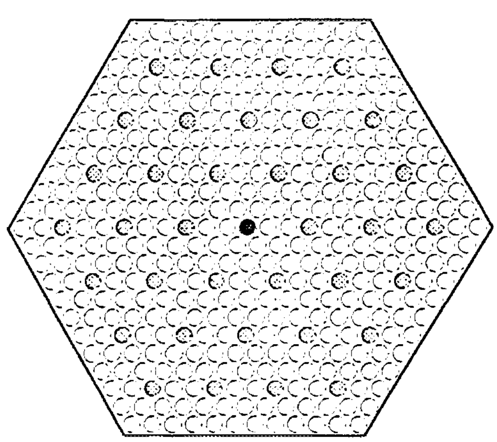
2. Reactor Design Concepts

In general, an integral type of reactor contains all major primary components such as core, steam generator, pressurizer, and reactor coolant pumps in a single pressurized reactor vessel, which mainly differs in concept from the loop type reactor. KAERI's advanced integral reactor also applies the same general definition of integral reactors.

2.1. Reactor Core and Fuel

The achievement of intrinsic safety and operational reliability is a concern of most importance in the core design. To this end, the low core power density and soluble boron free operation are implemented as major design features of the core. The low core power density and thus increased thermal margins with regard to the critical heat flux ensure the core thermal reliability under normal operation and accident conditions. This feature, furthermore, provides passive safety benefits with respect to the enhanced negative feedback for lower operating fuel temperatures and inherent power distribution stability. The elimination of soluble boron from the primary coolant becomes a major potential simplification for the advanced reactors. From the point of the view of the reactor control and safety, soluble boron free operation offers potential benefits through the presence of a strong negative moderator temperature coefficient over the entire fuel cycle. This design feature thus provides much improved passive response for a variety of performance transients and load changes. As a result of the above two important design features, the core is more stable and resistant to transients, and therefore provides improved operational flexibility. The longer refueling cycle such as 18months or longer is adopted for the purpose of improving the plant availabilty.

Fuel assembly adapts a semi-tight hexagonal geometry to improve the fuel utilization through a relatively high plutonium conversion ratio compared to the conventional LWRs. The fuel design is based on the existing Korean Optimized Fuel Assembly (KOFA) design technology. The hexagonal fuel assembly yields the lower moderator to fuel volume ratio(V_m/V_f) and the hardened neutron spectrum which result in stronger moderator temperature coefficients and higher plutonium conversion ratio. The fuel rods are the same as those of the KOFA except geometrical arrangement which is changed from the square array to the hexagonal array. Fuel utilizes low enrichment, uranium dioxide fuel, which is operated at a low specific power density(19.6kW/kgUO_2). The uranium enrichment of the fuel will be selected to achieve the 18 months(or longer) operating cycle. As shown in Fig.1, the fuel assembly is a hexagon with 22.9cm in lattice pitch and is provided to accommodate the control assembly in each fuel assembly. The fuel assembly consists of 360 fuel rods and 36 guide tubes for control absorbers and/or insertable burnable absorbers and 1 guide



Lattice Pitch: 22.9cm
Rod Pitch: 1.142cm
Flow Area: 196cm^2
No of Rods: 397
No of Fuel Rod: 360
No of GT for CR: 36
No of GT for Instrument: 1

FIG. 1. Hexagonal fuel assembly.

tube for central in-core instrument. The same fuel assembly is utilized in the core design regardless of the reactor power output.

For 100MWe and 600MWe power output as examples, the reactor core is rated at 300 MWt with 55 fuel assemblies and 1933MWt with 151 fuel assemblies, respectively. The corresponding average linear heat generation rates are 8.4 kW/m and 9.7 KW/m which are much lower that of conventional PWRs. Table 1 shows major design parameters of the conceptual designs for the core and fuel.

TABLE 1. BASIC DESIGN PARAMETERS OF ADVANCED INTEGRAL REACTOR

Reactor Core and Fuel

Nominal Core Power, MWt	1933(a)	300(b)
Power Density, KW/l	77.3(a)	66.7(b)
Avg. Linear Heat Rate, KW/m	9.7(a)	8.4(b)
Active Core Height, m	3.66(a)	1.8(b)
Effective Core Diameter, m	3.12(a)	2.0(b)
Number of FAs	151(a)	55(b)
Fuel Rod Descriptions		
Fuel Type	UO_2	
Enrichment(Equil.), w/o	~ 3.5	
Clad Material	Zircaloy-4	
Fuel Pellet OD, cm	0.784	
Clad OD, cm	0.91	

Primary Circuit

Design Pressure, MPa	17
Operating Pressure, MPa	12.5
Coolant Inlet Temperature, °C	285
Coolant Outlet Temperature, °C	315
Coolant Flow, Kg/sec	1.2×10^4(a) 1.8×10^3(b)

Pressurizer

Type	Gas/Steam Self-Pressurizer

Steam Generator

Steam Temperature, °C	290
Steam Pressure, MPa	4.7
Superheat, °C	30
Feedwater Temperature, °C	240
Tube Material	I690 T/T
Tube Diameter, mm	19

Reactor Coolant Pump

Type	Glandless, Wet Winding Canned Motor
Number	4

Containment Overpressure Protection

Type	Passive, Steam Driven Injector

Reactor Safety Systems

Decay Heat Removal	Passive, Natural Convection Hydraulic Valve/Heat Pipe
Reactor Shutdown	Control Rods/Boron Injection
Emergency Core Cooling	Not required

Note : (a) for 600MWe, and (b) for 100MWe Power Output

2.2. Primary Circuit

Fig. 2 shows the general arrangement of the primary components and internal structures of the reactor pressure vessel. Above the reactor core, helically coiled once-through steam generator is located between the core support barrel and reactor vessel. Thermal shields are provided around the core to reduce

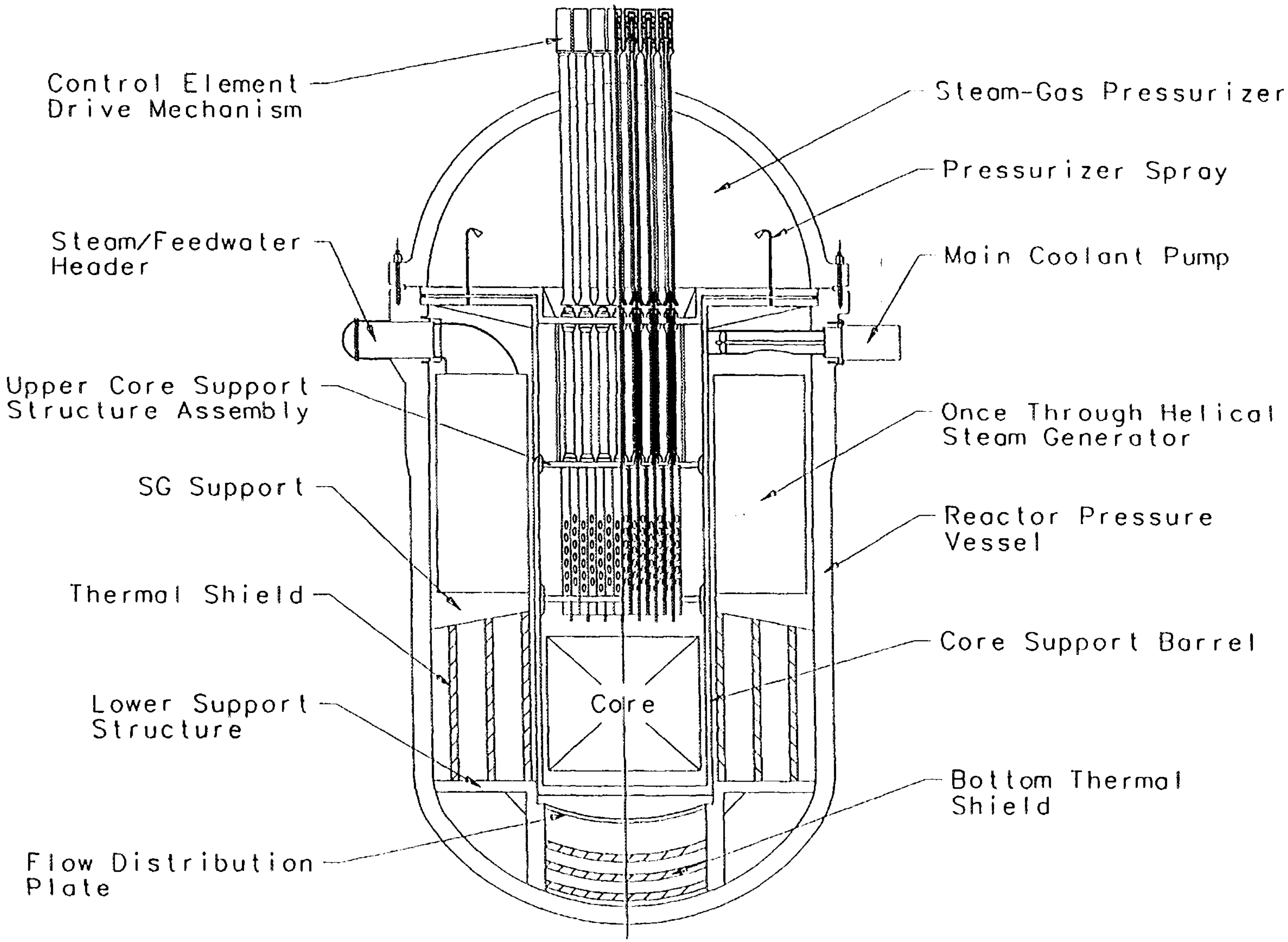


FIG. 2. General arrangement of primary components and reactor internals.

the neutron fluences on the reactor vessel. The canned motor pumps are horizontally installed on the reactor vessel above the steam generators. The upper plenum of the vessel forms a pressurizer to maintain the operating pressure of the reactor. Since all the primary system components are installed in a single pressure vessel, there is no primary pipings between major primary components and thus it completely eliminates the large break LOCA. The primary circuit is designed to provide the enhanced natural circulation capability through the sufficient temperature difference between cold and hot water along with the sufficient difference in height between the core and steam generator to produce the driving force to circulate the primary coolant. The reactor vessel is surrounded, as shown in Fig. 3, with another vessel called as safe guard vessel which contains water up to the level of the top of steam generator. The water in the safe guard vessel is pressurized with the nitrogen gas at approximately the atomospheric pressure, and is served as an interim heat sink for the emergency decay heat removal system that will be described in the next section. This section describes the design concepts of major primary components, and Table 1 summarizes some of basic design parameters of the reactor systems.

■ Steam Generator : The helically coiled once-through steam generator(SG) is located within the reactor vessel in the annular space between the core support barrel and the reactor vessel inner wall. The SG is designed to completely evaporate the secondary coolant in a single pass through the S/G tube side. Since the current design concept adopts primary circuit natural circulation operation to produce approximately 50% of full power for a relatively small power output reactor design, the SG will be located high above the core considering the current manufacturing capability of a single pressure vessel. The SG consists of groups of tube bundles, downcomer, feed water and steam headers, shrouds to guide the primary flow, and tube supporting structures. The design utilizes Inconel 690 tubing and the tube bundles are supported by perforated radial support plates so that the load can be transferred to the bottom support structure located on the supporting lug. The size of the SG will be selected depending on the sclae of power output with consideration of simplifying many of operational concerns including the access for in-service inspection and maintenance.

■ Pressurizer : The large free volume above the primary coolant level is designed as a self-pressurizing pressurizer This upper part of the reactor

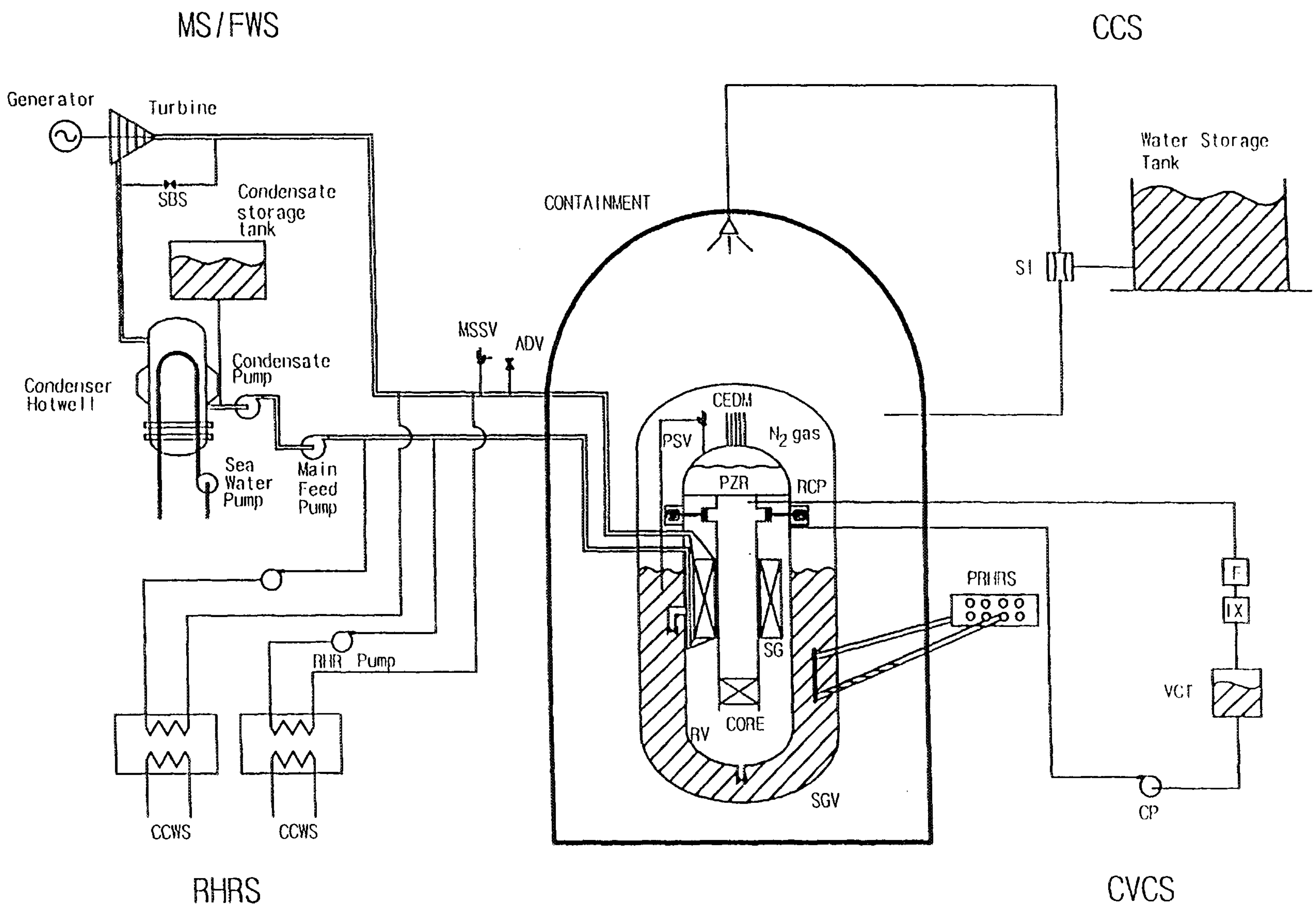


FIG. 3. Schematic diagram of advanced integral reactor systems.

vessel is thus filled with the mixture of nitrogen gas and steam providing a surface in the primary circuit where liquid and vapor are maintained in equilibrium at saturated condition. The pressure of the primary system is equal to the gas partial pressure plus the saturated steam pressure corresponding to the core outlet temperature. The reactor therefore operates at its own operating pressure matched with the system status. The nitrogen gas partial pressure is chosen to maintain subcooling at the core exit to avoid boiling in the hot channel during transients. The volume of gas space is large enough to prevent the safety valves from opening during the most severe design basis transients.

■ Reactor Coolant Pump : The reactor coolant pumps are sealed type canned rotor pumps with added inertia to increase the pump rundown time. With no shaft seals in the pump, the small LOCA associated with seal failure of the pump as in the conventional standard design is eliminated. The required number of pumps and pump capacity to circulate the primary coolant can be reduced by the design characteristics of the primary circuit natural circulation capability.

■ Control Element Drive Mechanism(CEDM) : The design of soluble boron free core results in the only use of control rods for the reactivity control and load change operation and thus requires a fine positioning control capability of the control rod. In addition, the adoption of a self-pressurizer in the upper plenum of the reactor vessel introduces difficulties in lubricating the moving parts with the primary coolant since the latch mechanism of control rods will be located in the steam-gas region of the pressurizer. These reasons yield the useless of the existing magnetic jack type CEDM. Consequently, a new concept of CEDM is developed and adopted. The design of CEDM consists of position encoder, brushless DC servo motor, lift magnet coil, rare earth permanent magnet rotor, driving tube, and split ball nut assembly. The fine control capability of CEDM is assured by the use of ball nut-lead screw mechanism. When the scram of the reactor is required, the current supply to the lift magnet coil is cut off once the signal is issued , and then the split ball nut releases the lead screw to drop down the control rods by gravity and spring forces. The worth of control rods provides sufficient shutdown margin at any conditions of reactor operation.

2.3. Engineered Safety Features

The safety concepts of the advanced integral reactor under currently conceptual development are basically taking advantages from the characteristics of intrinsic and passive safety principles on which most of small and medium reactors rely. The passive safety concept applies to the major engineered safety features as shown in Fig. 3 and described below.

■ Passive Decay Heat Removal System : When the normal decay heat removal is required, the steam generators with turbine bypass system are used to reject the heat to the condenser. This can be achieved by natural circulation on the primary side but requires feed pumps and other equipments on the secondary system. If the secondary system is not available, active decay heat removal systems with steam generators are used and the heat is removed through the component cooling system. Should there be no ac power available, the core decay heat is removed to the water contained in the safe guard vessel through the natural convection system, as shown in Fig. 3, with passive actuation of initiation valves installed on the side and bottom of the reactor vessel. The heat is then passively removed through the heat pipes to the outside of the containment. Therefore, there provides theoretically infinite time of heat removal without any intervention by operator. One of the advantages of the passive decay heat removal system usinh heat pipes is that the system can be continuously operating during normal operation to remove the heat transferred from the reactor vessel to the water in the safe guard vessel through the wet thermal insulation.

■ Passive Emergency Core Cooling System : Since all large primary circuit pipes are eliminated, the large LOCA is intrinsically not considered and thus no conventional emergency core cooling system is required. However, the break in the connection pipe from the chemical and volume control system(CVCS) may cause the loss of the primary inventory through the siphoning effect. To prevent the siphoning loss of the reactor water inventory in the hypothetical event of a CVCS line break, the installation of a siphon breaker is conceptually considered. Since the reactor vessel is always externally flooded with the water in the safe guard vessel, there is no need for the external emergency core make-up. The safe guard vessel is sized to provide a minimum of 72 hours heat removal without the operator intervention.

■ Reactor Shut-Down System : The reactor shut-down system is consisted of the control rods and the emergency boron injection system. The reactor trip at emergency is accomplished by simutaneous insertion of control rods into the reactor core by gravity following the control element drive mechanism de-energization which is actuated by trip signals from the automatic control system. In case of failure to actuate the eletromechanical protection system, the borated water from the emergency boron injection system shutdowns the reactor. The individual system is fully capable of shutdowning the reactor and provides sufficient shutdown margin to keep the reactor in a subcritical condition.

■ Passive Containment Cooling System : The containment overpressure protection is provided by a passive containment spray system. Since the hypothetical pipe break is small-sized, the pressurization rate of the containment is much slow compared to that of the conventional loop type reactors. When the energy removal from the containment is required to prevent the containment pressure from exceeding the design pressure, the steam injector driven containment spray system passively actuates as the containment energy released from the break is supplied to the system. The steam injector is a simple and compact passive pump that is driven by supersonic steam jet condition. The steam injector pumps up the water from a water storage tank to the spray nozzles located at the top of the containment.

3. Research and Development Activities

In parallel with preliminarily constructing the design concepts of an advanced integral reactor, various R&D subjects are concurrently under study. The purposes of those R&D activities are two folds : to provide the proper technical data for the design features, and to evaluate the technical feasibility and characteristics of those design concepts. Major R&D activities are as follows :

■ Hexagonal Semi-Tight Lattice Fuel Assemby : Neutronic Design and analysis methodology is under development for analyzing the reactor core with hehxagonal semi-tight lattice fuel assmblies. Thermal-hydraulic tests such as critical heat flux and pressure drop tests will be conducted to evaluate the

T/H phenomena and behavior of the fuel assembly. The suitable T/H analytical models including T/H correlations will also be developed.

▮ No Soluble Boron Core Concept : The use of no soluble boron in the core design causes to utilize large amount of lumped burnable absorbers to properly hold down the excess reactivity at the beginning of cycle and to install considerable number of control rods for the reactor control and operation. The optimization in the number of burnable absorbers and control rods is required with respect to the reactivity compensation with fuel burnup and reactor control through the cycle, and this study in conjunction with the core design with hexagonal fuel assemblies are thus investigated in this R&D subject.

▮ Natural Circulation for Integral Reactor : The natural circulation is an important design feature of the reactor. The thermal-hydraulic characteristics of the primary circuit is thus being investigated to prove and confirm the design concept through experimental tests and the analysis using computer codes.

▮ Helically Coiled Once-Through Steam Generator : A thermal-hydraulic design and performance anlaysis code - ONCESG for a once-through SG has been developed and tested against available design data of similar types of SG which are designed for other integral reactors. Further improvements of the code are under progress for the application to more complicated geometrical design and analysis. Experimental investigations are also being performed to generate the proper heat transfer and pressure drop correlation applicalble to the current design concept.

▮ Passive Equipments for Residual Heat Removal System : The characteristics of the two important passive installations, hydraulic valve and heat pipe, is currently investigated regarding their performance and reliability. A small scale of those equipments will be experimentally tested. Analytical models of those installations are also being developed for the use in the analysis of the thermal-hydraulic behaviors.

▮ Steam Injector Application to Passive Containment Cooling System : In order to investigate the performance and technical application of a steam injector concept, theoretical and experimental study is being conducted through this R&D acticity. A computer code is also under development for the analysis of thermal-hydraulic behaviors of the steam injector.

▮ <u>Wet Thermal Insulation</u> : This concept is implemented to properly protect the unnecssary heat transfer from the reactor vessel to the water contained in the safe guard vessel. An experimenatal investigation is underway for the proper material selection and performance tests for the wet thermal insulation concept.

▮ <u>Fluidic Diode Application to Passive Pressurizer Spray System</u> : A study on the fluidic diode device is experimentally being conducted for it's use in the passive pressurizer spray system. The study also includes the development of analytical models and computer codes for the analysis of the thermal-hydraulic behavior of the device.

▮ <u>Other R&D Activities</u> : Besides the above major R&D activities, several elemental technologies are currently being studied at KAERI to seek for their possible application to the advanced reactor design.

4. Summary and Remarks

A small and medium advanced integral reactor under currently conceptual development at KAERI based on PWR technology fundamentally utilizes the intrinsic and passive safety features to enhance the safety and reliability of the reactor. The fundamental safety charateristics of the reactor are summarized as follow :

- Low core power density that results in the increase in thermal margins provides much improved passive response for a variety of performance transients.
- Subtantially large negative MTC resulting from no use of soluble boron offers potential benefits on the inherent power stability and resistance to transients.
- Integral configuration of primary components in a single pressure vessel basically eliminates the large-size pipings and thus large break of loss of coolant accident.
- Large volume of primary coolant provides more thermal inertia and thus much enhanced resistance to transients.
- Large passive pressurizer significantly reduces the pressure increase for the decreased heat removal events.

- No reactor coolant pump seals eliminates a potential of small LOCA associated with the seal failure.
- Adoption of various passive safety systems enhances the reactor safety and reliability which are the key concerns in advanced reactor development.

The preliminarily established design concepts of the reactor require more detailed evaluation and analysis for both the integrated concept and individual design features to technically prove and confirm its concepts. The overall evaluation and analysis is now in progress. Advanced technologies adopted in constructing the design concepts are also independently being studied to assure its technical feasibility and to generate necessary basic data for the analysis and evaluation of integrated reactor design concepts. The further evaluation and analysis may possibly result in some changes and modifications in design concepts.