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AN INTEGRAL REACTOR DESIGN CONCEPT FOR A NUCLEAR CO-GENERATION PLANT

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Abstract

An integral reactor concept for nuclear cogeneration plant is being developed at KAERI as an attempt to expand the peaceful utilization of well established commercial nuclear technology and related industrial infrastructure such as desalination technology in Korea. Advanced technologies such as intrinsic and passive safety features are implemented in establishing the design concepts to enhance the safety and performance. Research and development including laboratory-scale tests are concurrently underway to evaluate the characteristics of various passive safety concepts and provide the proper technical data for the conceptual design. This paper describes the preliminary safety and design concepts of the advanced integral reactor. Salient features of the design are hexagonal core geometry, once-through helical steam generator, self-pressurizer, and seismic resistant fine control CEDMs, passive residual heat removal system, steam injector driven passive containment cooling system.

1.0 INTRODUCTION

The drought experienced due to the climatic anomalies and the worsening level of pollution have reduced inland water resources significantly for a number of years. A nuclear co-generation plant which can be used for sea water desalination as well as electricity generation can provide a solution in some coastal countries such as Korea and middle east nations. In this regard, Korea Atomic Energy Research Institute (KAERI) has undertaken a study for the development of advanced integral reactor for the application to these purposes as an attempt to expand the peaceful utilization of nuclear energy.

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 inherent characteristics of enhancing the reactor safety and performance through the removal of pipes connecting major primary components. Small and medium reactors with integral configurations of major primary components are actively being developed in many countries. The design concepts of those reactors vary with the purposes of application.

KAERI has been putting efforts to research and develop new and elemental technologies for the implementation into the advanced reactors. In parallel with those efforts, an advanced integral PWR with implementation of those technologies as well as passive safety features is under conceptual development. The reactor power of 300 MWt is considered as suitable size for energy supply to the industrial complexes, remotely located islands, and especially isolated area. The reactor core is conceptually designed with no soluble boron and hexagonal fuel assemblies to enhance the opertional flexibility and to improve the fuel utilization. The reactor safety system primarily functions 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 the design concepts.

2.0 System Description

2.1 Reactor Core and Fuel Design

Table 1 gives the basic reactor parameters. The fuel design is based on existing KOFA (Korean Optimized Fuel Assembly) design technology. Most design parameters of fuel rods are the same as those of the KOFA except geometrical arrangement which is changed from the square array to hexagonal array. The hexagonal fuel assembly yields the lower moderator to fuel volume ratio(V_m/V_f) and the hardened neutron spectrum which results in stronger

TABLE 1 BASIC REACTOR PARAMETERS

MIRERO - PLANT DATA			
Design Lifetime	60 years	Reactor Type	PWR
Thermal Power	300 MWt	Plant Style	Integral Primary Circuit
Primary Circuit		Pressurizer	
Design Pressure	17 MPa	Type	Integral with RV Self-Pressure Control
Operating Pressure	12.5 MPa		
Coolant Flow	1.8×10^3 Kg/sec		
Core Inlet Temp.	285 °C	Main Coolant Pump	
Core Outlet Temp.	315 °C	Number	4
		Type	Glandless, Wet Winding
Reactor Core			
Moderator	Light Water		
Fuel	Low Enriched UO_2	Containment	
Fuel Assembly	Hexagonal	Type	Passive, Steam Injector Driven
Reactivity Control	Fuel Loading, Burnable Poison, Control Element Assemblies, No Soluble Boron	Safety Systems	
		Decay Heat Removal	Passive, Natural Convection - Safe Guard Vessel, Heat Pipe, Hydraulic Valve
Clad Material	Zircaloy-4		
Power Density	66.7 KW/liter		
Avg. Linear Heat Generation Rate	8.4 Kw/m		
Refuel Cycle	24 months	Emergency Core Cooling	Not Necessary
Active Core Height	1.8 m		
Active Core Diameter	2.0 m		
Steam Generator			
Type	Helical - Once Through		
Steam Temp.	290 °C		
Steam Pressure	4.7 MPa		
Superheat	30 °C		
Feedwater Temp.	240 °C		
Feedwater Flow	174 Kg/sec		
Tube Material	I690 T/T		

moderator temperature coefficients and higher plutonium conversion ratio. The effective fuel rod length is reduced to 180cm. Fuel utilizes low enrichment, uranium dioxide, which is operated at a low specific power density(17 kW/kgUO_2). The uranium enrichment of the fuel is selected to achieve the 18 months(or longer) operating cycle. The fuel assembly section is a 22.9cm hexagon and the geometry is provided to accommodate control element 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 tube for central in-core instrument.

The core is rated at 300MWt and consists of 55 fuel assemblies. The average linear heat generation rate is 8.4kW/m which is much lower than that of conventional PWRs. The low power density and increased thermal margins with regard to critical heat flux ensure the core thermal reliability under normal operation and accident conditions. The core is designed to operate without the need for reactivity control using soluble boron over the whole power range. The elimination of soluble boron from the primary coolant is a major potential simplification for the advanced light water reactor. 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 and therefore improves reactor transients and load follow performance. Control rods provide the means of core reactivity control except for long term reactivity compensation for fuel depletion provided by the burnable poison and have enough shut down margin at any time under cold clean conditions including refueling conditions.

2.2 Primary Circuit

Internal Arrangements and Flow Paths

Figure 2 shows the general arrangement of the reactor pressure vessel and its internal structures. The core is low down in the vessel. Helically coiled once through steam generator is located between the core support barrel(CSB) and reactor vessel. Thermal shields are provided below the steam generator surrounding the core to reduce the neutron fluence level on the reactor vessel. There are four main coolant pumps installed on the reactor vessel above the steam generators. The upper plenum of the vessel forms a pressurizer to maintain the operating pressure.

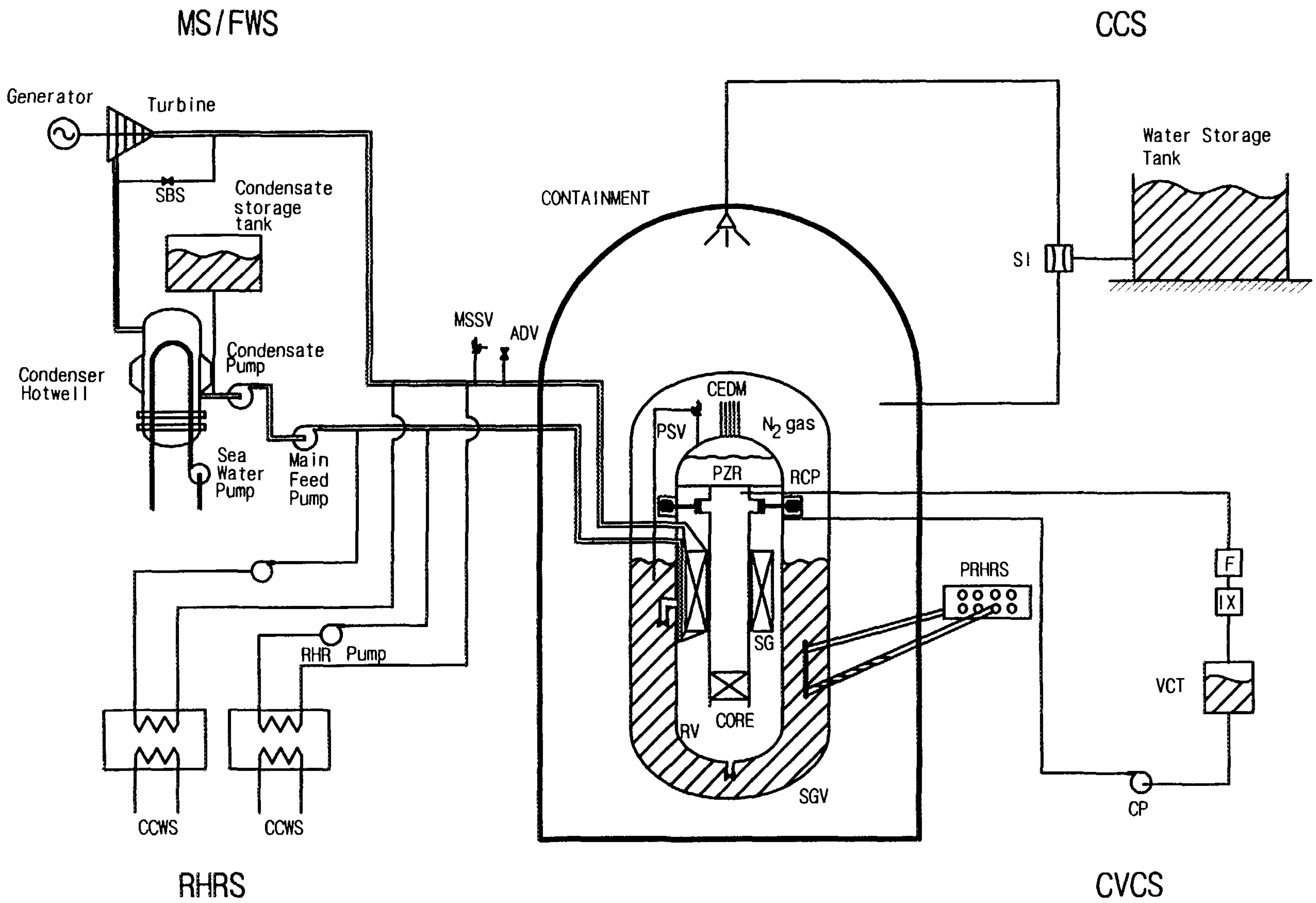

Figure 1. Schematic Diagram of Advanced Integral Reactor Systems

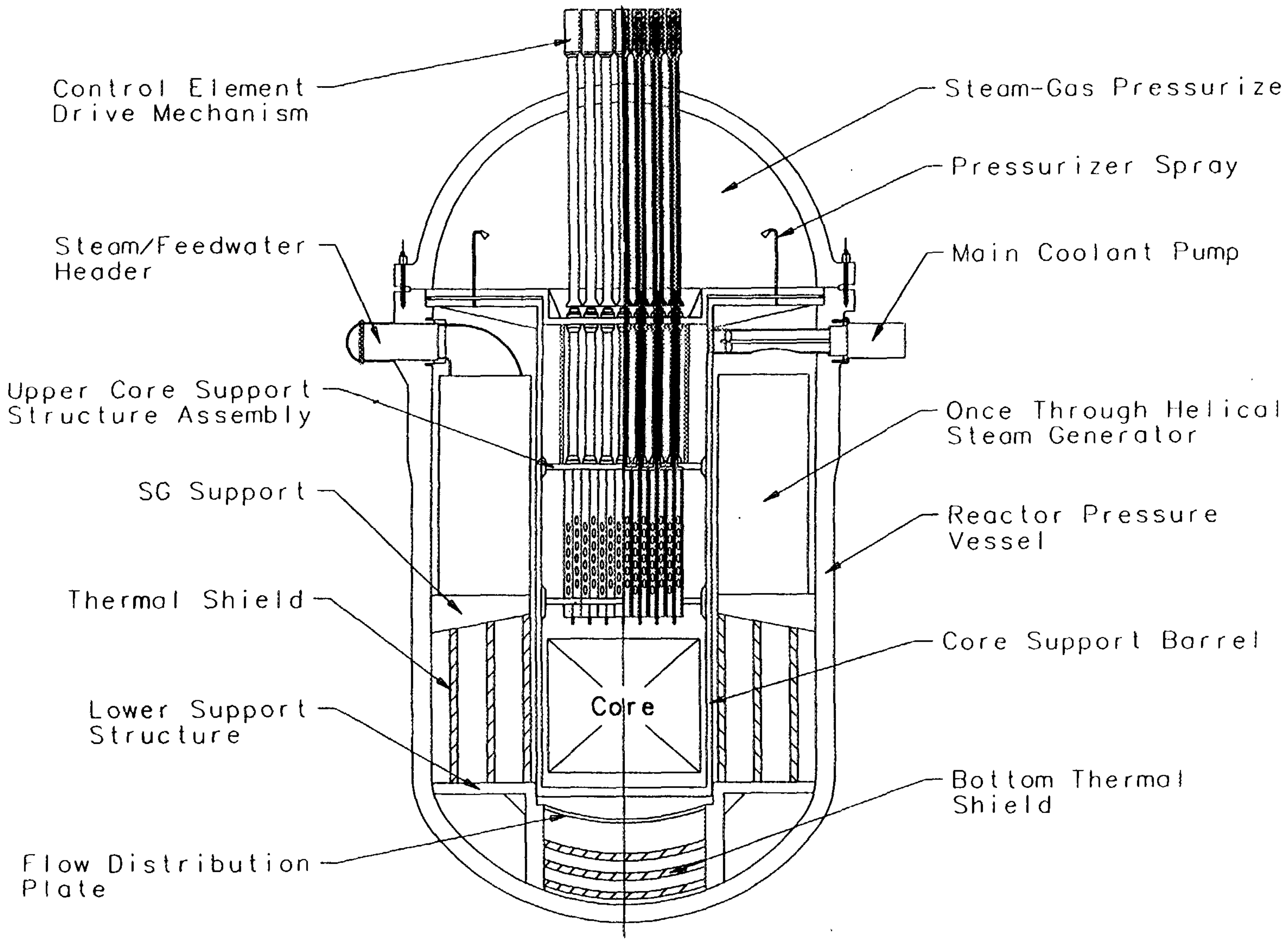

Figure 2. General Arrangement of MIRERO

The primary coolant flows up through the core and CSB riser, through the pumps, down through the steam generator and back to the lower plenum under the core. The current design is capable of 50% of full power operation only with natural circulation.

Steam Generators

The once-through steam generator is located within the reactor vessel in the annular space between the core support barrel and the reactor vessel inner wall. The secondary coolant is completely evaporated in a single pass through the steam generator. The steam generator tubes are divided into two groups that can be operated independently and the tubes in each group will be cross wound to reduce thermal twisting.

The steam generator consists of tube bundle, downcommer, two feed water and steam headers, shrouds to guide primary flow, and tube supporting structures. Approximately 1100 tubes are helically coiled in the effective heat transfer region and the effective coiling height is 3m. To assure equal steam quality from individual tubes, the length of tubes in the effective tube bundle is maintained as nearly equal as possible by varying the number of tube starts and helix angles in each tube column. The tube material is inconel 690 with 19 mm outer diameter through the whole tube length. The tube bundle is supported by eight perforated radial support plates, which transfer the load to the bottom support structure located on the supporting lug. Each tube can be accessed easily through the feed water and steam headers which are attached to the reactor vessel for in-service inspection and maintenance.

Pressurizer

The upper part of the RPV is filled with the mixture of nitrogen gas and steam providing a surface in the primary circuit where liquid and vapor are maintained in equilibrium under saturated condition. The pressure of the primary system is equal to the nitrogen partial pressure plus the saturated steam pressure corresponding to the core outlet temperature. Thus the reactor operates at its own operating pressure matched with the system status.

The nitrogen gas partial pressure of 2 Mpa is chosen to maintain subcooling at the core exit in order to avoid boiling in the hot channel during transients. The volume of gas space is large enough to prevent safety valves from opening during most severe design basis transient.

Reactor Coolant Pumps

The pumps are sealed type (i.e. glandless) canned motor pumps with added inertia to increase pump rundown time. With no shaft seals the small LOCA associated with seal failure in standard commercial designs is eliminated. With the primary water level lowered, they can be removed radially for servicing or replacement without having to remove the vessel closure head.

Control Element Drive Mechanism (CEDM)

The requirements for the CEDM are fine position control, lubrication by primary coolant, easy access to electrical parts, and high seismic resistivity. Soluble boron free reactivity control and load following operation over its full power range require a fine positioning capability of the control rod. The investigation of the failure frequency of operating CEDMs has revealed that the major source of trouble was in the electrical components requiring an easy access for maintenance. The long protruded part for the extension shaft stroke outside the reactor pressure vessel leads to high level of seismic excitation and reduces the margin to design stress limit. The current magnetic jack type CEDM used in the Korean Standard PWR is considered to be inadequate to meet the fine control requirements because it has only a step wise positioning. In addition the adoption of a self-pressurizer which occupies the upper plenum of the reactor vessel introduces difficulties in lubricating the moving part with the primary coolant since the latch mechanism would be located in the steam-gas region of the self pressurizer. Therefore, new concept of CEDM was proposed. The proposed design consists of position encoder, brushless DC servo motor, lift magnet coil, rare earth permanent magnet rotor, driving tube and split ball nut assembly. The rotor, driving tube and ball nut assembly are all connected into a single piece and lodged within the pressure housing which forms the pressure boundary. The encoder, DC servo motor, and lift magnetic coil are installed outside the pressure housing for easy maintenance. The use of a brushless DC servo motor with rare earth permanent magnet rotor allows a maintenance free operation of the motor. The fine control capability of the

CEDM is assured by the use of ball nut - lead screw mechanism and its lubrication with primary coolant is provided by placing this part below the water steam interface surface in the pressurizer. The ball nut assembly is of three pieces split type. Lift magnet located below the DC servo motor engages the ball nut to the lead screw by lifting the driving tube and the rotation of the rotor induces linear motion of the control element assembly up and down. The lead screw is part of extension shaft at the bottom of which a control element assembly is attached. When the scram signal is issued, the current supply to the lift magnet coil is cut off and the split ball nut releases the lead screw while dropping down by gravity and spring forces.

2.3 Engineered Safety Features

The safety concept is taking advantage from the intrinsic safety characteristics of integral reactor and pursuing passive safety principles common to most small and medium reactors. The fundamental safety characteristics are:

- Low ratio of power density to heat capacity resulting in a slow rise of fuel element temperature under accident conditions;
- A substantially negative moderator temperature coefficient resulting from no soluble boron usage generates beneficial effects on self-stabilization and limitation of reactor power;
- Integral Reactor Vessel eliminates large primary coolant pipes and thus large break loss of coolant accidents;
- Large Passive pressurizer significantly reduces pressure increase for decreased heat removal events;
- Large volume of primary coolant provides more thermal inertia and makes plant more forgiving;
- No RCP seals eliminates potential for seal failures, a concern during station blackout;
- The use of passive safety systems leads to directly to simplification in design since it eliminates the need for multiple redundant safety systems with their redundant safety grade power supplies.

2.3.1 Reactor Shut-Down System

The reactor trip in emergency is done by simultaneous insertion of the control rods into the core by gravity following the drive mechanism de-energization, which is actuated by trip signals from the automatic control system. In the case of failure to actuate the electromechanical protection system, the reactor shutdown is accomplished by the emergency boron injection system. Activation of the system is done by manually opening valves in the pipelines connecting the system to the reactor. Both shutdown systems ensures the reactor shutdown and its shutdown margin is sufficient enough to keep the cold clean reactor in a subcritical state.

2.3.2 Residual Heat Removal System

Normal decay heat removal when cooling down for maintenance and refueling, the steam generators with turbine bypass system are used and heat is rejected through the condensers. 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 system with steam generators are used to remove decay heat and heat is rejected through the component cooling system.

Should there be no ac power available, decay heat is removed by natural convection system which only requires battery power to operate the initiation valves and passive decay heat removal system which is composed of heat pipe and heat exchangers. The heat is ultimately rejected to atmosphere by natural convection of air flow. Thus, there is theoretically infinite time of heat removal without operator intervention. One of the advantages of heat pipe passive decay heat removal system is that this system is continuously operating during normal plant operation to remove the heat loss from reactor vessel through the wet thermal insulation.

2.3.3 Emergency Core Cooling System (ECCS)

The integrated primary system concept eliminates all large primary circuit pipe work, thus intrinsically eliminates large loss of coolant accidents. The largest pipe break in the primary circuit is the break of the connection pipe supplying chemical and volume control system(CVCS). To prevent siphoning off the reactor water inventory in the hypothetical event of a CVCS line break, open connections are made between the steam generators and pressurizer. Thus there is no possibility of rapid emptying of the reactor vessel requiring massive and early injection of ECCS water. Since reactor vessel is flooded all the time by the water in the safe guard vessel, there is no need for external emergency core make up. The safe guard vessel is sized to provide a minimum of 72 hours' heat removal without operator intervention.

2.3.4 Containment Overpressure Protection System

Since the maximum pipe break is small due to the integral nature, the pressurization rate of containment is slow. Energy released to the containment through the break point is removed using the steam injector driven containment spray system to prevent exceeding the containment design pressure. The steam injector is a simple, compact passive pump that is driven by supersonic steam jet condensation. The steam injector can operate even by atmospheric pressure steam. The steam from break point is supplied to the steam injector. The steam injector pumps up the water from a water storage tank on the ground to the spray nozzle located at the top of the containment.

3.0 Research and Development(R&D) Activities

To evaluate the characteristics of various passive safety concepts and provide the proper technical data for the conceptual design of the advanced integral reactor, the following R&D activities are being performed.

<u>Hexagonal Semi-Tight Lattice Fuel Assembly</u>

Numerical Analysis Technology is being development to analyze the hexagonal fuel assembly core with tight lattice. Some thermal-hydraulic experiments will be performed to understand the phenomena in the semi tight hexagonal lattice. The suitable thermal-hydraulic analytic model will be developed, especially thermal-hydraulic correlations, which are vital for the semi-tight hexagonal geometry. The developed model is incorporated into computer codes for the design and safety analysis.

<u>No Boron Core Concept</u>

The use of no soluble boron in the core design causes to utilize large amount of 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 conjuntion with the core design with hexagonal fuel assemblies are thus investigated in this R&D subject.

Natural Circulation Phenomena for Integral Reactor

To investigate thermo-hydraulic characteristics of primary circuit in natural circulation operation mode, an experimental test loop is being designed. An computer code is being developed to model the thermo-hydraulic behavior of the primary circuit.

Hydraulic Valve Application for PRHRS

To investigate the possibility of passive initiation of the isolation valve located inside Safe Guard Vessel, an experimental study is being performed.

Heat Pipe Application for PRHRS

A separate type heat pipe is experimentally studied for PRHRS. A computer code is developed to model the thermo-hydraulic behavior of the heat pipe.

Helically Coiled Tube Once Through Steam Generator

The thermal hydraulic design and performance analysis computer code, ONCESG, for a once through steam generator is developed. An experimental study is being performed to generate the heat transfer correlation and pressure drop correlation of the helically coiled tube once through steam generator.

Critical Heat Flux Test

An experimental test facility is being constructed to study critical heat flux and pressure drop correlation for the tight latticed hexagonal fuel assembly.

Steam Injector Application for PCCS

An experimental study is being conducted on a steam injector driven passive containment cooling system. A computer code is developed to model the thermo-hydraulic behavior of the steam injector.

Wet Thermal Insulation

An experimental investigation is being conducted for material selection and performance test for the wet thermal insulation system.

Fluidic Diode Application for Passive Pressurizer Spray System

An experimental study is being conducted on the fluidic diode device for passive pressurizer spray system. A computer code is developed to model the thermo-hydraulic behavior of the fluidic diode.

4.0 Summary

An advanced integral reactor is currently under conceptual development at KAERI. Main features of the reactor are summarized as follows;

Integrated Primary System

This feature implies that all primary components - core, steam generators, pumps, pressurizer - are contained in a single pressure vessel. This arrangement allows the complete removal of all primary circuit pipework, thereby the elimination of the large break LOCA.

Enhanced Core Design

The core is designed to operate without the need for reactivity control using soluble boron during the whole power range. The core is designed to enhance reactor operating safety margin by specifying a very low core power density and linear heat generation rate. The core has long refueling cycle(18 months or longer) to improve plant availability.

Self-Pressurized Primary System

The free volume above the reactor water level is used as steam/gas pressurizer. The operating pressure is determined by initial gas pressure and the steam pressure corresponding to the core exit temperature. Thus the reactor operates at its own operating pressure matched with the system status.

Enhanced Primary Circuit Natural Circulation Capability

The primary circuit natural circulation capability is enhanced by the very low head loss of the integrated primary system. The difference in temperature between cold and hot water together with the difference in height between steam generator and the core produces driving force to circulate primary coolant. The required pump capacity to circulate primary coolant is reduced by proper selection of these parameters.

Passive Engineered Safety Features

The design provides for all the required safety functions following plant transient or accidents through passive safety features. The residual heat removal, inventory control for the reactor coolant system, emergency core cooling, and containment heat removal are all passive and integrated within the containment system.