# A STUDY OF A NUCLEAR HYDROGEN PRODUCTION DEMONSTRATION PLANT

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The current energy supply system is burdened by environmental and supply problems. The concept of a hydrogen economy has been actively discussed worldwide. KAERI has set up a plan to demonstrate massive production of hydrogen using a VHTR by the early 2020s. The technological gap to meet this goal was identified during the past few years. The hydrogen production process, a process heat exchanger, the efficiency of an I/S thermochemical cycle, the manufacturing of components, the analysis tools of VHTR, and a coated particle fuel are key areas that require urgent development. Candidate NHDD plant designs based on a 200 MWth VHTR core and I/S thermochemical process have been studied and some of analysis results are presented in this paper.

**KEYWORDS** :VHTR, Hydrogen Production, I/S Thermochemical Process, TRISO Fuel, NHDD

## 1. INTRODUCTION

Significant changes in climates are underway on a global scale, and it is widely accepted that human activities are the main cause.[1] With the aim of attaining more economical fuel production while preserving the current ecological environment, a hydrogen economy has been widely discussed.[2] In this respect, hydrogen should be produced from water not from fossil fuels, as the latter will become increasingly scarce. Development of a technology for the production of hydrogen from water using nuclear energy has been actively discussed.[3] There is greater need for nuclear hydrogen in nations such as Korea where population density is very high and domestic energy resources are nearly absent.

KAERI has established a plan to demonstrate massive production of hydrogen using a very high temperature reactor (VHTR) by the early 2020s. The target is to achieve economic competitiveness against the future price of natural gas steam reforming considering hydrogen distribution costs. Another target is to obtain an operating license for a Nuclear Hydrogen Development and Demonstration (NHDD) reactor for the follow on commercial plants using the best available international and domestic technologies in the early 2010s.

## 2. GAPS IN CURRENT TECHNOLOGY

### 2.1 Hydrogen Production Process

There are several technological gaps on the path to realizing hydrogen production using nuclear technology, as such a production paradigm has yet to be physically demonstrated. To achieve high efficiency, a high temperature reactor or a very high temperature reactor (VHTR) should be used as a heat energy source. Among various methods of producing hydrogen from water, the sulphur-iodine thermo-chemical cycle (S/I cycle), the high temperature electrolysis of steam (HTE), and the hybrid sulphur cycle (HyS cycle) are actively being developed for VHTR coupling.

The S/I cycle was studied at General Atomics[4] and demonstrated as a closed cycle[5] at JAEA. However, realistic demonstration of the process under high pressure has not been accomplished to date. Unlike bench scale experiments where transparent quartz or pyrex wares can be used, in a high pressure environment, most equipment should be made of either metal or ceramics so as to withstand high pressure differences during normal operation and transients conditions. Thermochemical reaction data including vapour liquid equilibrium (VLE) at high pressure and high temperature are not sufficient for design, analysis,

and evaluation of the plant scale application. One of the main issues is to effectively separate hydrogen from hydrogen iodide (HI). In this regard, extractive distillation, reactive distillation, and permselective membrane separation combined with electro-electrodialysis have been suggested by different research groups. Lastly, prediction of hydrogen production efficiency is largely variant between an optimistic value of 51% to a pessimistic value of 33% at 850 degrees-C .[6]

High temperature electrolysis of steam is a rather simple process with relatively good efficiency.[7] However, the generic feature of electrolysis using narrow gap limits the benefit of economical production by scale up. The main focus of this research is to reduce over potential for electrolysis, improve mechanical stability during thermal transient, and improve the life time of the solid membrane and components.[8] The process relies on the development of solid oxide fuel cell (SOFC) technology, where a great deal of research and effort has already been concentrated. Depending on technological breakthroughs or development of the SOFC market for mass production, the economics of HTE may be improved.

A hybrid sulphur cycle is a combination of the sulphur decomposition section of the sulphur-iodine thermochemical cycle and the electrolysis process to cycle sulphur dioxide into sulphuric acid.[9] The HyS cycle is simpler than the I/S cycle. The decomposition process of sulphuric acid has common problem of the process heat exchanger as in I/S thermochemical cycle . The electrolysis of sulphur dioxide takes place in a very corrosive environment. One proposed solution that entails using a proton exchange membrane is expected to reduce the material problem and may allow for active development of PEM fuel cell. Reduction of the required over potential with increased electric current is a key challenge for the HyS cycle.

## 2.2 Process Heat Exchanger

A process heat exchanger (PHE) is a key component to transfer the heat energy from the nuclear reactor to the chemical reactions . The PHE is composed of helium gas channels and sulphuric oxides gas channels. A thin heat transfer medium of a PHE is required in order to minimize the temperature drop across the PHE for more efficient hydrogen production. However, the PHE will suffer the extreme environments of high corrosion, high temperature, and high differential pressure. Several concepts were proposed for PHE design, ranging from conventional helical type to plate fin type with ceramics or refractory metal. A metallic heat exchanger has a short lifetime due to the complex interaction between stress and corrosion. A ceramic heat exchanger with strong corrosion resistance is not easily manufactured and has inadequate thermal shock resistance because of its low toughness. A heat exchanger is currently being developed at KAERI to overcome both the short lifetime and the difficulty in manufacturing. The base material of the PHE is a high temperature super alloy for easier manufacturing compared to ceramic material. The contact surface with sulphuric gas and/or steam is coated with a corrosion resistant material to enhance corrosion resistance.[10]

Meanwhile, delamination along the interface can result from a stepwise change of the material properties between the base metal and the coated material. Ion beam mixing technology is expected to decrease the problem of delamination by smoothing the change of material properties at the interface.[11] A coupon specimen scale test has been performed to evaluate the effect of the developed coating technology with ion beam mixing on advanced corrosion resistance. As shown in Figure 1, a sample without ion bombardment shows flakes at the edge of the film after electrolytic etching, implying that corrosion initiates from the uncoated substrate to the film and then penetrates the film. However, such corrosion is not found in the ion bombarded sample after etching under the same conditions.

A hybrid concept is developed to meet the design requirements of the proposed PHE.[12] The hot helium gas channel is a compact semicircular shape similar to a printed circuit heat exchanger designed to withstand the high pressure difference between loops. The sulphuric acid gas channel is a plate fin shape that has enough space to install and replace the catalysts for sulphur trioxide decomposition.

There are numerous potential problems that should be investigated in order to apply the developed concepts on PHE to an actual reactor coupling. The small scale gas loop of 10 kW scale displayed in Figure 2 is being constructed to investigate the feasibility of the developed PHE concept.[13] The performance and structural integrity of the developed PHE will be studied in the test loop.

## 2.3 Efficiency of I/S Thermochemical Cycle

Based on the thermodynamic information of chemical reactions (1) to (3), a schematic chemical reaction flow diagram of the I/S process coupled to a very high temperature reactor (VHTR) can be drawn as shown in Figure 3. [14]

$$2H_2O + I_2 + SO_2 = 2HI + H_2SO_4 \quad (<120℃ \text{ exothermic reaction}) \qquad (1)$$

$$2HI = H_2 + I_2 \quad (400\sim500℃ \text{ endothermic reaction}) \qquad (2)$$

$$H_2SO_4 = H_2O + SO_2 + 0.5O_2 \quad (>850℃ \text{ endothermic reaction}) \qquad (3)$$

The I/S process coupled to the VHTR is composed of an intermediate heat exchanger, a Bunsen reaction section (section 1), a sulfuric acid concentration and decomposition section (section 2), and a hydrogen iodide concentration

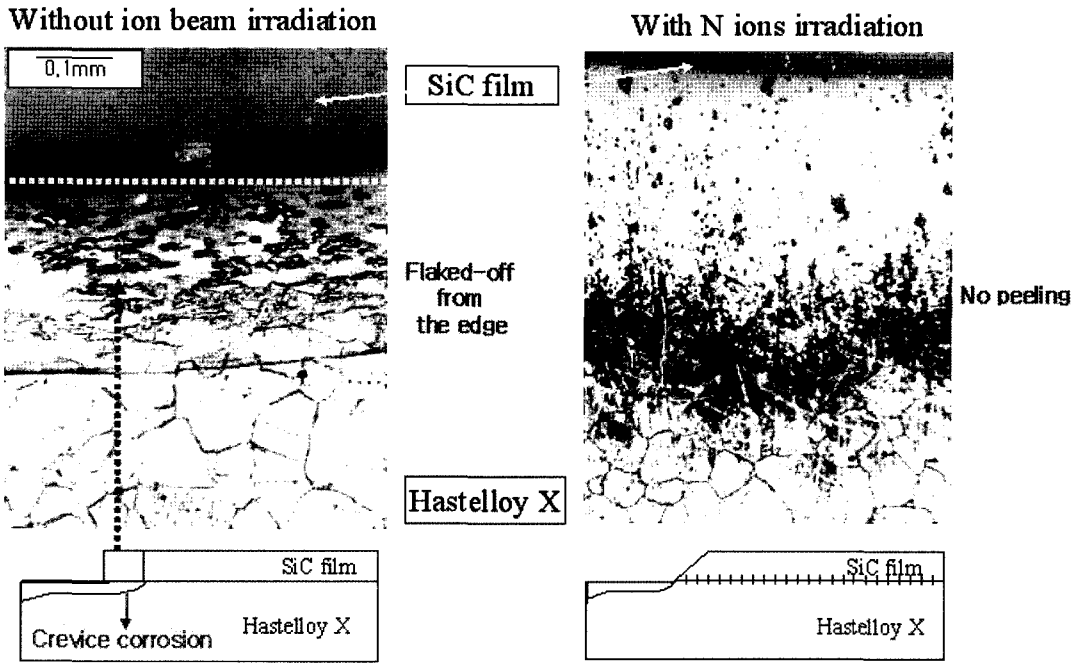



Fig. 1. Effect of Ion Beam Mixing on Corrosion Resistance

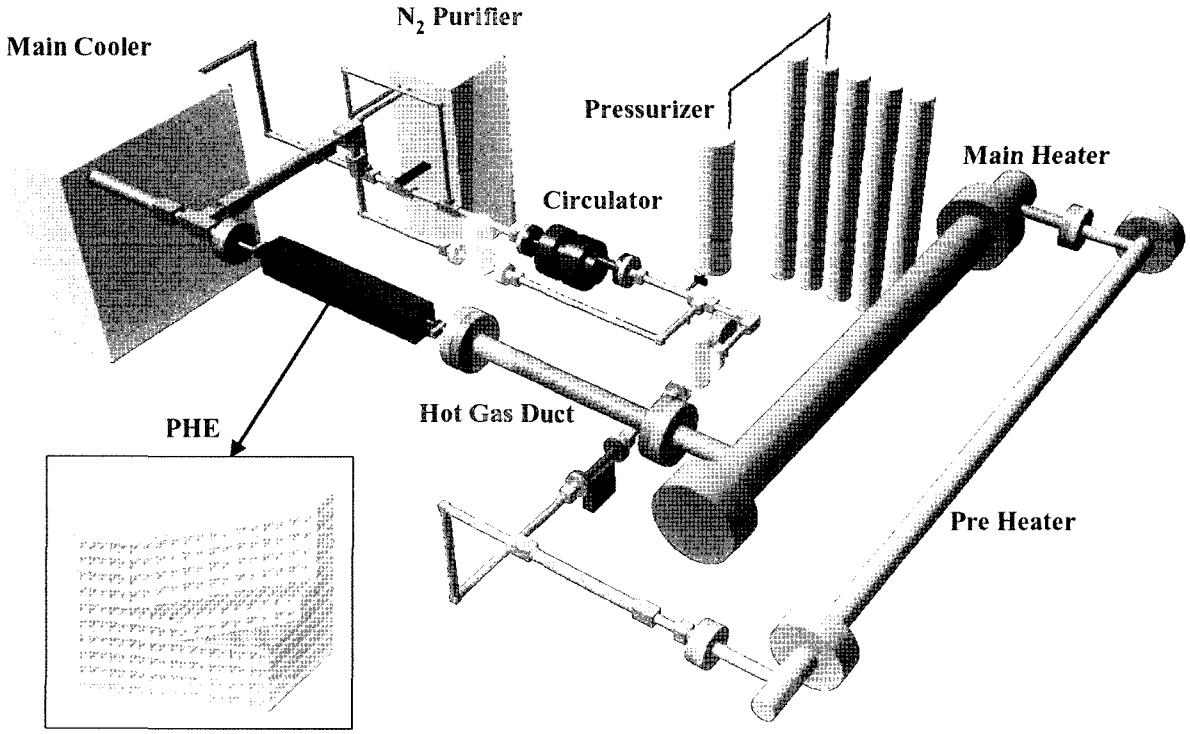



Fig. 2. Layout of Gas Loop and Process Heat Exchanger

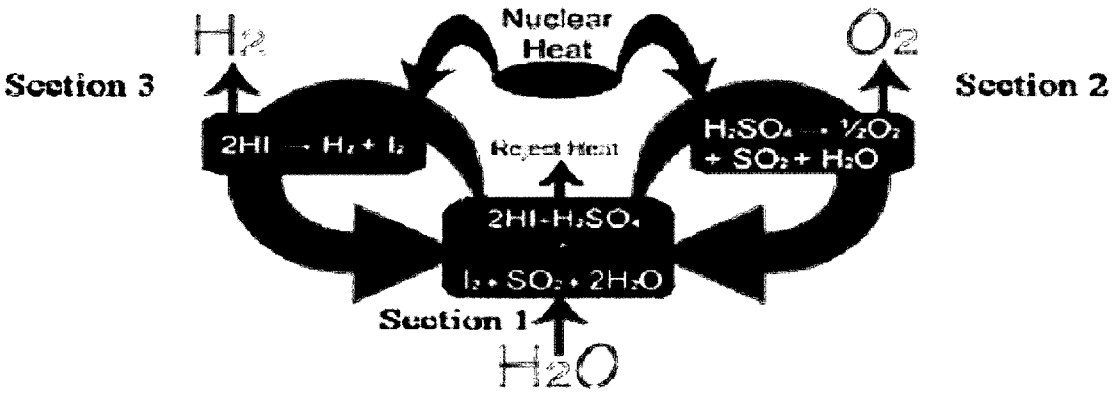



Fig. 3. Schematic Chemical Reaction Flow Diagram of I/S Process

and decomposition section (section 3).

The required operational temperature and pressure of the I/S process are as high as 850°C and 2.2 MPa, respectively. [15] The sulfur trioxide decomposer requires a high operation temperature of 850°C. By the general theory of gas phase decomposition, its operation pressure should be maintained as low as possible even when the heat exchanging capability is increased by increasing the pressure. Based on these general rules, we can establish a preliminary I/S process to evaluate its thermal efficiency. Of course it should be continuously modified to obtain the optimized thermal pathway for the process through evaluations.

In order to estimate the thermal efficiency of nuclear hydrogen production by the I/S cycle, under the assumption that the heat loss at a pipe and each piece of equipment to the surrounding environment is negligible, we have calculated the heat required to produce hydrogen equivalent to a 200 MWth VHTR. The thermal efficiency is then defined by the following equation.

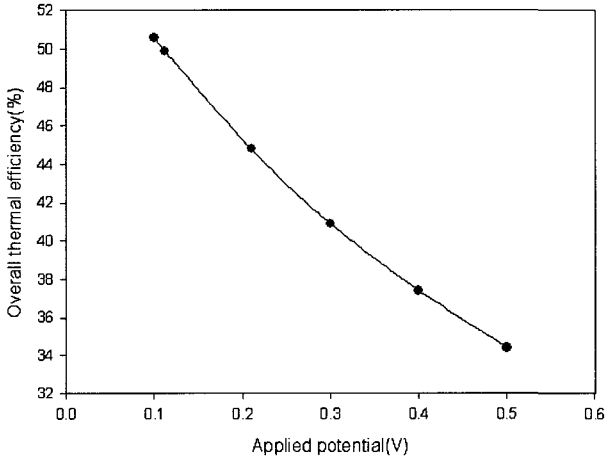



Fig. 4. Overall Thermal Efficiency VS. Applied Potential Required in Electrodialysis

$$\eta_{eff.} = HHV/(Q_{thermal} + Q_{electrical}) \quad (4)$$

where HHV is the high energy value of hydrogen, and $Q_{thermal}$ and $Q_{electrical}$ are the thermal and electrical energies consumed to produce hydrogen. Figure 4 shows the typical calculation results of the thermal efficiency of the I/S process as a function of the performance of an electrodialysis reactor, one of the key components in the I/S process. [16]

The I/S cycle is composed of three sections, sulfuric acid decomposition, Bunsen reaction, and HI decomposition section. The sulfuric acid decomposition section produces oxygen and sulfur dioxide from sulfuric acid. The Bunsen reaction section produces HI and sulfuric acid from iodine and sulfur dioxide with water. The HI decomposition section produces hydrogen and iodine, and then separates hydrogen from iodine. The overall result is the decomposition of water into hydrogen and oxygen.

## 2.4 Manufacturing Aspects for Reactor Vessel and Internals

One of the physical factors that limits the power level of the NHDD reactor is the dimensions of the reactor pressure vessel. Domestic nuclear equipment suppliers can produce a ring forged reactor pressure vessel with up to 6.5m diameter. Although a larger diameter pressure vessel can be manufactured by using a longitudinal welding, longitudinal welding is not recommended due to the possibility of failure in the welded region after long neutron irradiation and corrosion in the helium environment.

As for the material of the reactor pressure vessel, two potential candidates are a SA508 grade3 vessel, which is widely used in conventional pressurized water reactors, and a modified 9Cr-1Mo vessel, which has been developed for high temperature applications. The SA508 has benefits in that it is a qualified ASME material and there is an abundance of experience with it. However, its temperature is limited to 371℃ and conditionally up to 538℃. Therefore, a cooled vessel design concept needs to be applied in order to use the SA508 vessel. A modified 9Cr-1Mo is also endorsed in ASME Subsection NH as class 1 components at an elevated temperature. However, more design data is needed for application at elevated temperature over its lifetime. Manufacturing capabilities including large ring forging, welding of thick structures, and heat treatment are not well established for the modified 9Cr-1Mo. [17]

Alloy 800H or Hastelloy-X, which have been verified for high temperature application, can be used for most of the metallic reactor internal structures where mechanical strength is of little concern. [18] For core applications, neutron activation of cobalt and impurities should be carefully studied.

For extremely high temperature structural components such as control rods or control rod guide tubes, metallic material should be avoided. Ceramics or a carbon fiber reinforced carbon composites are a potential solution. Material properties of ceramics are suitable for such application. However, there are a number of technical difficulties for application as reactor internal structural components. The difficulties include low machinability due to low toughness, insufficient bonding methodology, the absence of an internationally acceptable design code, and lack of long term and high temperature irradiation data.

## 2.5 Analysis Tools of VHTR

The safe and reliable operation of a nuclear reactor system should be demonstrated using design and safety analysis tools and methodology, which is the most important process in licensing applications. However, most existing computer code systems were developed quite long ago, when the basic understanding of theoretical and experimental phenomena and computing power were poor. As such, they may not fulfill the accuracy and licensing requirements demanded by strengthened current regulations and also may fail to resolve key technical issues raised in modern VHTR designs. Thus, it is necessary to improve or develop design and safety code systems by incorporating advanced models and techniques.

Furthermore, in the VHTR there is no means to directly measure the in-core power or temperature distributions required in the safety demonstration due to its high temperature. Therefore, reliable prediction using a computer code system is of utmost importance. Temperature measurement conducted in an AVR core in 1974 by using graphite pebbles with melting wires revealed that local hotspots are higher than the predicted values. [19] This clearly shows the need to enhance the accuracy and reliability of the code systems.

In addition, there are generic modeling issues related to the design and safety analysis of a VHTR. The core is basically doubly heterogeneous, composed of TRISO fuel particles in the fuel block or pebbles. In the PBR core, fuel pebbles flow along the core during normal operation. In view of therm-fluid and safety, complex multi-dimensional phenomena should be considered: flow distribution in the inlet and upper plenums, flow and temperature distribution in the pebble core, including flow streaming near the reflector wall, core bypass flow through graphite structural gaps, flow mixing and streaking in the outlet plenum, hot plume rise in the upper plenum during an accident, etc. [20, 21] In the case of an air or water ingress event into the reactor system, graphite chemically reacts with the

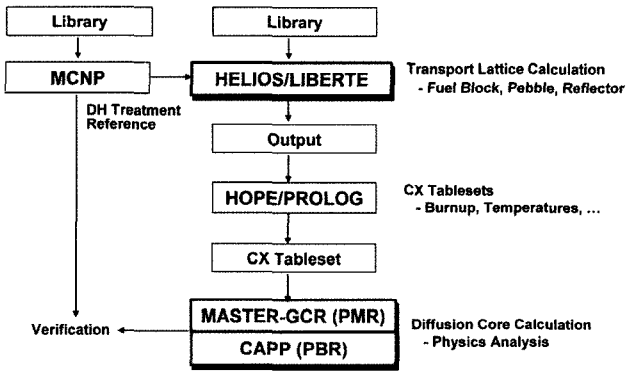



Fig. 5. Nuclear Design Code System [24, 25]

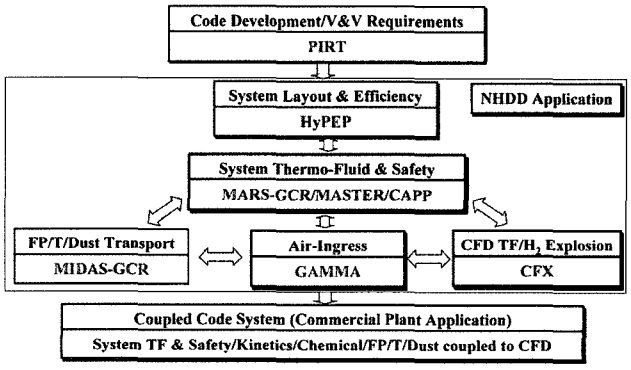



Fig. 6. Thermo-Fluid and Safety Analysis Code System [26, 27]

former to produce heat, and may weaken the structural strength. Special concerns of an NHDD plant that is dedicated to hydrogen production are the intermediate loop performance, the tritium contamination of the hydrogen product, and explosion of hydrogen in storage.

In this context, KAERI is developing key code systems for nuclear design, [22, 23] thermo-fluid design, and system safety analysis. In order to meet recent regulations and to resolve emerging safety issues, the code systems are developed to maximally incorporate advanced technologies. Figures 1 and 2 show the code systems for the nuclear design and the thermo-fluid and safety analysis, respectively.

## 2.6 Coated Particle Fuel

The inherent safety of a high temperature gas-cooled reactor (HTGR) lies, to a large extent, on the characteristic of the fuel used: As a fuel basis, the HTGR uses coated fuel particles, which are dispersed in a graphite matrix to form fuel elements, either in the form of so-called pebbles, or in a compact, the latter being inserted in haexagonal graphite block to form a fuel assembly. In the current operating HTGR designs, TRISO (TRI-ISOtropic)-coated fuel particles are used [28, 29] in HTR-10 in China and HTTR in Japan. The TRISO-coated fuel particle consists of a kernel microsphere of oxide or an oxycarbide fuel material (e.g., $UO_2$) and coating layers of porous buffer pyrolytic carbon (PyC), inner dense PyC (IPyC), silicon carbide (SIC), and outer dense PyC (OPyC). The role of these coating layers is to maintain appropriate mechanical integrity of such a structure during manufacturing and in-reactor service, and to retain fission products within the particle. The SiC coating layer lends mechanical strength to the particle and acts as a barrier to the diffusion of metallic fission products that diffuse easily through the IPyC coating layer. Typical microscopic features of the coated particle fuel are illustrated in Figure 7. The technologies for manufacturing of the kernel are based on a wet chemical gelation method while coating is based on a fluidized-bed chemical vapor deposition method. These approaches have matured since the early 1980s. Typical schematic flow diagrams are shown elsewhere. [30, 31]

The performance of SiC-TRISO coated particle fuels has been and is being tested and verified through a wealth of studies related to manufacturing technology and irradiation testing, in conjunction with AVR (Arbeitsgemeinschaft VersuchsReaktor) in Germany, the Fort St. Vrain HTGR in the US, HTR-10 in China, and HTTR in Japan. This large body of work, initiated by the Dragon Project, has been led by the Juelich Research Center in Germany. China and Japan, utilizing the same technology in principle, have benefited from these efforts in establishing their own technologies.

The difference in performance-related behavior of coated particle fuels developed by Germany and the US has been analyzed by D. Petti et al., [32] who indicated that potential variation in coated particle fuel failure arises from differences in kernel materials (UO2 vs. UCO) and the coating process.

Although SiC has good properties, it has a tendency to lose mechanical integrity at higher temperatures, i.e., above 1700℃ (1973 K), by thermal dissociation and transformation of $\beta$-SIC to $\alpha$-SiC. [33, 34] Consequently, the fuel temperatures are limited to well below 1700℃ during design-basis accidents in the current HTGR designs. Zirconium carbide (ZrC), known as a refractory and chemically stable compound and currently utilized as a high temperature ceramic material, is one of the candidates to replace the SiC coating layer for higher temperature application of the TRISO-coated fuel particles. The results of early irradiation experiments of ZrC-coated fuel particles were indicative . [35, 36, 37] It has been demonstrated

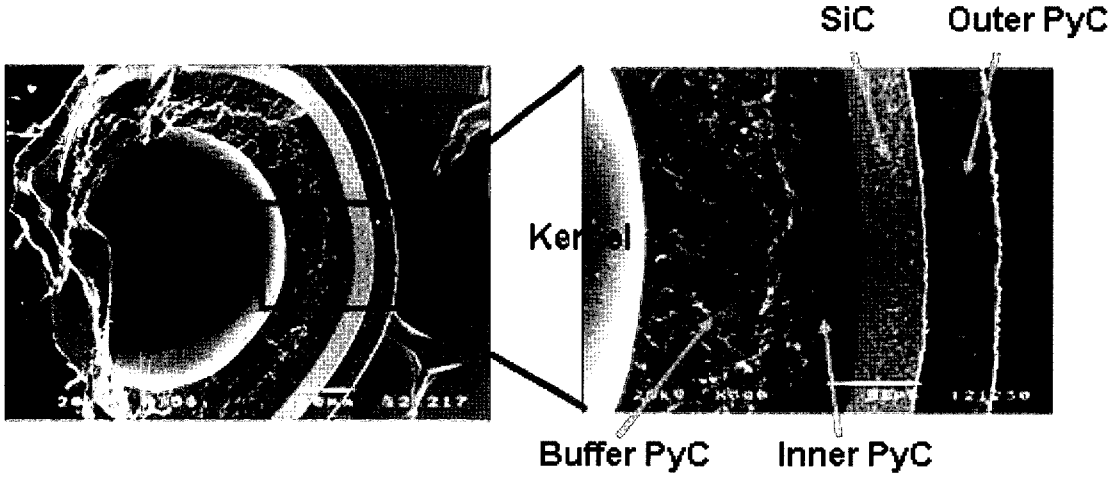



Fig. 7. Typical Microstructure of Coated Particle Fuel Showing Kernel and Coating Layers

that ZrC-TRISO coated fuel particles have much higher temperature stability than normal SiC-TRISO coated fuel particles. ZrC coating layers also have higher resistance to chemical attack by fission products such as palladium than SiC coating layers. [38] However, research and results on the behavior of fission products in the ZrC coating layer are still limited. [39] Thus, it is apparent that further R&D is needed for conclusive evaluation of the fission product retention behavior of the ZrC coating layer during irradiation in VHTR conditions.

## 3. A CANDIDATE DESIGN OF NHDD PLANT

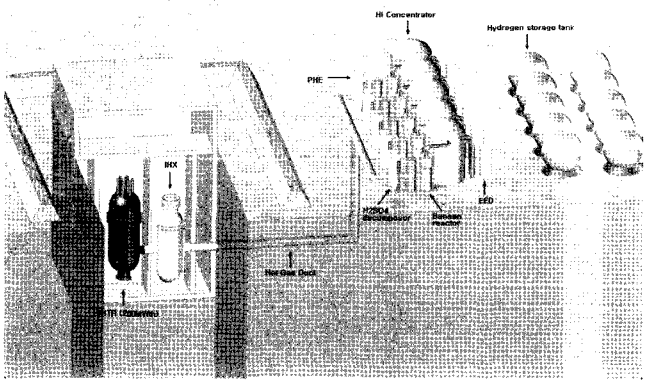



Fig. 8. Layout of Nuclear Hydrogen Demonstration Plant

### 3.1 General Concept

Construction of a demonstration reactor system is necessary for the demonstration of safe and economic production of nuclear hydrogen, before commercial deployment of nuclear energy technology. In comparison with a PWR plant system, there are numerous items that must be demonstrated, such as the coupling between the nuclear reactor and chemical plant, natural decay heat removal sufficient to guarantee fuel integrity in any accidental shutdown condition, radioactivity release to the environment or to produced hydrogen, and so on.

To mitigate licensing concerns in the near term, a forged vessel is selected for the VHTR. A forged vessel concept can eliminate concerns related to whole vessel break up; however, cooling of the conventional SA508 vessel will reduce overall thermal efficiency. On the other hand, it can provide a clean helium layer where any abnormal increase of radioactivity can be monitored, thus preventing further release to environment.

As displayed in Figure 8, an underground reactor installment concept is employed. An underground structure allows conduction to the ground even in a loss of passive cooling event without overheating the fuel above the safety limit, thus providing a lower rate of damage than a surface based structure. An underground concept can protect the reactor from possible explosion of hydrogen storage tanks and accumulated leakage in the chemical plant.

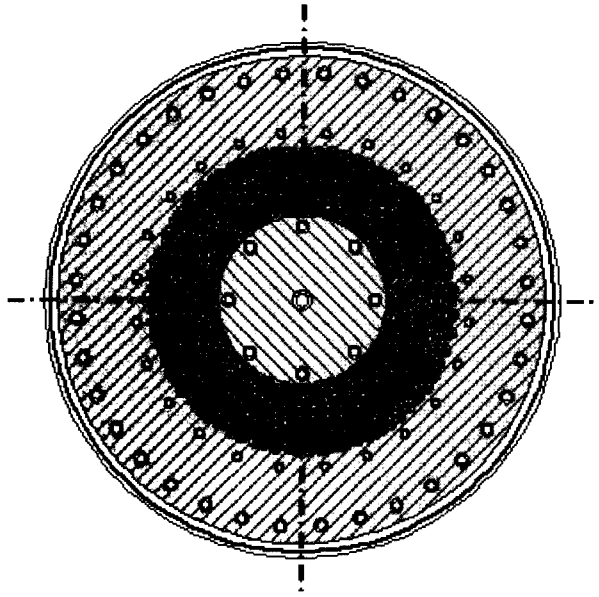
The chemical plant will require more scheduled shutdowns than a nuclear reactor, for replacing catalysts and components such as valves and seals. As such, the hydrogen production plant is divided into 5 trains so that the VHTR core can remain in operation while one of the hydrogen production trains is shutdown for maintenance. This concept requires a manifold of hot gas ducts to distribute heat from the reactor to the trains.

### 3.2 Candidate Reactor Designs

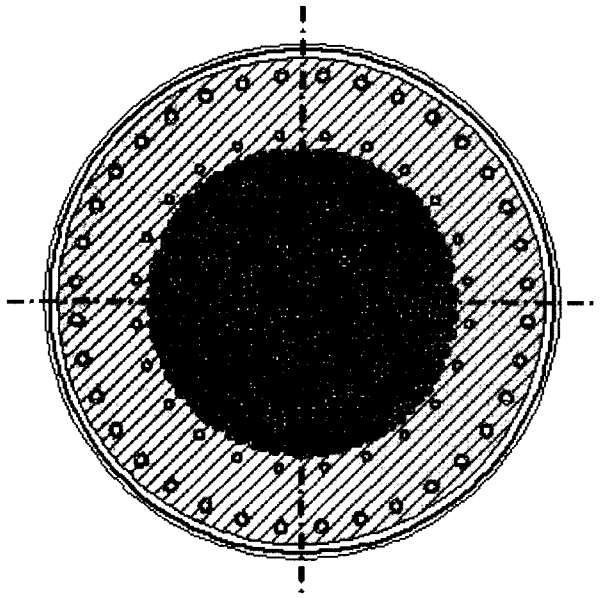
Four candidate core designs are explored for the 200MWth demonstration plant, two of which are prismatic modular reactor cores (PMR) and the others are pebble-bed modular reactor cores (PBR). The PMR and PBR candidates are scaled down from their corresponding references, that is, the 600MWth GT-MHR core [40] and the 400MWth PBMR core, [41] respectively. Here, the 200MWth power is selected, since it is of proper size for application to an oil-refinery plant as well as a hydrogen production plant.

Among the two candidate PBR cores, one is an annular core with an inner reflector (PBR200-CD1), and the other is a cylinder core without an inner reflector (PBR200-CD2). Figure 9 shows a cross-section view at the vessel midplane of the PBR cores. The PBR reactors are assumed to use the same sphere fuels as used in the PBMR reactor. Each pebble has a 6cm diameter and nominally contains 15,000 UO2 TRISO coated micro-spheres imbedded in a graphite matrix. Each pebble contains 9g of U, and fuel enrichment is 9.76w/o for the equilibrium cycle. The active height of the core is 8.73m for the two candidate cores. The pebbles are piled randomly in the core with a volumetric filling fraction of 0.61. The thickness of the outer graphite reflector is 90cm. The average power density of the core of PBR200-CD1 is 4.79w/cc, while that of PBR200-CD2 is 3.56w/cc. The power density of PBR200-CD2 is lower, since it does not have an inner reflector, which limits the maximum fuel temperature during reactor accidents. The fueling scheme employed is a continuous on-line multi-pass method similar to the designs used in the PBMR reactor. The pebbles are added to the top of the reactor while used fuel pebbles are removed at the bottom to keep the reactor at full power. On average, each fuel pebble is assumed to make six passes through the reactor before finally being discharged to the spent fuel storage tanks.

The two PMR candidate cores are annular with an inner reflector, as shown in Figure 10 (PMR200-CD1 and PMR200-CD2). PMR200-CD1 is composed of 48 fuel

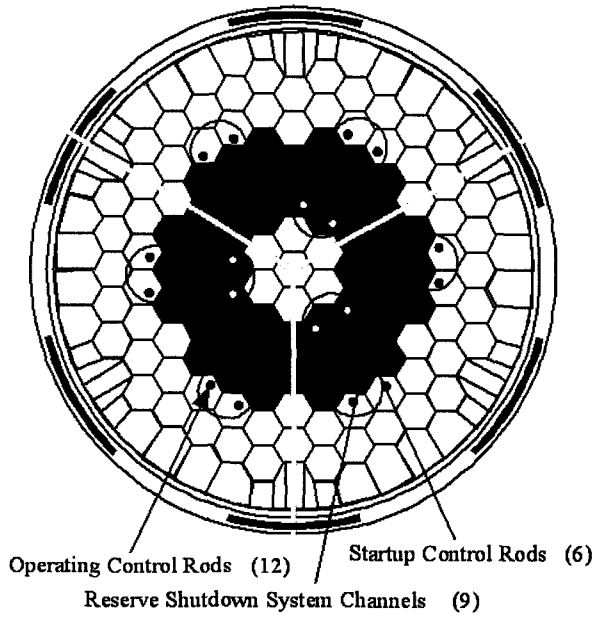


(a) PBR200-CD1

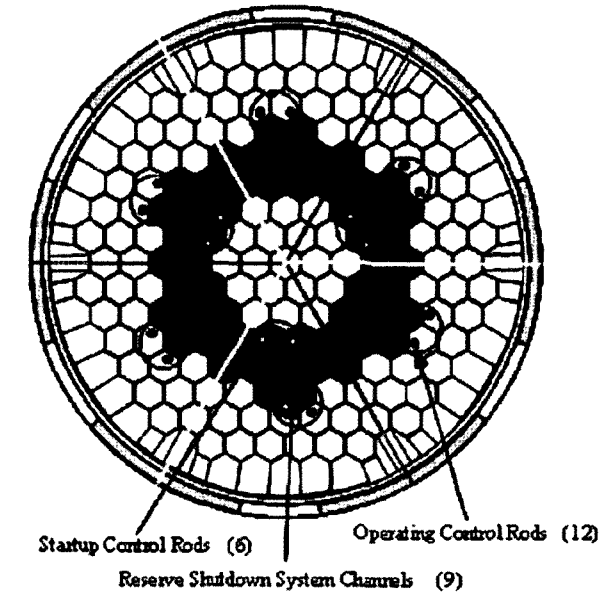


(b) PBR200-CD2

Fig. 9. Plane View of PBR200 Candidate Cores




(a) PMR200-CD1




(b) PMR200-CD2

Fig. 10. Plane View of PMR200 Candidate Cores

columns, each of which consists of 7 fuel blocks stacked axially, while PMR200-CD2 is composed of 54 fuel columns consisting of 6 fuel blocks in each column. The hexahedral fuel block has essentially the same dimensions of the GT-MHR fuel block with 79.3 cm height and 36 cm across flat size . It contains a regular pattern of fuel and helium holes with a ratio of two fuel holes per helium hole. In a fuel hole, 15 fuel compacts are loaded, each of which has a diameter of 1.245 cm and a height of 4.93 cm and contains UO2 TRISOs. There are three types of blocks containing fuel in the active core (i.e., standard blocks, reserved shutdown blocks, and control blocks). The reserved shutdown and control blocks differ from the standard blocks because they contain eccentrically located large diameter channels of different sizes (9.53 cm and 10.16 cm) for traveling control rods and reserved

shutdown absorber balls. The numbers of the reserved shutdown and control blocks are 9 and 6, respectively, for both PMR200-CD1 and PMR200-CD2. The average power densities of PMR200-CD1 and PMR200-CD2 are 6.69 w/cc and 6.94 w/cc, respectively. These values are comparable to that of the GT-MHR core. The two PMR cores are enclosed by graphite reflectors of almost the same thickness as the GT-MHR core except the inner reflector. The fuel reload scheme assumed is a two batch reload scheme with an 18 month cycle length.

Preliminary core analyses for all the PMR and PBR candidate core designs were performed using VSOP94 code system, [42] the results of which were reported elsewhere. [43, 44] The core characteristic parameters including the power and the temperature coefficients were evaluated for the equilibrium core conditions. Although the VSOP94 code system was developed for the analysis of pebble-bed reactors, we found that it could be extended to the analysis of prismatic modular reactor cores. From the results of the temperature coefficients analysis, it is shown that all the temperature coefficients of all the candidate cores, including the isothermal temperature coefficients shown in Figure 11, are negative for all operating conditions at equilibrium core conditions.

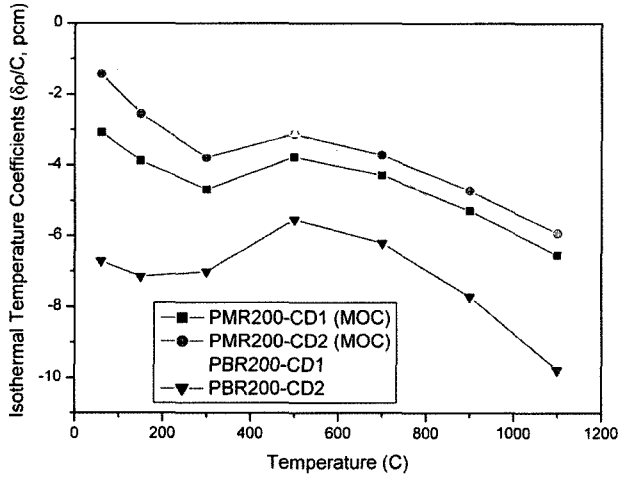



Fig. 11. Isothermal Temperature Coefficients of 200MWth Candidate Cores

## 3.3 Safety Analysis

Among the four candidate cores for the 200MWth demonstration plant, the safety of two cores (PMR200-CD1 and PBR200-CD1) was investigated for two limiting events; high pressure conduction cooldown (HPCC) and low pressure conduction cooldown (LPCC) accidents. The GAMMA code, a multi-dimensional and multi-component mixture analysis code, [45] was used for the transient simulation.

Table 1 summarizes the system parameters and predicted initial conditions for the transient analysis. In the safety analysis, core power distributions obtained from the nuclear design were used. For a consistent analysis, an air-cooled reactor cavity cooling system (RCCS) was assumed for both designs, while the vessel designs were scaled-down from the corresponding reference designs, GT-MHR and PBMR.

**Table 1**. System Parameters for Safety Analysis

| Parameter | PMR200 | | PBR200 | |
|---|---|---|---|---|
| Thermal power, MWth | 200 | | 200 | |
| Equivalent active core inner/outer radius, cm | 50/140 | | 80/147 | |
| Thickness of outer reflector, cm | 100 | | 90 | |
| Effective core height, cm | 555.1 | | 873 | |
| Average power density, W/cc | 6.69 | | 4.79 | |
| Parameter | Design | Calculated | Design | Calculated |
| Temperature, ℃ | | | | |
| RCS inlet/outlet | 490/950 | 490/950 | 490/950 | 490/950 |
| RCCS inlet/outlet | 43/217 | 43/217.5 | | 43/169 |
| Peak fuel temp. | - | 1165.1 | | 1156 |
| Pressure, MPa | | | | |
| RCS inlet/outlet | 7.0/- | 7.0/6.95 | 7.0/- | 7.0/6.8744 |
| Flow rate, kg/s | | | | |
| RCS flow | 83 | 82.985 | 83.3 | 83.2 |
| RCCS flow | - | 10.34 | - | 9.5 |
| CR bypass flow | - | 9.5 | - | 7.9 |
| Heat loss to RCCS, MW | - | 1.85 | | 1.22 |

### 3.3.1 HPCC Accident

The HPCC accident is initiated by loss of forced cooling caused by a primary circulator trip. After the circulator trip, it is assumed that the reactor scrams immediately and the

coolant flow decreases linearly in 60 seconds. The system pressure also decreases linearly, from 70 bar to 50.3 bar in 8 hours.

During an HPCC accident, the core heats up by a power-cooling-mismatch, and then starts to cooldown by continued conduction and radiation cooling to the RCCS. Natural circulation cooling established within the core facilitates heat removal from the core to the RCCS. In both designs, the peak fuel temperatures were far below the safety limit of 1600℃. On the other hand, the reactor pressure vessel temperature is a greater safety concern due to potential creep deformation under a long period of high temperature and high pressure conditions.

Figure 12 shows the calculated maximum RPV temperatures and compares them with the results of reference PMR600 and PBR400 (GT-MHR and PBMR with a helium exit temperature of 950℃). [46, 47] Since the scaled-down 200MWth candidate designs have larger power-to-surface ratios for radiation, the maximum RPV temperatures are below the limit. [48] Deviation of the transient response during the early phase is due to the core barrel cooling system in the PBR.

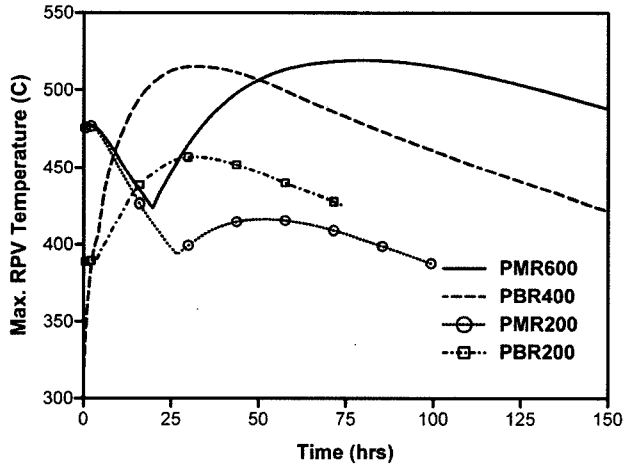



Fig. 12. Maximum RPV Temperatures During HPCC Accident

### 3.3.2 LPCC Accident

The LPCC accident is initiated by a loss of coolant event caused by a break of a connection pipe. Immediately following the break, the coolant is discharged to the reactor cavity. It is assumed that the reactor scrams immediately and the coolant flow and the system pressure decrease to zero flow and atmospheric pressure, respectively, in 10 seconds.

In the LPCC event, the peak fuel temperature is of greater concern than the peak RPV temperature, since core heat removal takes place only by conduction and radiation at low pressure, where the mechanical stress to the RPV is negligible. Figure 13 shows the calculated maximum fuel temperatures and compares them with the results of reference PMR600 and PBR400. [49] Compared to PMR600 and PBR400, the maximum fuel temperatures of the candidate 200MWth reactors are far below the fuel failure limit (1600℃) and the temperature responses are faster due to the larger power-to-surface ratio for radiation.

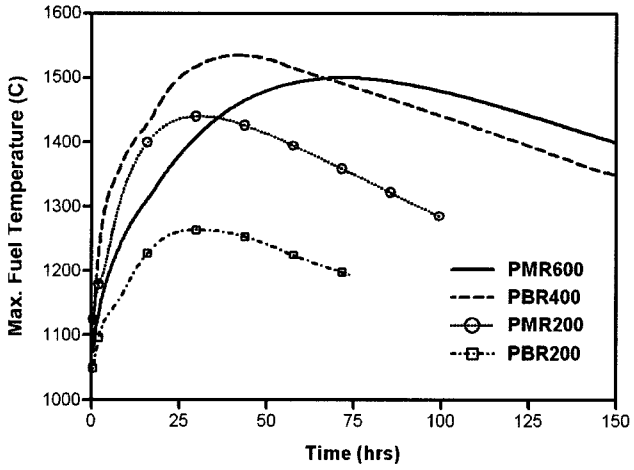



Fig. 13. Maximum Fuel Temperatures During LPCC Accident

## 4. CONCLUSIONS

Deployment of a hydrogen production system using VHTR technology is required to enhance security and efficiency of domestic fuel supplies in the future. There are still numerous technical challenges that must be resolved in relation to hydrogen production systems using VHTR heat. However, VHTR based hydrogen production is regarded as the most practical and economical approach. Most problems have viable solutions that can be identified and verified during further research and development in the near future.

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