

# CONCEPT DESIGN AND PHYSICS ANALYSIS OF GENERATION IV NUCLEAR ENERGY SYSTEMS

Sponsored by the Reactor Physics Division

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Session Organizer: Gray Chang (INEEL)

## 1. Evolution of Small Reactor Design—Advanced 4S Concept, *S. Hattori, M. Uotani, N. Ueda (CRIEPI–Japan)*

### INTRODUCTION

The concept of the Super Safe, Small and Simple (4S) Reactor was proposed to create green belts in the desertification area [1, 2]. The 4S Reactor has the following distinct features: greater simplicity; easy to maintain, inspect, and operate; less influence of human factors; high reliability; and improved availability and capacity. The original design of the 4S Reactor, however, has a rather high building due to the vertical arrangement of the components of the primary heat transfer circuit, and a steam generator (SG) arranged separately from the reactor vessel might enlarge the volume of the building.

The purpose of this paper is to describe a highly compact small reactor concept, the Advanced 4S Reactor, to meet global energy demand.

### ADVANCED 4S CONCEPT

The Advanced 4S Reactor, shown in Fig. 1, is conceived from the innovative reactor designs, the 4S Reactor and Modular Double Pool Reactor (MDP) [3, 4], which have been studied at CRIEPI.

The reactor assembly consists of two vessels: a primary vessel and a secondary vessel. The core and primary circuit are installed in the primary vessel, while the heat transfer tubes for the steam generating and secondary circuit are located in the annular space formed between the primary vessel and the secondary vessel. Electrical power output is set to be 5 MWe based on the world demand survey by IAEA.

The enriched U-Zr fueled core is designed so as to extend the life to as long as thirty years without fuel refueling by using burnable poison (ZrH-Gd) to compensate for the reactivity loss. The reactivity control system is the same as the original 4S Reactor; the reactivity is controlled only by the vertical movement of the annular reflector during plant startup, shutdown, and power generation.

The primary coolant flow is induced by natural circulation during rated operation. The simple flow configuration in the primary vessel and integrated fuel pin concept without wrapper tubes reduces the friction loss [5].

A unique feature of the Advanced 4S Reactor is that the heat is transferred from the primary coolant to the secondary coolant through heat transfer tubes that unite the primary vessel. The inlet and outlet chambers of the secondary coolant are incorporated with the vessel wall. The secondary coolant flows upward in the tubes driven by the magnetic pump, while the primary coolant flows down outside the tubes from the top of the chimney by natural circulation.

Double pool configuration makes it possible to reduce the volume of the reactor building, which improves the plant construction costs.

### TOWARD FURTHER EVOLUTION

Natural circulation cooling is able to simplify the reactor configuration. The primary vessel, however, requires a tall struc-

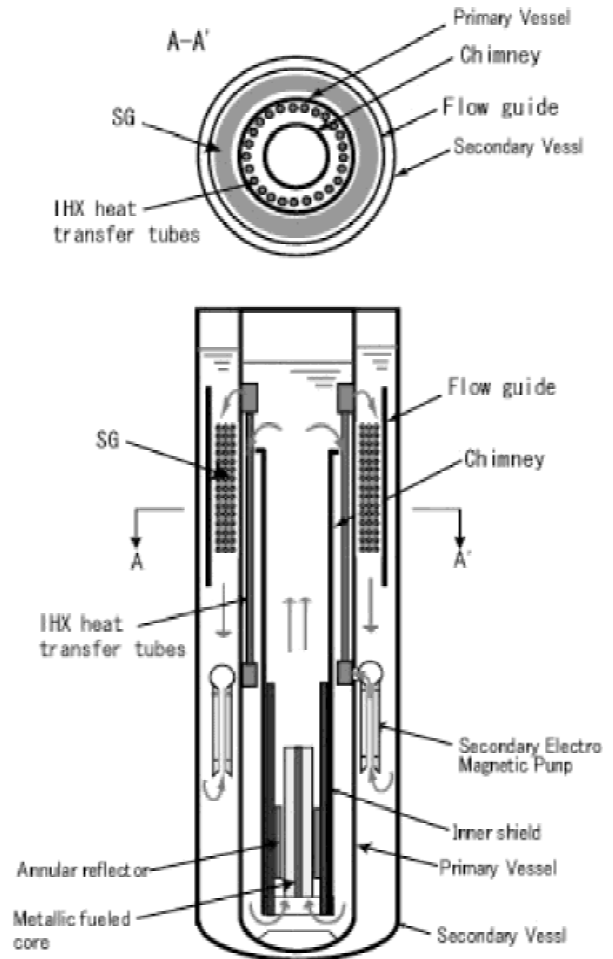


Fig. 1. Concept of Advanced 4S Reactor.

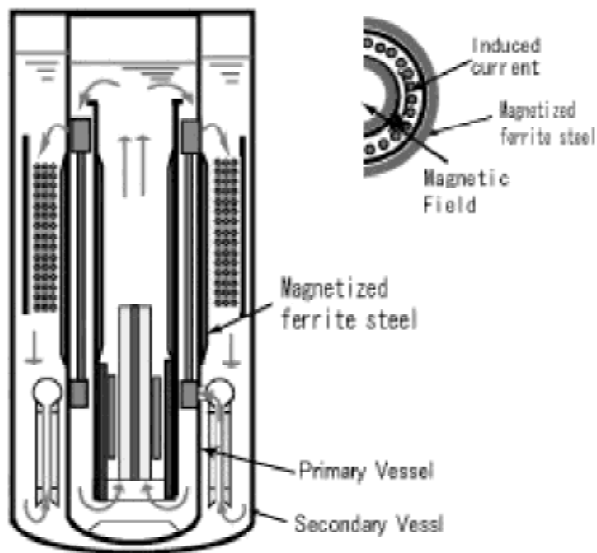


Fig. 2. Innovative concept of Advanced 4S Reactor with flow coupler.

ture like a chimney to secure the natural convection head. We originated an innovative concept of Advanced 4S Reactor with an annular electromagnetic flow coupler [6] to shorten the vessel height, shown in Fig. 2.

The annular electromagnetic flow coupler, which is a kind of a DC electromagnetic pump, consists of “generator channels” and “pump channels” in series in an annular arrangement connected electrically in the radial magnetic field. In the magnetic field, the secondary coolant flow in the generator channels driven by electromagnetic pump induces circumferential current. The current and the magnetic field drive the primary coolant in the pump channels. 3D numerical analysis of the flow coupler and sodium tests were conducted successfully [7]. In the innovative concept, magnetized ferrite steels are located outside of the primary vessel and inside of the chimney.

The innovative concept of the Advanced 4S Reactor with the flow coupler is expected to realize a highly compact small reactor to meet global energy demand, though the design of the flow coupler remains to be optimized.

1. S. Hattori, N. Handa, “Use of Super-Safe, Small, and Simple LMRs to Create Green Belts in Desertification Area,” *Trans. Am. Nucl. Soc.*, **60** (1989).
2. S. An, A. Minato, “Study on the Application of Nuclear Energy to Human Welfare and Safety,” presented at 4th Annual Scientific and Technical Conf. Nuclear Society (NE’93), Russia, 1993.
3. U.S. Patent 4645633 (1987).
4. I. Kinoshita et al., “Development of Small Modular Double Pool Reactor for Early Realization of FBR Practical Application,” *Proc. Int. Conf. Fast Reactors and Related Fuel Cycles*, Kyoto, 1991.
5. M. Kambe, M. Uotani, “Innovative Fast Breeder Reactor Concept ‘RAPID’ for Improvement of Reactor Performance and Proliferation Resistance,” *Nucl. Eng. Des.*, **170** (1997).
6. S. Hattori et al., “Studies on Applicability of Flow Coupler to LMFBR Plant,” presented at Int. Conf. Fast Breeder Systems:

Experience Gained and Path to Economical Power Generation, Richland, 1987.

7. T. Ikeda et al., “Sodium Tests on Head-Flow Characteristics for a Prototype Annular Electromagnetic Flow Coupler,” presented at LIMET-88, Avignon, 1988.

## 2. The Design Concept of PWR-2000 Based on APR1400, Soon Heung Chang, Yoon Sun Chung (KAIST–Korea), Won-Pil Baek (KAERI, Taejeon–Korea)

### INTRODUCTION

Because of the rapid growth of electric demand, the lack of domestic energy sources, and the limitations related to other energy sources, Korea has been heavily relying on nuclear energy. Presently, nuclear power shares about 40% of Korea’s electricity generation and should be further increased in the range of 40 to 50%.

There are advanced light water reactor concepts of 1300 to 1500 MWe at various stages of operation, construction, or development around the world: ABWR by General Electric, Toshiba, and Hitachi; System 80+ by ABB-CE; EPR by Framatome and Siemens; APWR by WH and Japanese companies [1]; APR1400 by KEPCO; etc. APR1400 will be built in Korea to start commercial operation in 2010 [2]. To further improve economic competitiveness, there are some efforts to increase the power rating up to 1700 MWe.

In general, higher power rating provides advantages from the viewpoint of economics and site area, if it is technically feasible and suitable for the electric power grid system. In this regard, we started to develop the concept of an extra-large scale PWR, PWR-2000. The name of PWR-2000 indicates the PWR that is of 2000 MWe rating and suitable for the new millennium.

This paper presents the overall system configuration with important design parameters and future developmental directions of PWR-2000.

### SYSTEM CONFIGURATION

#### Reactor and Reactor Coolant System

The nuclear steam supply system of the PWR-2000 is designed to operate at thermal output of 5900 MWt to produce an electric power of 2000 MWe.

*Reactor.* Basically, the design characteristics of APR1400 are maintained except for the increase in the number of fuel assemblies. Two options of the core are being assessed based on 17×17 Westinghouse fuel assemblies and 16×16 ABB-CE fuel assemblies. The core design would allow the refueling interval of 18 to 24 months.

*Reactor Coolant System.* The reactor coolant system consists of a reactor and three reactor coolant loops, as shown in Fig. 1. Each loop consists of a hot leg, two cold legs, one steam generator, and two reactor coolant pumps. A pressurizer is connected to one of the hot legs. By increasing the number of coolant loops in proportion to the power, most of the reactor coolant system components of APR1400, except for the reactor vessel and pressurizer, can be used without any modifications.

#### Safety System Features

The safety systems consist of a 6-train safety injection system (SIS), a safety depressurization system, an in-containment refueling water storage tank, a shutdown cooling system, a 6-train emergency feedwater system, and a containment spray system.

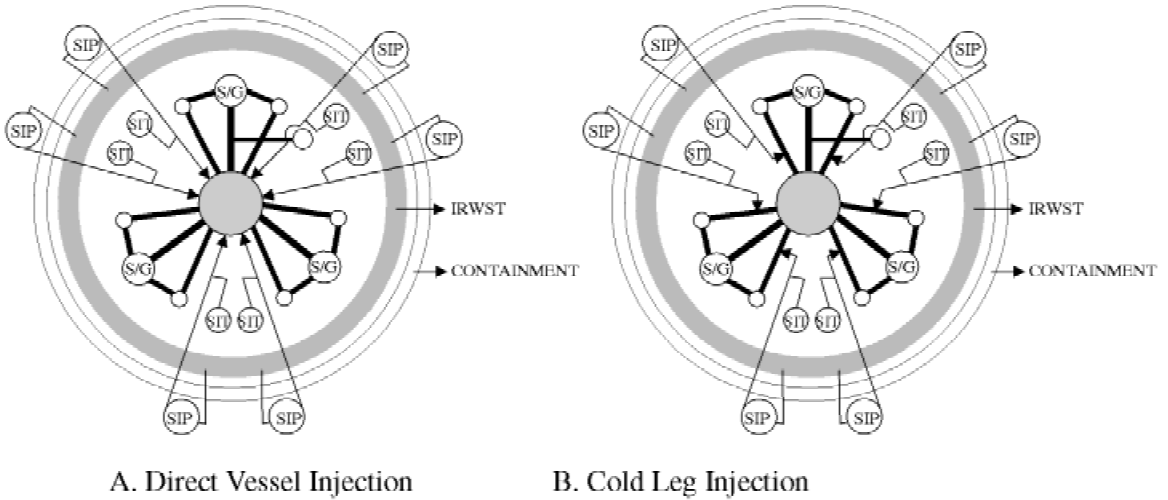


Fig. 1. Arrangement of PWR-2000 RCS and SIS.

Therefore, the safety system is basically the same as that for APR1400, but the effective emergency injection per power is increased considering the single failure criterion. Presently, two safety injection strategies are considered: direct vessel injection and cold leg injection as shown in Fig. 1. They are both estimated to satisfy safety requirements; further analysis is required to fix the injection method.

**COST ESTIMATE**

The construction cost of PWR-2000 has also been estimated by applying the method used for APR1400 [3, 4]. Assum-

ing that 2 units are constructed in Korea, the construction cost of PWR-2000 would be 890 \$/kWe, which is significantly smaller than 1146 \$/kWe for APR1400.

**CONCLUSION**

In order to prepare for the increase in energy demand, this paper suggests a new extra-large scale PWR concept with maximum utilization of APR1400 design features. The preliminary conceptual design has been performed as summarized in Table I, compared to APR1400.

TABLE I  
Summary of PWR-2000 Preliminary Design

	PWR-2000		APR1400
Type and Electric Power, MWe	PWR, 2000		PWR, 1400
Plant Lifetime, years	60		60
Refueling Interval, months	18–24		18–24
Containment	Double Cylindrical Building		Double Cylindrical Building
Reactor Coolant System	3-loop		2-loop
Thermal Power, MWth	5900		4000
Coolant Pressure, psia (MPa)	2250 (15.5)		2250 (15.5)
Coolant Inlet/Outlet Temp., °C	292/326		296/327
Total Mass Flow, kg/s	29,450		20,889
Pressurizer Heat Capacity, kW	3540		2400
Fuel Assembly Type	17 × 17 (264 rods/ass.)	16 × 16 (236 rods/ass.)	16 × 16 (236 rods/ass.)
No. of Fuel Assembly	293	333	241
Effective Core Height, m	4.267	4.267	3.81
Core Equivalent Diameter, m	4.16	4.28	3.65
Average Linear Heat Rate, kW/m	17.90	17.60	18.37
Reactor Vessel ID, m	5.22	5.37	4.64
No. of Trains	6		4
Injection Methods	Direct Vessel Injection or Cold Leg Injection		Direct Vessel Injection
No. of High Pressure Injection Pump	3		2
No. of Low Pressure Injection Pump	3		2

More detailed analysis will follow in the aspects of safety analysis, constructability analysis, design optimization, and design improvement.

1. Y. Aeba, E. H. Weiss, "APWR-Mitsubishi, Japan/Westinghouse, USA," IAEA-SM-353/46, pp. 467-481.
2. S. J. Cho, D. W. Jeong, "Research Activities and Design Requirements for the Next Generation Reactor in Korea," KEPKO Research Center (July 1995).
3. G. H. Ha, Y. C. Kang, "KNGR Economic Analysis Using the EPRI ALWR Cost Estimating Groundrules," TR.95ZJ16.J1998.78, KEPRI (Oct. 1998).
4. G. H. Ha, Y. C. Kang, "The Final Evaluation Report for the Cost Estimate of KNGR," TR.95ZJ16.J1999.212, KEPRI (Feb. 1999).

### 3. Design of Burnable Particles for Reactivity Control in HTRs, V. Berthou, J. L. Kloosterman, H. van Dam, T. H. J. J. van der Hagen (Delft UT-Netherlands)

During the operation of a nuclear reactor, the reactivity loss due to fuel burnup and fission product poisoning must be compensated by some means of long-term reactivity control. This can be achieved by adding burnable poison to the fuel. In this study we focus on heterogeneous poisoning in which burnable particles (TRISO-coated particles containing burnable poison only) are mixed with the fuel particles in the graphite matrix. Because of the many degrees of freedom, the optimum dimensioning and concentration of the burnable particles is not straightforward.

To minimize the reactivity swing as a function of irradiation time, we used the SCALE4.2 code system [1] with JEF2.2 nuclear data libraries [2] to perform numerous burnup calculations on a macro-cell that consists of a burnable particle surrounded with a layer containing a homogeneous mixture of fuel particles

and graphite. For the fuel considered first, the ESKOM PMBR [3, 4] has been chosen with an average core power density of  $3 \text{ MW} \cdot \text{m}^{-3}$ . The fuel particles contain 8% enriched  $\text{UO}_2$  with a uranium mass of 9 gram per pebble. One pebble contains more than 13 000 fuel particles, each with a radius of 0.5 mm. For the poison in the burnable particle, we have chosen  $\text{B}_4\text{C}$  with either natural boron or 100% enriched  $^{10}\text{B}$ .

When we consider a "black" burnable particle, which means that every neutron that hits the surface will be absorbed, the effective absorption cross section of the particle is related to its geometrical cross section [5, 6]. In spherical geometry, different radii are considered with a constant volume ratio of the fuel and the burnable particles. This means that the number of burnable particles per pebble increase when the radius of each particle decreases. The smaller the burnable particle, the more homogeneous the poison is distributed, and the faster the particles will burn. In Fig. 1, the reactivity is shown as a function of time for the reference case without poison and for two radii of the burnable particles. The reactivity swing is smallest for the burnable particle with a radius of 0.3 mm.

In the case of a spherical burnable particle with a radius of 0.5 mm, both natural boron and enriched boron (100% of  $^{10}\text{B}$ ) have been considered. The initial reactivity is lower for the case with enriched boron, which indicates that the natural boron particle cannot be considered as a "black" one. Its diameter is about 3 absorption mfp.

For the same radius, the effective absorption cross section is larger for a cylindrical shape. As a result, the cylindrical particle with a radius of 0.5 mm is fully burnt already after 1000 Equivalent Full Power Days (EFPD), while the spherical particle still contains significant amounts of  $^{10}\text{B}$  after 1600 EFPD. For the same effective absorption cross section (this means a smaller radius for the cylindrical particle) and the same ratio of fuel volume and poison volume, the reactivity swing is smallest for the cylindrical particle.

Figure 2 shows the reactivity for a cylindrical burnable particle with a radius of 0.5 mm. The poison consists of  $\text{B}_4\text{C}$  with 100% of  $^{10}\text{B}$ . For the ratio of the fuel volume and the poison volume equal to 10300, the reactivity swing is only 2.2% up to 1600 EFPD. At this time, the burnable particle is fully burnt.

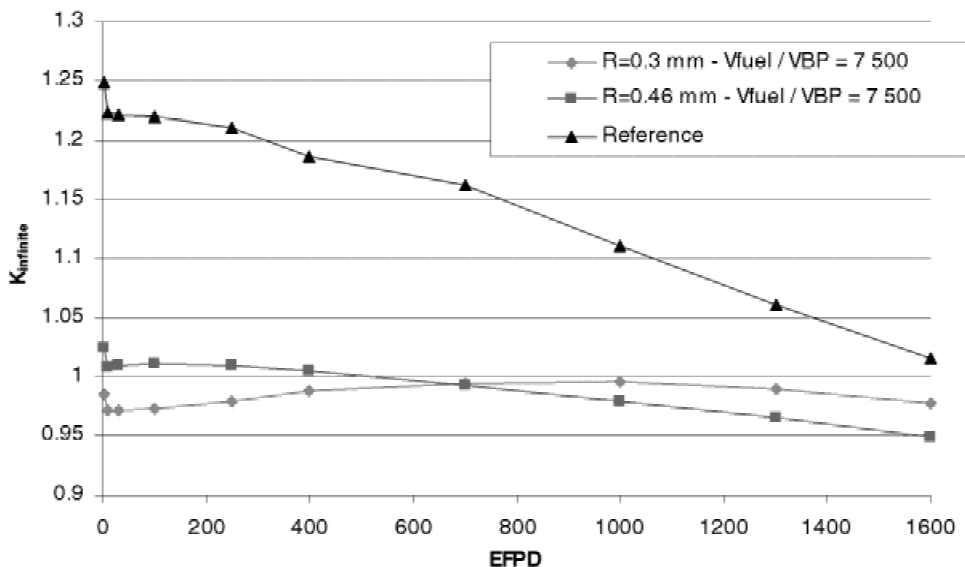


Fig. 1.  $K_{\infty}$  as a function of irradiation time for the reference case without BP, and for spherical burnable particles with different radii.

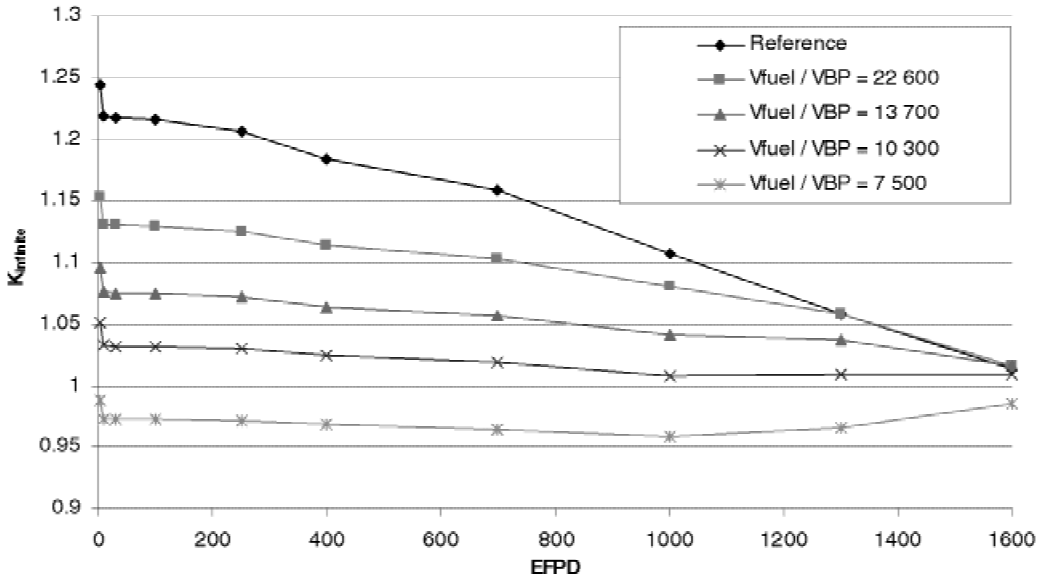


Fig. 2.  $K_{infinite}$  as a function of irradiation time for the reference case without BP and for the cylindrical burnable particles (0.5 mm of radius) with the volume ratio between the fuel particle and the burnable particle as a parameter.

As a conclusion, we can state that the heterogeneous poisoning of HTR fuel seems quite promising. We have designed burnable particles containing  $B_4C$  that reduce considerably the reactivity swing. Further studies will focus on burnable particles that contain graphite or fuel surrounded with a thin layer of poison, on different poison materials (e.g., Er or Gd), and on different fuel compositions. The ultimate aim is to abandon active long-term reactivity control in HTRs.

The authors like to acknowledge the European Commission for co-funding this research that is performed in the project “High Temperature Reactor Physics and Fuel Cycle Studies (HTR-N)” [7] together with other European Organizations.

1. SCALE-4.2, “Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations,” Oak Ridge National Lab. (1994).
2. JEF-2.2, “The JEF-2.2 Nuclear Data Library,” JEFF17, Nuclear Energy Agency (2000).
3. J. H. Gittus, “The ESKOM Pebble-Bed Modular Reactor,” *Nucl. Energy*, **38**, 4, 215 (1999).
4. D. R. Nicholls, “Status of the Pebble-Bed Modular Reactor,” *Nucl. Energy*, **39**, 4, 231 (2000).
5. H. Van Dam, “Long-Term Control of Excess Reactivity by Burnable Particles,” *Ann. Nucl. Energy*, **27**, 733 (2000).
6. H. Van Dam, “Long-Term Control of Excess Reactivity by Burnable Poison in Reflector Regions,” *Ann. Nucl. Energy*, **27**, 63 (2000).
7. “High Temperature Reactor Physics and Fuel Cycle Studies (HTR-N),” FIKI-CT-2000-00020, available on the Internet (<http://www.cordis.lu/fp5/projects.htm>).

#### 4. Matrix Formulation for Automated Modeling of Pebble Circulation in the PEBBED Code, H. D. Gougar, W. K. Terry, A. M. Ougouag, C. B. Neill (INEEL)

##### INTRODUCTION

The PEBBED [1] technique provides the foundation for fuel cycle analysis and optimization in pebble bed cores in which the fuel elements are continuously flowing and, if desired, recirculating. The original PEBBED-1.0 code was limited to two simple fuel recirculation schemes. The current version incorporates a novel nuclide mixing algorithm that allows for sophisticated recirculation patterns using a user-supplied matrix. This provides the capability to perform extensive fuel-cycle optimization studies using modern optimization methods.

##### NUCLIDE FLOW IN RECIRCULATING CORES

Reference 1 describes the PEBBED algorithm in which the equations for neutron flux and nuclide distribution in a pebble bed core are solved simultaneously. A key step in the algorithm is the computation of the entry plane density for each axial flow channel. These values depend upon the procedure governing recirculation and on the burnup increments accrued by pebbles on successive passes through the core. Starting with the fresh fuel concentrations of the pebbles entering each channel, the exit plane values are computed from the current iterate of the flux and then mixed according to the recirculation scheme to generate the entry plane densities for the next pass. This is repeated until the pebbles exceed the discharge burnup. The exit plane values are then averaged according to the recirculation scheme in order to produce the actual entry plane nuclide densities. The entry plane nuclide flow rate is derived as follows.

One can show that the nuclide density at the entry plane of channel  $i$  is given by

$$N_i = \sum_{p=1}^P \left\{ {}^1\hat{N}_i^p \cdot {}^1\alpha_i^p \cdot \alpha_i^p + \sum_{m=1}^{M_{max}-1} \sum_{j=1}^J \frac{{}^m\hat{N}_j^p \cdot \alpha_j \cdot \alpha_j^p \cdot m \cdot \alpha_j^p \cdot {}^m\alpha_{ij}^p}{\alpha_i} \right\}, \tag{1}$$

where

$\alpha_i$  = fraction of pebble flow that passes through channel  $i$

$\alpha_j^p$  = fraction of channel  $j$  flow that consists of type  $p$  pebbles

${}^m\alpha_j^p$  = fraction of type  $p$  pebbles in channel  $j$  flow that are on their  $m$ 'th pass

${}^m\alpha_{ij}^p$  = fraction of type  $p$  pebbles in flow channel  $j$ , and on pass  $m$ , that are diverted to channel  $i$ , following this  $m$ 'th pass

${}^m\hat{N}_j^p$  = number density of the nuclide of interest within pebbles of type  $p$ , exiting channel  $j$ , after completing their  $m$ 'th pass

$f_j$  = flow rate of pebbles through channel  $j$ .

The flow partition coefficients ( $\alpha$ ) and the total core flow rate span a subspace of the overall domain over which fuel optimization can be performed. The multipliers formed by these coefficients in Eq. (1) become the elements of a recirculation matrix  $\mathfrak{R}$ . The values for  $\alpha_j$ ,  $\alpha_j^p$ , and  ${}^m\alpha_{ij}^p$  uniquely determine  ${}^m\alpha_j^p$  and are dependent upon the number of pebble types and the flexibility of the pebble loading and discharge mechanisms. All are computed in advance of the flux calculation. For example, if the core is to contain only one pebble type such as in the HTR Modul 200 design [2], then  $p = 1$ , and

$$\alpha_j^p = \alpha_j. \quad (2)$$

The pebbles in this design are loaded and unloaded in a purely random manner. The values of the recirculation matrix elements are thus determined entirely by the maximum number of passes that the average pebble completes before final discharge ( $M_{\max}$ ) and the fractional flow area of each of the flow channels ( $\alpha_j$ ). For  $J$  flow channels in the core, the recirculation matrix ( ${}^m\mathfrak{R}$ ) takes the form

$${}^m\mathfrak{R} = \frac{1}{M_{\max}} \begin{bmatrix} \alpha_1 & \alpha_2 & \dots & \alpha_J \\ \alpha_1 & \alpha_2 & \dots & \alpha_J \\ \dots & \dots & \dots & \dots \\ \dots & \dots & \dots & \dots \\ \alpha_1 & \dots & \dots & \alpha_J \end{bmatrix}. \quad (3)$$

The Pebble-Bed Modular Reactor (PBMR) design under consideration by the South African utility ESKOM uses two pebble types (graphite and fuel) flowing through five channels in the core [3]. The innermost channel is composed of only graphite pebbles to limit power peaking. The second channel is a 50/50 mixture of fuel and graphite pebbles and the outer three channels consist only of the fueled type. The radial placement and discharge of pebbles is not burnup (pass) dependent. The recirculation matrix ( ${}^m\mathfrak{R}$ ) can be expressed as two submatrices: one for fuel,

$$\mathfrak{R}^f = \frac{1}{M_{\max}\alpha^f} \begin{bmatrix} 0 & 0 & 0 & 0 & 0 \\ 0 & .25\alpha_2 & .5\alpha_3 & .5\alpha_4 & .5\alpha_5 \\ 0 & .5\alpha_2 & \alpha_3 & \alpha_4 & \alpha_5 \\ 0 & .5\alpha_2 & \alpha_3 & \alpha_4 & \alpha_5 \\ 0 & .5\alpha_2 & \alpha_3 & \alpha_4 & \alpha_5 \end{bmatrix}, \quad (4a)$$

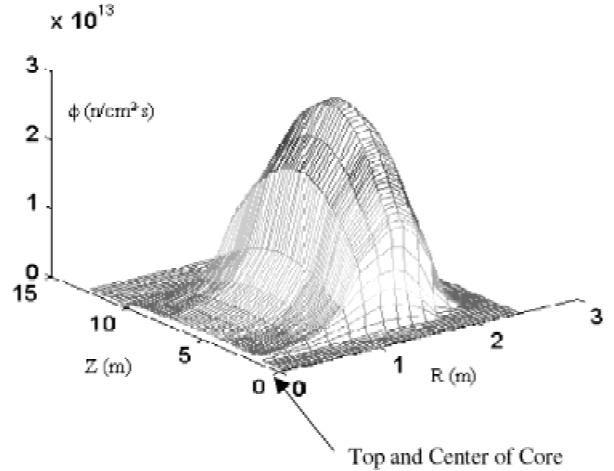


Fig. 1. Fast flux in ESKOM PBMR core.

and one for graphite,

$$\mathfrak{R}^g = \frac{1}{M_{\max}\alpha^g} \begin{bmatrix} \alpha_1 & .5\alpha_2 & 0 & 0 & 0 \\ .5\alpha_1 & .25\alpha_2 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \end{bmatrix}. \quad (4b)$$

The coefficients  $\alpha^f$  and  $\alpha^g$  refer to the fraction of pebbles in the core composed of fuel and graphite, respectively.

## RESULTS

A fast flux map (Fig. 1) is shown for the ESKOM case described above. The significant drop in flux near the core center indicates the lack of fuel in the central graphite column.

## CONCLUSION

We present an enhancement to the PEBBED-1.0 fuel cycle analysis code that allows for the computation of complex and user-specified pebble recirculation patterns. PEBBED-2.0 provides a potentially powerful tool for analyzing equilibrium core configurations. The addition of the recirculation matrix will allow the analyst to perform detailed sensitivity studies and efficient fuel cycle optimization using advanced optimization techniques.

1. W. K. Terry, H. D. Gougar, A. M. Ougouag, "Deterministic Method for Fuel-Cycle Analysis in Pebble-Bed Reactors," *Trans. Am. Nucl. Soc.*, **83**, 278 (2000).
2. H. Frewer, W. Keller, R. Pruschek, "The Modular High Temperature Reactor," *Nucl. Sci. Eng.*, **90**, 411 (1985).
3. H. de Haas, Kuijper, Oppe, "PBMR: Evaluation of the Equilibrium Core Enrichment," presented at 3rd Mtg. IAEA CRP-5, Oarai, Japan, March 2001.

**5. A Supercritical CO<sub>2</sub> Brayton Cycle for Advanced Reactor Applications, V. Dostal, P. Hejzlar, M. J. Driscoll, N. E. Todreas (MIT)**

A closed Brayton cycle employing supercritical CO<sub>2</sub> as the working fluid (Feher cycle) was proposed but not developed in 1967 [1]. A recompression cycle of this type is worth reconsideration for Generation IV reactors because it develops high thermodynamic efficiency (e.g., 45%) at modest turbine inlet temperatures (e.g., 550°C) due to the low compressive power expended on CO<sub>2</sub> near its critical point (31°C, 7.285 MPa) compared to nearly ideal gas working fluids such as helium.

The low core exit temperature that this cycle permits greatly expands the roster of suitable candidate materials for advanced reactor cores and system components and also increases thermal-hydraulic margins to core damage under transient and accident scenarios.

Experience with supercritical CO<sub>2</sub> gas pipelines for oil recovery operations in the western United States since about 1985, and with the 14 CO<sub>2</sub>-cooled AGR units in the United Kingdom (core exit T ≈ 650°C; p = 4.2 MPa) in the same time frame have also now established the previously lacking industrial experience base for relevant technology. Improvements in regenerator design are also a relevant favorable development.

Figure 1 shows one specific design of a supercritical CO<sub>2</sub> cycle, here an indirect version developed for a lead-bismuth eutectic (LBE)-cooled reactor application. However, for CO<sub>2</sub>-coolable cores, a direct cycle is readily substituted, with the reactor core in place of the intermediate heat exchanger in Fig. 1. Efficiencies around 55% are attainable at higher temperatures (e.g., 850°C). For comparison, the helium-cooled Pebble Bed Modular Reactor (PBMR) has a predicted efficiency of 44% for T<sub>max</sub> = 850°C, with P<sub>max</sub> = 8 MPa.

Parametric studies show that cycle thermodynamic efficiency is most sensitive to the effectiveness of the regenerative

heat exchangers, here specified as an ambitious 98%. Careful design of the regenerator will, therefore, be one of the most important issues. Preliminary studies have shown that around 1.8 times more heat is transferred in a supercritical CO<sub>2</sub> heat exchanger occupying the same volume as a typical HTGR helium heat exchanger. This indicates the potential for achieving high regenerator effectiveness.

Component sizes and other particulars differ from those now under design for the helium-cooled pebble bed modular reactor, because of the different fluid properties. In particular, the density of supercritical CO<sub>2</sub> is about 40 times higher than that of helium. Despite the lower specific heat, the volumetric flow rates for the supercritical CO<sub>2</sub> cycle should be around 6 times less than for the helium cycle generating the same amount of power. Thus, the size of the plant components should be much smaller than for the case of helium. Figure 2 shows the advantage of the supercritical CO<sub>2</sub> cycle over the helium cycle, especially at lower temperatures (helium Brayton cycle with 3 compressors and 2 intercoolers, working between the same temperatures as the supercritical CO<sub>2</sub> recompression cycle).

Because of its physical properties, the CO<sub>2</sub> working fluid turbomachinery will be similar to the widely familiar open cycle gas turbines used for electricity generation and airplane propulsion.

The expected cost of the supercritical CO<sub>2</sub> cycle should be very competitive. High thermodynamic efficiency increases the plant power generation per unit primary system volume. Low temperatures should result in higher reliability and longer life of the plant components, therefore increasing the plant capacity factor. Small-size components will also result in savings in the capital cost relative to a helium-cooled unit.

In conclusion, a supercritical CO<sub>2</sub> Brayton cycle with recompression appears to be well suited to many Generation IV reactor requirements. At MIT, we are currently evaluating its use for a lead-bismuth cooled fast breeder reactor, a CO<sub>2</sub>-cooled fast breeder reactor, and as an alternative coolant for a PBMR-type thermal reactor.

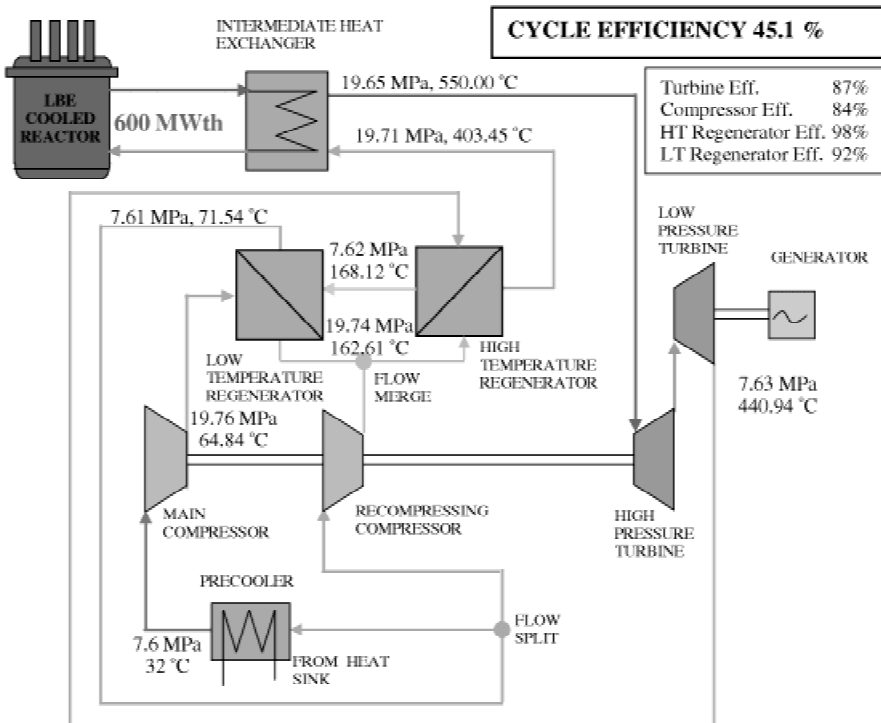


Fig. 1. Layout of the supercritical CO<sub>2</sub> recompression cycle for LBE-cooled reactor.

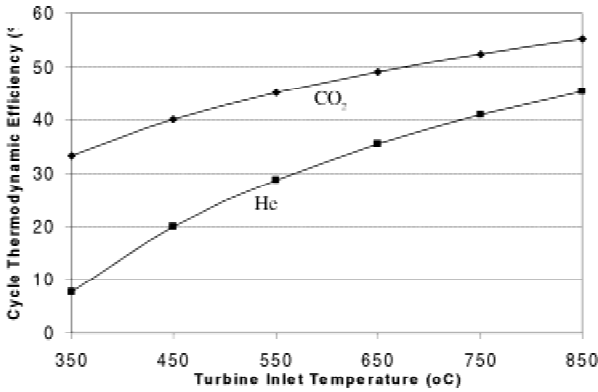
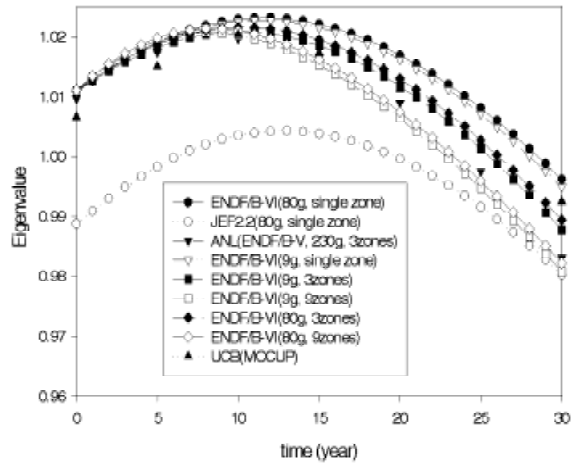
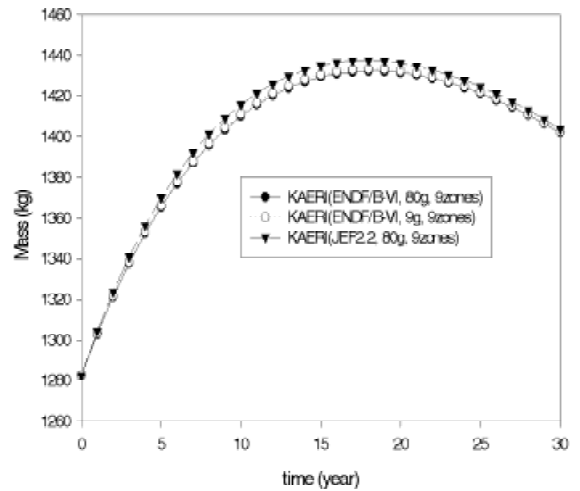


Fig. 2. Cycle thermodynamic efficiency of supercritical CO<sub>2</sub> versus helium.



(a) Eigenvalue versus time (lead cavity)



(b) Mass (kg) of Pu-239 versus time

Fig. 1. Results of depletion calculation.

1. E. G. Feher, "The Supercritical Thermodynamic Power Cycle," presented at Advances in Energy Conversion Engineering, August 13–17, 1967.

## 6. Comparison of the ENDF/B-VI and JEF2.2 Nuclear Data for the ENHS Benchmark Problem, Ser Gi Hong, Yeong Il Kim (KAERI–Korea)

### INTRODUCTION

Recently, the Encapsulated Nuclear Heat Source (ENHS) [1], which is a new Pb or Pb-Bi cooled highly modular fast spectrum reactor concept, has been proposed and studied. The proposed core was designed to satisfy proliferation resistance, simplified control, improved safety, etc. The neutronic characteristics of the core depend on the computer codes used, including basic nuclear data library, cross section processing code, and core calculation methods (diffusion theory or transport theory). Therefore, in the preliminary stage, it is important that several computer code systems are applied for clearly defined problems and inter-compared to clarify the calculated results. In this paper, the calculation results for the ENHS neutronics benchmark problem are presented. We used the REBUS-3/DIF3D code [2] for core neutronics calculation, the TRANSX code [3] for the multi-group cross section preparation, the KAFAX-E66 library [4] (based on ENDF/B-VI.6), and the KAFAX-F22 [5] (based on JEF2.2) as the master nuclear libraries.

### COMPUTATIONAL METHOD AND CODES

The detailed description of the benchmark problem are given in Refs. [1] and [6]. We tested the reference case (250 MWth,

TABLE I  
Basic Quantities (BOC)

Quantities	KAERI (ENDF/B-VI)	KAERI (JEF2.2)	UCB (ENDF/B-VI)
Initial Conversion Ratio	1.1764	1.1819	1.1629
Lead Cavity Worth ( $\delta k$ )	+0.0272	+0.0262	+0.02319
Void Reactivity (core) ( $\delta k$ )	+0.0194	+0.0218	+0.01938
Hot-Cold Reactivity Swing ( $\delta k$ )	-0.0076	-0.0075	-0.00466
Initial DIF3D Keff (fine group)	1.010295(150g)	0.988721(80g)	
Initial TWODANT Keff(25g)	1.0126089	0.990984	1.015(MCNP)

TABLE II  
One Group Effective Cross Sections

Nuclides	Capture			Fission		
	KAERI (ENDF/B-VI)	KAERI (JEF2.2)	ANL (ENDF/B-V)	KAERI (ENDF/B-VI)	KAERI (JEF2.2)	ANL (ENDF/B-V)
Pu-239	0.3093	<b>0.3487</b>	0.3128	1.6579	1.6741	1.678
Pu-240	0.3679	<b>0.4114</b>	0.3740	0.3854	0.3526	0.3858
Pu-241	0.3186	<b>0.4627</b>	0.3311	2.1266	2.1766	2.109
Pu-242	0.3226	<b>0.3462</b>	0.3858	0.2690	0.2421	0.2732
U-235	0.3957	<b>0.4054</b>	0.4074	1.5954	1.6197	1.586
U-238	0.2072	<b>0.2143</b>	0.2126	0.0334	0.0309	0.0339
FPU235	0.1958	0.2017	0.2393			
FPU238	0.2605	0.2659	0.3026			
FPPU239	0.2828	0.2893	0.3330			
FPPU240	0.2955	0.3013	0.3452			
FPPU241	0.3020	0.3131	0.3593			
Np-237	1.1278	1.1736	1.138	0.3371	0.3005	0.3581
Am-241	1.2225	1.5675	1.282	0.2587	0.2247	0.2785
Am-242m	0.2360	0.4321	0.2269	3.2401	2.7459	3.178
Am-243	1.1053	1.3208	0.9512	0.1869	0.1723	0.2206
Cm-242	0.1912	0.3479	0.1815	0.1454	0.5779	0.1470
Cm-243	0.1628	0.1511	0.1566	2.2309	2.8921	2.178
Cm-244	0.6421	0.4554	0.6237	0.4411	0.4009	0.4424
Cm-245	0.2385	0.2431	0.2384	1.9339	2.3084	2.329
Cm-246	0.1675	0.1725	0.1616	0.2640	0.2324	0.2638

4 m core height, 120 W/cm, 1.15 P/D ratio, 10.9% Pu in heavy metal) proposed by Greenspan et al. Two nuclear data libraries (ENDF/B-VI, JEF2.2) were used. These nuclear data libraries were processed into the MATXS format by the use of the NJOY code and typical fast breeder spectrum. The numbers of neutron energy groups for ENDF/B-VI and JEF2.2 are 150 and 80, respectively. The MATXS format libraries were transformed into the ISOTXS format by TRANSX [3]. In TRANSX, the self-shielding effect was treated with the constant escape method. Basically, the REBUS-3/DIF3D [2] code (80 group or 9 group) was used in the depletion calculation. The lumped fission product model, based on isotopic data for 172 fission product isotopes was used to model the fission products from 17 fissionable isotopes and one dummy isotope. The eighty group microscopic cross section of the lumped fission products was generated by weighting with fission yield (ENDF/B-VI based). The decay chain spans the range from U-234 to Cm-246. For ENDF/B-VI, the eighty and nine group cross sections were made by condensing the 150 group one with the spectrum of 150 group DIF3D calculation. The space dependent effect of isotopic composition changes was analyzed by changing the zone division for depletion.

## RESULTS AND CONCLUSIONS

The results at BOC are given in Table I. The twenty-five group cross sections for TWODANT [7] were made by condensing with the spectrum of the fine group DIF3D calculation. The percent error of eigenvalues between TWODANT and MCNP [1] is approximately 0.2% for ENDF/B-VI. It is considered that this error is due to the difference between the Monte Carlo method and the  $S_N$  method. However, the discrepancy between JEF2.2 and ENDF/B-VI is large (approximately 2%). This is due to the larger capture cross section in JEF2.2 (see Table II). Therefore, if the results of JEF2.2 are correct, initial criticality is not assured for this problem. Other quantities (conversion ratio,  $K_{\text{eff}}$  differences) show good agreements comparatively. For heavy isotopes, one group effective cross sections calculated by REBUS-3 are given in Table II. The eigenvalues for all cases increase ini-

tially and then decrease. This initial increase of eigenvalues is due to the formation of Pu-239 (this is main fissile material) from the fission of U-238. Figure 1a shows the eigenvalue change versus time (years) for the lead cavity case. It must be noted that there are significant differences in the trends of eigenvalue versus time for three zone divisions. However, it is found that the number of energy groups has no effect both on the trend of eigenvalue versus time. Therefore, use of the nine energy group cross section set will give a significant saving of the computer running time without loss of accuracy. The eigenvalues of KAERI (ENDF/B-VI, 9 zones) and ANL [6, 8] agree fairly well while the eigenvalue of UCB [8] decreases more slowly. The change of Pu-239 mass (kg) in Fig. 1b shows that the difference between ENDF/B-VI and JEF2.2 is at most 2.0 kg at 30 years.

1. E. Greenspan, H. Shimada, K. Wang, "Long-Life Cores with Small Burnup Reactivity Swing," *Proc. ANS. Int. Topl. Mtg. Advances in Reactor Physics and Mathematics and Computation into the Next Millennium (PHYSOR2000)*, May 7-12, 2000.
2. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, Argonne National Lab. (Mar. 1983).
3. R. E. MacFarlane, "TRANSX 2: A Code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport Codes," LA-12312-MS, Los Alamos National Lab. (Dec. 1993).
4. J. D. Kim, "KAFAX-E66," Calculation Note No. NDL-23/01, Nuclear Data Evaluation Lab. Internal Report (2001).
5. J. D. Kim, C. S. Gil, KAERI/TR/-842/97, Korea Atomic Energy Research Institute (1997).
6. R. E. Alcouffe et al., "User's Guide for TWODANT: A Code Package for Two-Dimensional Diffusion-Accelerated, Neutral-

Particle Transport,” LA-10049-M, Los Alamos National Lab. (1984).

7. K. N. Grimm, “ENHS Neutronics Benchmark-Argonne Results,” ANL Intra-Laboratory Memo (2000).
8. K. Grimm et al., “ENHS Neutronics Benchmark,” ENHS Project Mtg. Presentation Material, June 25–26, 2001, ANL.

## 7. Dancoff Calculations for PBMR Fuel Using TransLAT 3-D Lattice Code, *Charlotte H. Potze, Steven P. Baker, Dean B. Jones (TransWare, San Jose)*

### INTRODUCTION

The increasing interest in High Temperature and Pebble Bed Modular Reactors has fueled a need for codes that are able to handle spherical fuel. Often, the methodology for Light Water Reactors with cylindrical fuel cannot be applied to the spherical fuel. In many codes it is not even possible to model the PBMR spherical fuel elements made of thousands of spherical fuel kernels. Currently, Monte Carlo methods are used to model spherical fuel, but TransWare Enterprises Inc. has developed a 3-D deterministic method as an alternative.

TransLAT [1] 3-D lattice physics software uses integral transport theory methods and 3-D arbitrary geometry techniques to solve neutron flux, gamma flux, and eigenvalue problems. This paper addresses the calculation of the infinite Dancoff factor for the spherical fuel kernels in the PBMR fuel assembly using TransLAT. The results from the TransLAT calculation are compared to the analytical results presented by Bende et al. [2].

### DESCRIPTION OF WORK

The Dancoff factor is the probability that a neutron leaving one absorber region does not have any collision in the moderator before it reaches another fuel region. The Dancoff factor for each absorber region is calculated using 3-D arbitrary geometry ray-

tracing techniques. TransLAT uses a 3-D ray-tracing methodology to determine the Dancoff interactions for arbitrary positioned and shaped fuel regions. All the rays going out of each absorber region are followed through the moderator until they reach another fuel region. The transmission probabilities are integrated along these paths, and the Dancoff factor is the sum of the probabilities due to the rays leaving that region.

The Dancoff factors are calculated for an infinite cubic lattice of fuel kernels surrounded by a graphite box. The kernels are the same TRISO particles as described in Ref. 2 that consist of a fuel sphere surrounded by four layers of coating. The fuel spheres have a radius of 250  $\mu\text{m}$ , and the coating layers all contain carbon in some form with thicknesses of 95, 40, 40, and 35  $\mu\text{m}$ . The fuel and each carbon layer are modeled explicitly in TransLAT. Calculations are performed for twelve kernel densities within the fuel zone of the pebble ranging from 5000 to 60 000 kernels. The dimensions of the graphite box surrounding the kernel are calculated by dividing the volume of the 2.5 cm radius pebble fuel zone by the number of kernels in each pebble. This unit cube is modeled in TransLAT with reflective boundary conditions.

### RESULTS

The results from the Dancoff calculation with TransLAT are given in Fig. 1 together with the analytical results calculated by Bende et al. The agreement is around 1% for all kernel densities greater than 10 000. The TransLAT results are 7% lower for the 5000 kernel problem. However, it was reported by Bende et al. that the analytical results were 4% higher than the results from MCNP. This suggests that the results with TransLAT are within 3% of the MCNP calculation for the 5000 kernel problem.

### CONCLUSIONS

The results obtained with TransLAT are in excellent agreement with the results from Bende et al. The next step is to model a pebble containing the kernels to determine the non-infinite Dancoff factors. This will then lead to the eigenvalue calculation for the complete system of fuel and moderator pebbles.

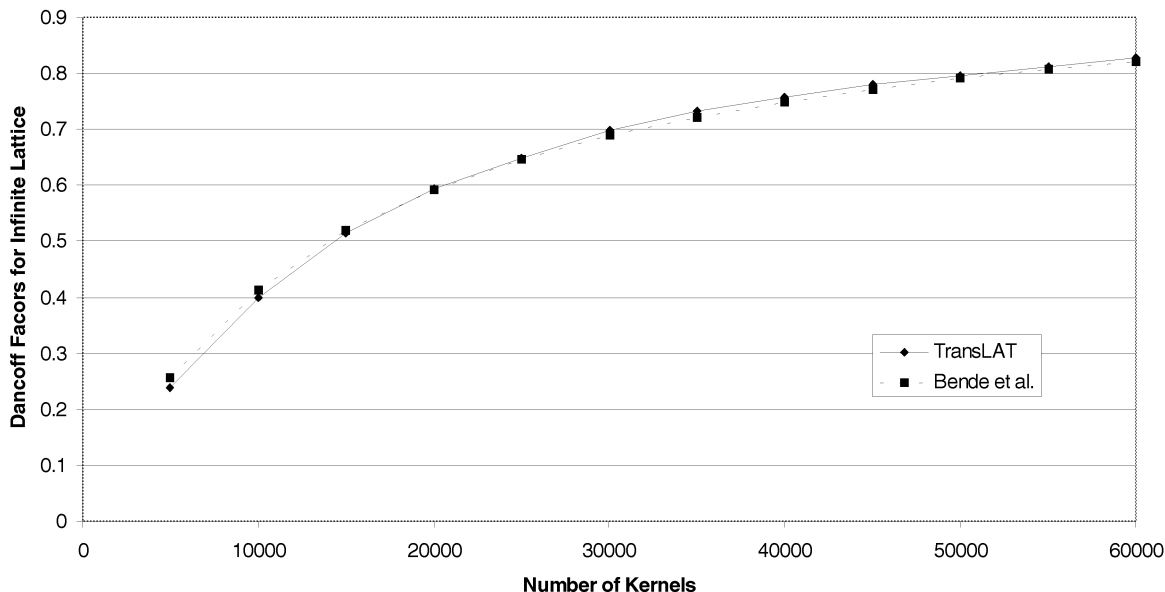


Fig. 1. Dancoff factor versus kernel density.

1. "TransLAT 3-D Lattice Physics Software," TransFX Computer Software Manuals, TWE-TFX-001 (June 2001).
2. E. E. Bende, A. H. Hogenbirk, J. L. Kloosterman, H. Van Dam, "Analytical Calculation of the Average Dancoff Factor for a Fuel Kernel in a Pebble Bed High-Temperature Reactor," *Nucl. Sci. Eng.*, **133**, 147 (1999).

## 8. An Evaluation of Eigenvalue Calculations Using Three-Dimensional Deterministic Methods, Dean B. Jones, Steven P. Baker (TransWare, San Jose)

### INTRODUCTION

The growing interest in next generation reactors is producing advanced fuel and reactor system designs of significant het-

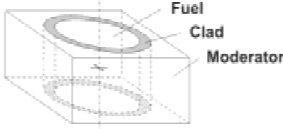
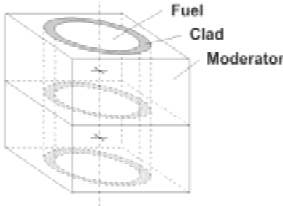
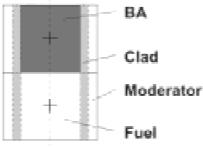
erogeneity. These designs require more advanced transport methods, especially methods that are capable of flexible and accurate three-dimensional modeling. The current standard for accurate 3-D modeling and analysis is based on Monte Carlo methods. While Monte Carlo methods can provide benchmark quality results, their routine application for production calculations is not generally practical. As an alternative, TransWare Enterprises Inc. is developing advanced transport software based on 3-D deterministic methods.

This paper describes TransWare's new TransLAT [1] 3-D lattice physics software and provides an evaluation of the software for solving light water reactor fuel lattices. Initial testing of the new TransLAT 3-D software has focused on solving simple pin cell problems, although more complex problems are being investigated. In all instances, the results calculated with TransLAT are compared to the MCNP4C [2] Monte Carlo code.

### METHODOLOGY

The TransLAT 3-D lattice physics software is a nuclear fuel burnup code that is capable of modeling highly heterogeneous

TABLE I  
TransLAT and MCNP4C Results for Three Test Configurations

Geometry Model	Code / Method (1,2)	K-Infinite	Delta K (LAT - MCNP)
<b>Simple Pin Cell</b> Isometric View 	<b>MCNP4C</b>	$1.34132 \pm 0.00039$	
	<b>TransLAT</b> MCPS-2D	1.34205	0.0007
	MOCS-2D	1.34191	0.0006
	MOCS-3D	1.33836	-0.0030
<b>Stacked Pin Cell</b> Isometric View 	<b>TransLAT</b> MOCS-3D	1.33908	-0.0022
<b>Stacked Pin Cell</b> Side View 	<b>MCNP4C</b>	$0.52231 \pm 0.00029$	
	<b>TransLAT</b> MOCS-3D	0.52446	0.0022

1. MCNP cases are run to 2 million histories.
2. MCPS-2D are two-dimensional, Method of Collision Probabilities cases.  
MOCS-2D are two-dimensional, Method of Characteristics cases.  
MOCS-3D are three-dimensional, Method of Characteristics cases.

fuel designs in three-dimensional geometry. TransLAT is a full featured software package that includes extensive user input and error checking capabilities, resonance self-shielding treatments, fuel burnup calculations, and editing calculations. The significant features tested in this paper include the 3-D geometry modeling capabilities and the performance of the Dancoff, resonance, transport and eigenvalue calculations.

The geometry modeling technique in TransLAT is based upon the MARS [3] software package. The MARS methodology provides a very general and flexible approach for describing three-dimensional geometry and is easily interfaced to solution techniques using ray-tracing methods. In TransLAT, ray-tracing is used to perform the Dancoff and particle transport calculations.

TransLAT incorporates an integrated Dancoff and resonance treatment capability for determining resonance self-shielding. The resonance calculation is based upon the space-dependent resonance self-shielding method described by Williams et al. [4]. The Dancoff treatment described by Williams is very flexible and easily handles regular and irregular lattices in two-dimensional geometry. For TransLAT, the Dancoff treatment has been extended to also handle three-dimensional geometries. The 3-D Dancoff treatment shows excellent agreement with comparable 2-D treatments and has recently been verified for spherical fuel designs [5].

The resonance self-shielding method described by Williams assumes fuel lumps of right circular cylinder design in two-dimensional planar space. The current formulation, although restrictive for more exotic fuel designs, is acceptable for three-dimensional modeling providing that the fuel lumps are described as right circular cylinders and the axial meshing of the fuel is specified explicitly by user input.

TransLAT supports two deterministic transport modules for calculating neutron and gamma-ray fluxes. The primary transport solver is based on the Method of Characteristics technique and is capable of solving both 2-D and 3-D transport problems. The second transport solver is based on the Method of Collision Probabilities, which performs only 2-D transport calculations. Results for both transport solvers are presented in this paper, as appropriate for the test problems.

The comparisons between TransLAT and MCNP are facilitated by a model translation routine in TransLAT that generates MCNP inputs from the TransLAT model description. To further minimize discrepancies in the comparisons, new cross sections for the  $^{235}\text{U}$  and  $^{238}\text{U}$  fuel isotopes are generated from ENDF/B-VI release 5 for both codes.

## RESULTS

Three test configurations are evaluated in this paper. Table I illustrates the problem geometries and lists the results of the TransLAT and MCNP calculations for the test configurations.

The first test configuration is a simple 3-region pin cell that is easily solved with 2-D and 3-D methods. This problem provides an excellent cross comparison of the different transport methods in the TransLAT software. Table I shows that both TransLAT 2-D solvers produce excellent agreement with MCNP. The 3-D calculation shows a slight underprediction of the eigenvalue by 0.0030 delta K.

The second test configuration is a column of fuel comprised of two simple pin cells. This test configuration is functionally identical to the first test configuration, except that a column of pin cells are modeled to further test the axial integration capability of the TransLAT 3-D solver. Consistent with the single pin cell problem, the TransLAT 3-D solver under-predicts the eigenvalue by 0.0022 delta K.

The third test configuration is a column of pin cells similar to the second test problem, except the top pin cell is replaced with a burnable absorber pin cell. This configuration poses a more challenging calculation of the TransLAT 3-D solver. The resulting zebra-style fuel configuration is run with both the TransLAT 3-D solver and MCNP. Table I shows that TransLAT over-predicts the eigenvalue by 0.0022 delta K.

## CONCLUSIONS

The excellent agreement between the TransLAT 2-D results and MCNP provides confidence that the modeling approach and cross sections are properly implemented in TransLAT. The 3-D results provide a strong indication that the 3-D deterministic methods are capable of producing very accurate results as well. Testing will continue with TransLAT to evaluate more complex problems.

1. "TransLAT 3-D Lattice Physics Software, TransFX Computer Software Manuals," TWE-TFX-001 (June 2001).
2. "MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C," LA-13709-M, J. F. Briesmeister, Ed. (Apr. 2000).
3. J. T. West, M. B. Emmett, "MARS: A Multiple Array System Using Combinatorial Geometry," Oak Ridge National Lab., Radiation Shielding Information Center (Dec. 1980).
4. M. L. Williams, R. Raharjo, "Space-Dependent Resonance Self-Shielding," *Nucl. Sci. Eng.*, **126**, 19 (1997).
5. C. H. Potze, S. P. Baker, D. B. Jones, "Dancoff Calculations for PBMR Fuel Using TransLAT 3-D Lattice Code," *Trans. Am. Nucl. Soc.*, **85**, 113 (2001).

## 9. Preliminary Assessment of the Ease of Detection of Attempts at Dual Use of a Pebble Bed Reactor, A. M. Ougouag, H. D. Gougar (INEEL)

### INTRODUCTION

The Pebble Bed Reactor (PBR) concept is receiving emphatic renewed interest. In particular, an international consortium [1] is intent on developing and deploying such a reactor within a few years, with the ultimate goal of international commercialization and deployment of large numbers in developing countries and elsewhere. This optimistic business assessment stems from the numerous inherently and passively safe features of the design. Modular design allows high technology fabrication to be shifted to centralized locations with deployment in a low technology market. The routine recirculation of the fuel pebbles and the on-line defueling and refueling of these reactors raises questions about their potential use as production facilities for weapons materials. However, this feature also allows the reactors to operate with very little excess reactivity. In this paper we demonstrate that the dual use of a PBR (simultaneous production of power and weapons materials) would be easily and promptly detected.

### METHODOLOGY

The PEBBED code [2] computes directly the asymptotic (equilibrium) fuel-loading pattern of a PBR, given the fresh fuel composition. This asymptotic pattern is that which is established well after (>3 years) the initial loading and persists for the remainder of the operating life of the reactor. The pattern and its properties are highly predictable. Presumably the result of extensive optimization, it is expected that reactor owners will stick to it. Departures from this pattern could be viewed as suspicious and as possible attempts at diversion of fuel for dual use. Any departure from the pattern will result in noticeable changes in fresh fuel requirements, power production, and/or discharge isotopics. All three attributes could easily be monitored via an instituted safeguards regime and via spent fuel repurchase. As continuous burnup of discharged pebbles is part of the fuel management policy, the information on the isotopics could also be made available on-line or via the transmission of recorded data sets to the safeguards authority. Uninterrupted fuel supply would be contingent upon acceptable reactor use.

The PBR owner is assumed to be a low technology country without front-end fuel cycle facilities (i.e., enrichment capability) and thus dependent on a supplier country for its fresh fuel needs. The supplier country is party to a non-proliferation regime and agrees to enforce safeguards on its fuel customers. Either the spent fuel is reclaimed, or information on discharged pebbles average isotopics is required. Finally, it is assumed that for economic reasons the on-hand fresh fuel inventory of the PBR owner is maintained as low as practical. In this paper, we assume that after the initial loading the fuel supplier periodically provides ninety days of fresh fuel to the PBR owner, just prior to stock exhaustion.

The PEBBED code is first used to estimate the fresh fuel requirements of a PBR operated following the asymptotic pattern with no attempt at dual use. The code is also used to estimate the fresh fuel requirements of a similar reactor operated with dual use intent. The modeled legitimate reactor is loosely based on the Kraftwerk Union HTR Modul 200, with a 10.0-m core height and a 3.0-m diameter. Graphite reflectors surround the core. The void space between the top of the pebble bed and the top reflector is about 80 cm. The fresh fuel pebbles contain 7 g of uranium enriched to 7.8%. They travel through the core with a mean velocity of 15 cm/day. The core produces 200 MWt of power. In the illicit use cases, target pebbles containing natural uranium (NU) are assumed inserted into the core in the proportions of 0.1% and 0.4% of the overall fuel mix, respectively. The 0.4% content is a physical limit corresponding to the highest number of NU pebbles that can be incorporated into the core while retaining the same critical multiplication factor via the addition of supplementary fresh fuel pebbles. This hypothetical limit corresponds to the plenum above the pebble bed being filled. It cannot be achieved in practice because there is no mechanical means for filling the plenum uniformly to its top, and it would be precluded from acceptance because of its hindrance of coolant flow. Nevertheless, this model provides an upper assessment of the highest Pu-239 rate production possible with this reactor. The 0.1% NU pebbles loading was chosen arbitrarily with the goal of dissimulating the dual use. Reactivity is maintained by the addition of

about 18 cm of fuel mix. The PEBBED code explicitly models the two types of pebbles and assumes different circulation patterns for each. The regular fuel is recirculated a sufficient number of times to achieve the normal nominal burnup. The NU pebbles are circulated once then removed (OTTO) in order to maximize the Pu-239 quality. The results from the PEBBED runs are used to assess the likelihood of detection of dual use attempts.

## RESULTS

Results from PEBBED runs were used to generate the information presented in Table I. In the 0.1% NU case, the fresh fuel supply would run out about 19 days prior to the predicted exhaustion of the on-hand fresh fuel. This will result in an outage of the reactor, an unexpected and highly detectable event. Similarly, if the PBR operator were to lower the power in order to extend operation until the receipt of a new supply of fresh fuel, the nearly 21% power decrease would be noticeable and would require explanation under safeguard agreements. Furthermore, the power decrease would imply lower fuel consumption than originally anticipated and would, under a rational safeguards regime, imply a reduced delivery of fuel at the following supply date. If the performance is repeated, it would eventually lead to increasingly shorter fuel reserves. Such a mode of operation would be uneconomical and politically questionable, as the dual use would become apparent. The illicit patterns of performance would be discovered during the first three months fuel-use-period of their occurrence provided the on-hand supply is replenished to result in stocks meant to last only three months. In contrast, the time required for accumulation of the Pu-239 is very long: 92 years with continuous operation (unlimited fuel supply) and as high as 118 years if fuel shipments are restricted to the requirements of power production. In the 0.4% NU case, the detection would occur after only four days of operation as the fuel supply would be exhausted. The accumulation time would be 23 years (continuous) or 492 years (intermittent).

The last data entry line in Table I shows the residual U-235 content of the discharged fuel pebbles for each case. Although

TABLE I  
Prediction of Fuel Cycle Needs for Three PBR Operation Modes\*

	Regular PBR Core	PBR Core with 0.1% NU Pebbles	PBR Core with 0.4% NU Pebbles
Number of pebbles in core	382 979	389 872	413 617
Fraction of NU pebbles	0	0.001	0.004
Core Height (m)	10.0	10.18	10.80
Pebble transit speed (cm/day)	15	15	15
Transit time (days)	67	68	72
Daily discharge (mix)	5 745	5 745	5 745
NU pebbles in daily discharge	0	6	23
Number of passes (regular pebbles)	17	17	17
Number of passes (NU pebbles)	NA	1	1
Regular pebbles in daily discharge	5 745	5 739	5 722
Daily fresh fuel requirement (number of pebbles)	338	338	337
Re-supply required for 90 days operation	30 413	30 383	30 291
Number of extra required regular fuel pebbles at initial loading	0	6 504	28 984
Number of days fuel supply will be short	0	19	86
Pu-239 content of one discharged NU pebble (mg)	NA	26	26
Estimated number of NU pebbles needed for one weapon (5000g)	NA	192 160	191 278
Time to accumulation (years, continuous operation)	NA	92	23
Time to accumulation (years, interrupted operation)	NA	118	492
Residual U-235 content of discharged fuel (mg/pebble)	251.9	251.7	251.0

\*Numbers of pebbles, days, and years are rounded to integers.

the differences appear small, they are well within the detection limits of modern assay methods. Therefore, the discharge isotopics could also provide an effective tool for detecting attempts at dual use. However, this application will require the prior establishment of a database for legitimate discharge isotopics based on measurements, thus eliminating the error in prediction that can arise from the uncertainty in cross section data.

**CONCLUSIONS**

It is clear that the PBR is a poor tool for production of Pu-239 in all circumstances, even if a continuous fresh fuel supply is assured. Indeed the lowest accumulation period of 23 years for a single device cannot be construed as the basis for a successful proliferation program. Furthermore, any attempt at dual use would be detected promptly and long before the significant accumula-

tion of prohibited materials. Detection would occur within the first three months of illicit use in both cases considered. The results presented here apply to a hypothetical reactor similar in many of its features to the HTR-Modul 200. The models assumed random circulation. The method should be applied to other reactor designs with a comprehensive examination of recirculation patterns.

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1. D. R. Nichols, "Status of the Pebble Bed Modular Reactor," *Nucl. Energy*, **39**, 4, 231 (2000).
2. W. K. Terry, H. D. Gougar, A. M. Ougouag, "Deterministic Method for Fuel Cycle Analysis in Pebble-Bed Reactors," *Trans. Am. Nucl. Soc.*, **83**, 278 (2000).