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<sup>1</sup>Japan Atomic Energy Agency

<sup>2</sup>Jozef Stefan Institute

<sup>3</sup>Senior Reactor Physics Consultant

**March 2009**

**Idaho National Laboratory  
Idaho Falls, Idaho 83415**

**<http://www.inl.gov>**

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**EVALUATION OF THE START-UP CORE PHYSICS TESTS AT  
JAPAN'S HIGH TEMPERATURE ENGINEERING TEST REACTOR  
(FULLY-LOADED CORE)**

**Evaluator**

**John D. Bess**  
**Idaho National Laboratory**

**Nozomu Fujimoto**  
**Japan Atomic Energy Agency**

**Internal Reviewers**  
**Barbara H. Dolphin**  
**Idaho National Laboratory**

**Independent Reviewers**

**Luka Snoj**  
**Jozef Stefan Institute**

**Atsushi Zukeran**  
**Senior Reactor Physics Consultant**

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEFF-KIN-RRATE

## Status of Compilation / Evaluation / Peer Review

Section 1	Compiled	Independent Review	Working Group Review	Approved
1.0 DETAILED DESCRIPTION	YES	YES	YES	YES
1.1 Description of the Critical and / or Subcritical Configuration	YES	YES	YES	YES
1.2 Description of Buckling and Extrapolation Length Measurements	NA	NA	NA	NA
1.3 Description of Spectral Characteristics Measurements	NA	NA	NA	NA
1.4 Description of Reactivity Effects Measurements	YES	YES	YES	YES
1.5 Description of Reactivity Coefficient Measurements	YES	YES	YES	YES
1.6 Description of Kinetics Measurements	YES	YES	YES	YES
1.7 Description of Reaction-Rate Distribution Measurements	YES	YES	YES	YES
1.8 Description of Power Distribution Measurements	NA	NA	NA	NA
1.9 Description of Isotopic Measurements	NA	NA	NA	NA
1.10 Description of Other Miscellaneous Types of Measurements	NA	NA	NA	NA
Section 2	Evaluated	Independent Review	Working Group Review	Approved
2.0 EVALUATION OF EXPERIMENTAL DATA	YES	YES	YES	YES
2.1 Evaluation of Critical and / or Subcritical Configuration Data	YES	YES	YES	YES
2.2 Evaluation of Buckling and Extrapolation Length Data	NA	NA	NA	NA
2.3 Evaluation of Spectral Characteristics Data	NA	NA	NA	NA
2.4 Evaluation of Reactivity Effects Data	NA	NA	NA	NA
2.5 Evaluation of Reactivity Coefficient Data	NA	NA	NA	NA
2.6 Evaluation of Kinetics Measurements Data	NA	NA	NA	NA
2.7 Evaluation of Reaction Rate Distributions	NA	NA	NA	NA
2.8 Evaluation of Power Distribution Data	NA	NA	NA	NA
2.9 Evaluation of Isotopic Measurements	NA	NA	NA	NA
2.10 Evaluation of Other Miscellaneous Types of Measurements	NA	NA	NA	NA



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<b>Section 3</b>	<b>Compiled</b>	<b>Independent Review</b>	<b>Working Group Review</b>	<b>Approved</b>
3.0 BENCHMARK SPECIFICATIONS	YES	YES	YES	YES
3.1 Benchmark-Model Specifications for Critical and / or Subcritical Measurements	YES	YES	YES	YES
3.2 Benchmark-Model Specifications for Buckling and Extrapolation Length Measurements	NA	NA	NA	NA
3.3 Benchmark-Model Specifications for Spectral Characteristics Measurements	NA	NA	NA	NA
3.4 Benchmark-Model Specifications for Reactivity Effects Measurements	NA	NA	NA	NA
3.5 Benchmark-Model Specifications for Reactivity Coefficient Measurements	NA	NA	NA	NA
3.6 Benchmark-Model Specifications for Kinetics Measurements	NA	NA	NA	NA
3.7 Benchmark-Model Specifications for Reaction-Rate Distribution Measurements	NA	NA	NA	NA
3.8 Benchmark-Model Specifications for Power Distribution Measurements	NA	NA	NA	NA
3.9 Benchmark-Model Specifications for Isotopic Measurements	NA	NA	NA	NA
3.10 Benchmark-Model Specifications of Other Miscellaneous Types of Measurements	NA	NA	NA	NA
<b>Section 4</b>	<b>Compiled</b>	<b>Independent Review</b>	<b>Working Group Review</b>	<b>Approved</b>
4.0 RESULTS OF SAMPLE CALCULATIONS	YES	YES	YES	YES
4.1 Results of Calculations of the Critical or Subcritical Configurations	YES	YES	YES	YES
4.2 Results of Buckling and Extrapolation Length Calculations	NA	NA	NA	NA
4.3 Results of Spectral Characteristics Calculations	NA	NA	NA	NA
4.4 Results of Reactivity Effect Calculations	NA	NA	NA	NA
4.5 Results of Reactivity Coefficient Calculations	NA	NA	NA	NA
4.6 Results of Kinetics Parameter Calculations	NA	NA	NA	NA
4.7 Results of Reaction-Rate Distribution Calculations	NA	NA	NA	NA
4.8 Results of Power Distribution Calculations	NA	NA	NA	NA
4.9 Results of Isotopic Calculations	NA	NA	NA	NA
4.10 Results of Calculations of Other Miscellaneous Types of Measurements	NA	NA	NA	NA
<b>Section 5</b>	<b>Compiled</b>	<b>Independent Review</b>	<b>Working Group Review</b>	<b>Approved</b>
5.0 REFERENCES	YES	YES	YES	YES
Appendix A: Computer Codes, Cross Sections, and Typical Input Listings	YES	YES	YES	YES

## EVALUATION OF THE START-UP CORE PHYSICS TESTS AT JAPAN'S HIGH TEMPERATURE ENGINEERING TEST REACTOR (FULLY-LOADED CORE)

**IDENTIFICATION NUMBER:** HTTR-GCR-RESR-001

CRIT-SUB-REAC-COEF-KIN-RRATE

**KEY WORDS:** critical configuration, graphite-moderated, graphite-reflected, helium coolant, HTTR, low enriched uranium (LEU), pin-in-block fuel, prismatic fuel, thermal reactor, TRISO, uranium oxide fuel

### SUMMARY INFORMATION

#### 1.0 DETAILED DESCRIPTION

The High Temperature Engineering Test Reactor (HTTR) of the Japan Atomic Energy Agency (JAEA)<sup>a</sup> is a 30 MWth, graphite-moderated, helium-cooled reactor that was constructed with the objectives to establish and upgrade the technological basis for advanced high-temperature gas-cooled reactors (HTGRs) as well as to conduct various irradiation tests for innovative high-temperature research. The core size of the HTTR represents about one-half of that of future HTGRs, and the high excess reactivity of the HTTR, necessary for compensation of temperature, xenon, and burnup effects during power operations, is similar to that of future HTGRs. During the start-up core physics tests of the HTTR, various annular cores were formed to provide experimental data for verification of design codes for future HTGRs (Ref. 1, p. 310).

The Japanese government approved construction of the HTTR in the 1989 fiscal year budget; construction began at the Oarai Research Establishment in March 1991 and was completed May 1996. Fuel loading began July 1, 1998, from the core periphery. The first criticality was attained with an annular core on November 10, 1998 at 14:18, followed by a series of start-up core physics tests (Figure 1.1) until a fully-loaded core was developed on December 16, 1998. Criticality tests were carried out into January 1999. The first full power operation with an average core outlet temperature of 850°C was completed on December 7, 2001, and operational licensing of the HTTR was approved on March 6, 2002. The HTTR attained high temperature operation at 950 °C in April 19, 2004. After a series of safety demonstration tests, it will be used as the heat source in a hydrogen production system by 2015. A short history of the HTTR project is shown in Figure 1.2, and a plan for the future of the HTTR project is shown in Figure 1.3 (Ref. 3, pp. 12-14).

A very high temperature reactor critical assembly (VHTRC) was constructed<sup>b</sup> to provide reactor physics experiments to study the detailed neutronic characteristics of the HTTR core design (Ref. 15, pp. 21-23). Seven different core designs were employed (Ref. 3, p. 19). Further summary of the validation efforts using the VHTRC is discussed elsewhere.<sup>c,d,a</sup>

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<sup>a</sup> JAEA was established on October 1, 2005 by integrating the Japan Atomic Energy Research Institute (JAERI) and the Japan Nuclear Cycle Development Institute (JNC).

<sup>b</sup> H. Yasuda, et al., "Construction of VHTRC (Very High Temperature Reactor Critical Assembly)," JAERI-1305, Japan Atomic Energy Research Institute, Tokaimura, 1986.

<sup>c</sup> I. Murata, K. Yamashita, S. Maruyama, R. Shindo, N. Fujimoto, Y. Sudo, and T. Nakata, "Evaluation of Local Power Distribution with Fine-Mesh Core Model for High Temperature Engineering Test Reactor (HTTR)," *J. Nucl. Sci. Tech.*, **31**(1): 62-72 (January 1994).

<sup>d</sup> N. Fujimoto, N. Nojiri, and K. Yamashita, "Validation of the Nuclear Design Code System for the HTTR using the Criticality Assembly VHTRC," *Nucl. Eng. Des.*, **233**: 155-162 (2004).

Hot zero-power critical,<sup>b</sup> Rise-to-power,<sup>c</sup> irradiation tests,<sup>d</sup> and safety demonstration testing<sup>e,f</sup> have also been performed with the HTTR, representing additional means for computational validation efforts. Power tests were performed in steps from 0 to 30 MW, with various tests performed at each step to confirm core characteristics, thermal-hydraulic properties, and radiation shielding (Ref. 13, p. 284). The high-temperature test operation at 950 °C represented the fifth and final phase of the rise-to-power tests.<sup>g</sup> The safety tests demonstrated inherent safety features of the HTTR such as slow temperature response during abnormal events due to the large heat capacity of the core and the negative reactivity feedback.<sup>h</sup> A more detailed summary of the rise-to-power with safety tests is shown in Table 1.1.

The experimental benchmark performed and currently evaluated in this report pertains to the data available for the fully-loaded core critical from the initial six isothermal, annular and fully-loaded, core critical measurements performed at the HTTR. Additional experiments performed during the start-up physics tests include reactivity worth (excess reactivity, shutdown margin, and control rods), isothermal temperature coefficient, and reaction rate measurements. A subcritical evaluation of the fully-loaded core was also recorded.

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<sup>a</sup> F. Akino, M. Takeuchi, T. Ono, and Y. Kaneko, "Experimental Verification and Analysis of Neutron Streaming Effect through Void Holes for Control Rod Insertion in HTTR," *J. Nucl. Sci. Tech.*, **34**(2): 185-192 (February 1997).

<sup>b</sup> J. C. Kuijper, X. Raepsaet, J. B. M. de Haas, W. von Lensa, U. Ohlig, H.-J. Ruetten, H. Brockmann, F. Damian, F. Dolci, W. Bernnat, J. Oppe, J. L. Kloosterman, N. Cerullo, G. Lomonaco, A. Negrini, J. Magill, and R. Seiler, "HTGR Reactor Physics and Fuel Cycle Studies," *Nucl. Eng. Des.*, **236**: 615-634 (2006).

<sup>c</sup> S. Nakagawa, Y. Tachibana, K. Takamatsu, S. Ueta, and S. Hanawa, "Performance Test of HTTR," *Nucl. Eng. Des.*, **233**: 291-300 (2004).

<sup>d</sup> T. Shibata, T. Kikuchi, S. Miyamoto, and K. Ogura, "Assessment of Irradiation Temperature Stability of the First Irradiation Test Rig in the HTTR," *Nucl. Eng. Des.*, **223**: 133-143 (2003).

<sup>e</sup> Y. Tachibana, S. Nakagawa, T. Takeda, A. Saikusa, T. Furusawa, K. Takamatsu, K. Sawa, and T. Iyoku, "Plan for the First Phase of Safety Demonstration Tests of the High Temperature Engineering Test Reactor (HTTR)," *Nucl. Eng. Des.*, **224**: 179-197 (2003).

<sup>f</sup> S. Nakagawa, K. Takamatsu, Y. Tachibana, N. Sakaba, and T. Iyoku, "Safety Demonstration Tests using High Temperature Engineering Test Reactor," *Nucl. Eng. Des.*, **233**: 301-308 (2004).

<sup>g</sup> S. Fujikawa, H. Hayashi, T. Nakazawa, K. Kawasaki, T. Iyoku, S. Nakagawa, and N. Sakaba, "Achievement of Reactor-Outlet Coolant Temperature of 950 °C in HTTR," *J. Nucl. Sci. Tech.*, **41**(12): 1245-1254 (December 2004).

<sup>h</sup> S. Nakagawa, D. Tochio, K. Takamatsu, M. Goro, and T. Takeda, "Improvement of Analysis Technology for High Temperature Gas-Cooled Reactor by using Data Obtained in High Temperature Engineering Test Reactor," *J. Power Energy Syst.*, **2**(1): 83-91 (2008).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE

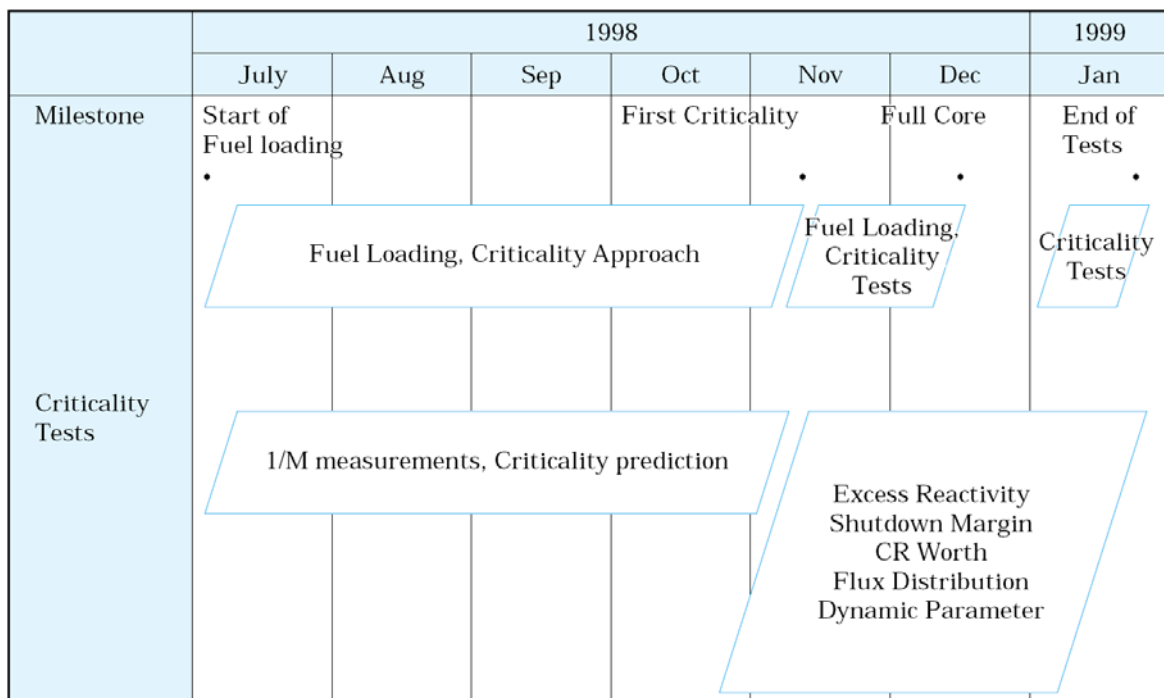
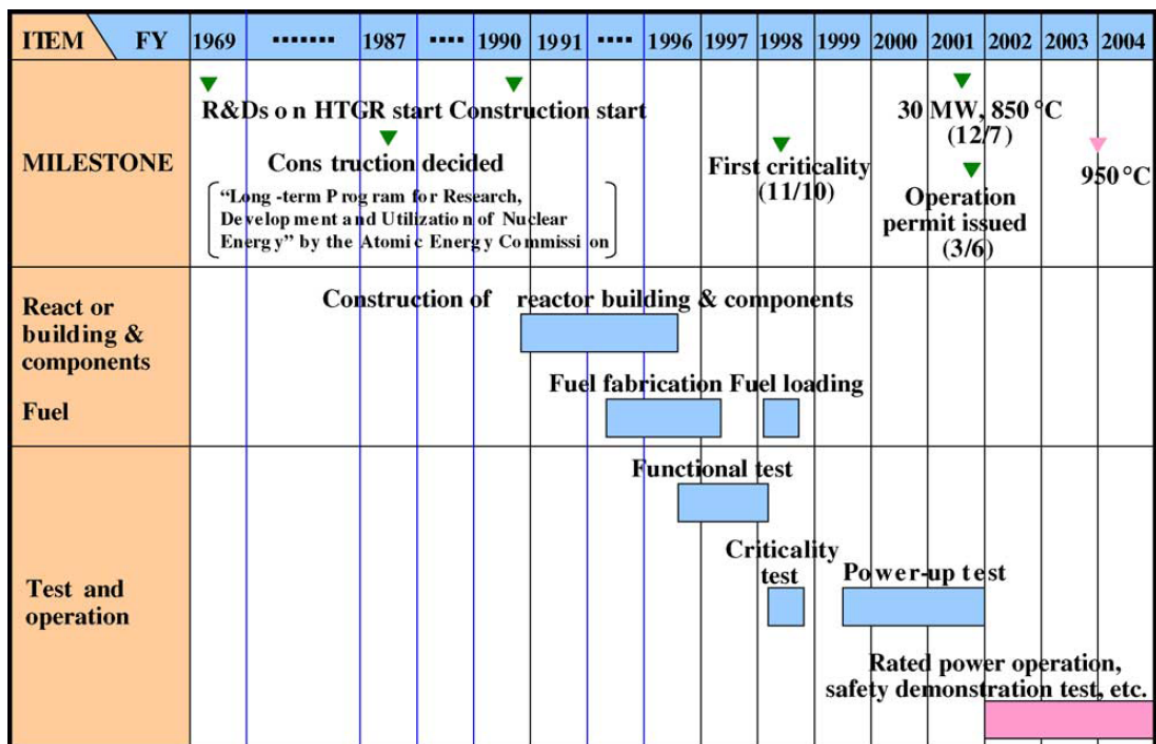
Figure 1.1. Progress of Start-Up Core Physics Tests.<sup>a</sup>

Figure 1.2. History of the HTTR Project (Ref. 3, p. 12).

<sup>a</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

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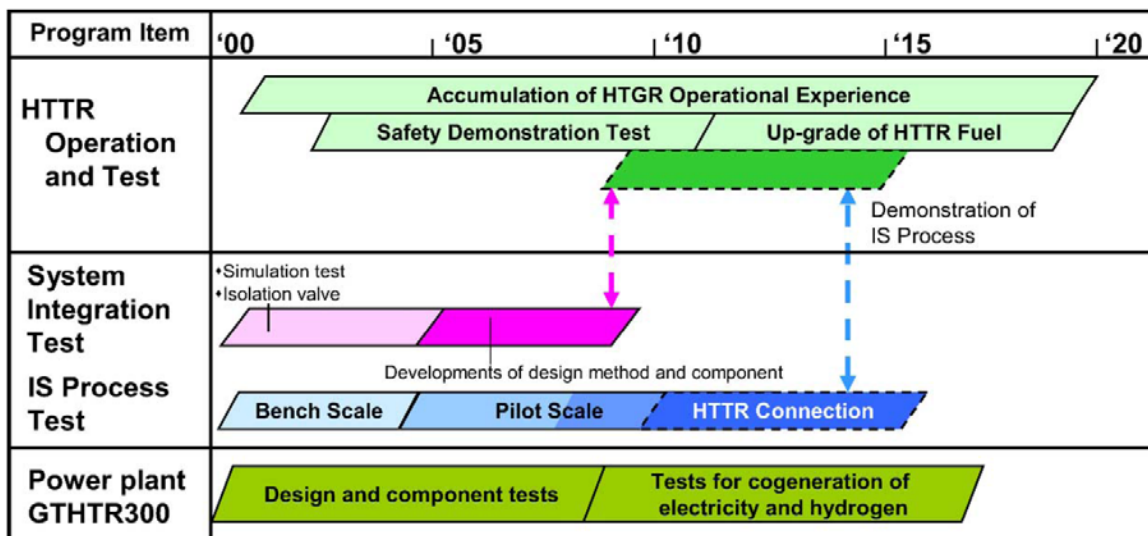


Figure 1.3. Present and Future Plans of the HTTR Project (Ref. 3, p. 13).

Table 1.1. HTTR Rise-to-Power and Safety Operation History.<sup>a</sup>

Operation name	Operation mode	Max. reactor thermal power (MW)	Duration	Operation time (MWd)	Remarks
1st phase rise-to-power test (PT-1)	● ▲	9	23 April–8 May 2000	193	a)
	◆	9	11–26 May 2000		
	◆	9	30 May–6 June 2000		
2nd phase rise-to-power test (PT-2)	● ▲	20	3–8 July 2000	63	b)
	◆	20	29 January–12 February 2001	423	
	▲	20	16 February–1 March 2001		
3rd phase rise-to-power test (PT-3)	■ ◆	20	14 April–7 May 2001	689	
	◆	20	11–16 May 2001		
	▲	30	21 May–8 June 2001		
4th phase rise-to-power test (PT-4)	● ▲	30	23 October–14 December 2001	1,293	Achievement of 850°C, 30 MW Operation permit issued.
	◆	30	25 January–6 March 2002	984	
1st operation cycle (RP/RS-1)	● ◆	30	30 May–17 June 2002	370	Safety demonstration test
	▲	9	21 June–1 July 2002	92	
2nd operation cycle (RS-2)	● ▲	30	5 February–14 March 2003	658	Safety demonstration test
3rd operation cycle (RP-3)	● ◆	20	16–21 May 2003	72	a)
4th operation cycle (RS-4)	● ▲	9	8–11 August 2003	25	Safety demonstration test
5th operation cycle (RS-5)	● ▲	30	27 January–25 February 2004	549	Safety demonstration test
	◆	18	29 February–5 March 2004		
5th phase rise-to-power test (PT-5)	■ ▲	30	31 March–1 May 2004	679	Achievement of 950°C, 30 MW Operation permit issued.
	◆	30	2 June–2 July 2004	626	
Total operation time (include other operations, i.e. core physics tests)				6,717	

●: Rated operation mode, ■: High-temperature test operation mode, ▲: Single-loaded operation mode, ◆: Parallel-loaded operation mode

a) Automatically reactor shut-down caused by a signal of PPWC flow-rate low

b) Automatically reactor shut-down caused by a loss of off-site electric power by a thunderbolt

<sup>a</sup> S. Fujikawa, H. Hayashi, T. Nakazawa, K. Kawasaki, T. Iyoku, S. Nakagawa, and N. Sakaba, "Achievement of Reactor-Outlet Coolant Temperature of 950 °C in HTTR," *J. Nucl. Sci. Tech.*, **41**(12): 1245-1254 (December 2004).

## 1.1 Description of the Critical and / or Subcritical Configuration

### 1.1.1 Overview of Experiment

The initial start-up core physics tests for the High Temperature Engineering Test Reactor were performed between the months of July 1998 and January 1999. The HTTR facility is at the Oarai Research and Development Center of the JAEA.

Core physics tests, including start-up and power-up operations, were planned in order to ensure core performance and reactor safety of an HTGR. Results were obtained for a core burn-up of up to 5 GWD/t. During these tests, the critical approach, the excess reactivity, the shutdown margin, the control rod worth, the reactivity coefficient, the neutron flux distribution, and the power distribution were measured and compared with the calculated results (Ref. 13, p. 283).

First criticality was attained on November 10, 1998, and the core was fully loaded in December 1998. During the fuel loading period, various tests were conducted under cold clean conditions as the start-up core physics tests. During these tests, the power was limited to below 30 W and the helium pressure was 1 bar. The tests for the critical approach, excess reactivity, shutdown margin, neutron flux distribution and control rod worth were conducted (Ref. 13, p. 284). Elsewhere the pressure is listed as 1 atmosphere (Ref. 1, p. 312). One atmosphere is equal to  $1.01325 \times 10^5$  Pa, or 1.0325 bar.

Of the initial six isothermal, annular and fully-loaded, core critical measurements that were performed, only the fully-loaded core was evaluated in this benchmark analysis. It was determined to have a total uncertainty between -0.606 and +0.703 %  $\Delta k$ . Dominant uncertainties are the impurities in the IG-110 graphite blocks and impurities in the PGX graphite blocks. Biases could not be determined for this experiment and the benchmark model  $k_{\text{eff}}$  value is greater than  $3\sigma$  from the unbiased expected  $k_{\text{eff}}$  for most of the cross-section libraries analyzed.

Currently the calculations using the benchmark model calculate a  $k_{\text{eff}}$  2 % greater than the experimental benchmark value. It is believed that a systematic bias causes this difference from the expected benchmark value. It is currently difficult to obtain the necessary information to improve the confidence in the benchmark model; the necessary data is controlled because the configuration of the HTTR analyzed in this benchmark is the same core currently in operation. Once this information is made available, the HTTR benchmark can be adjusted as appropriate.

Subcritical and warm critical configurations were recorded but have not been evaluated for benchmark quality.

### 1.1.2 Geometry of the Experiment Configuration and Measurement Procedure

Section 1.1.2.1 discusses the HTTR site and facility as gathered from the references. Section 1.1.2.2 contains information regarding many of the internal mechanisms of the HTTR. Both of these sections are useful in understanding the overall HTTR description. However, most of the information pertinent to this benchmark evaluation of the reactor core can be found in Section 1.1.2.3 through Section 1.1.2.6.

#### 1.1.2.1 Facility Description

##### Site and Facility

The HTTR is located on JAEA's Oarai Research and Development Center site, which is approximately 100 kilometers north of the Tokyo metropolitan area, near the Pacific Ocean (Figures 1.4 and 1.5). The plant area is 200 m by 300 m in size (Figure 1.6). The 48 m by 50 m reactor building is situated in the

central area of the plant with an 80-m high exhaust stack to the north, as shown in Figure 1.7 (Ref. 2, p. 5). Air cooling towers are located on the roof of the reactor building (Ref. 15, p.9).

Construction of the HTTR took approximately 7 years (Figure 1.8).

Typically important Japanese buildings must be built on base rock material. The HTTR reactor building, however, is supported on an Ishizaki-layer, which is a sand layer formed during the Quaternary era. Boring and seismic surveys demonstrated strength and support of this layer equivalent to that of base rock; therefore, the seismic safety of the facility was recognized.<sup>a</sup> Significant seismic design guidelines were incorporated into the HTTR such that all anticipated earthquakes would not cause a severe accident.<sup>b</sup>

The reactor building is comprised of five levels, three of which are underground, as shown in Figures 1.9 and 1.10.

The major design specifications of the HTTR are provided in Table 1.2.

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<sup>a</sup> K. Kunitomi and S. Shiozawa, "Safety Design," *Nucl. Eng. Des.*, **233**: 45-58 (2004).

<sup>b</sup> K. Iigaki and S. Hanawa, "Seismic Design," *Nucl. Eng. Des.*, **233**: 59-70 (2004).



Figure 1.4. Location of JAEA's Oarai Research and Development Center.



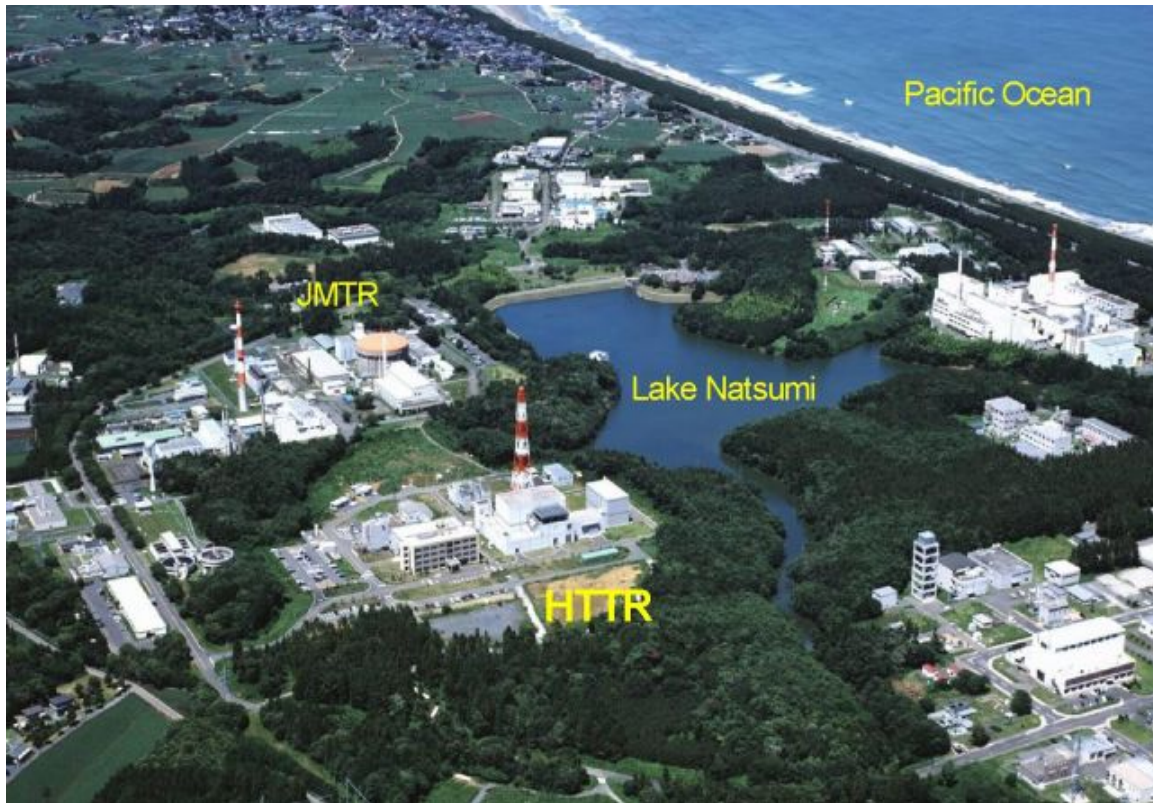


Figure 1.5. Aerial View of the HTTR Facility (<http://httr.jaea.go.jp>).

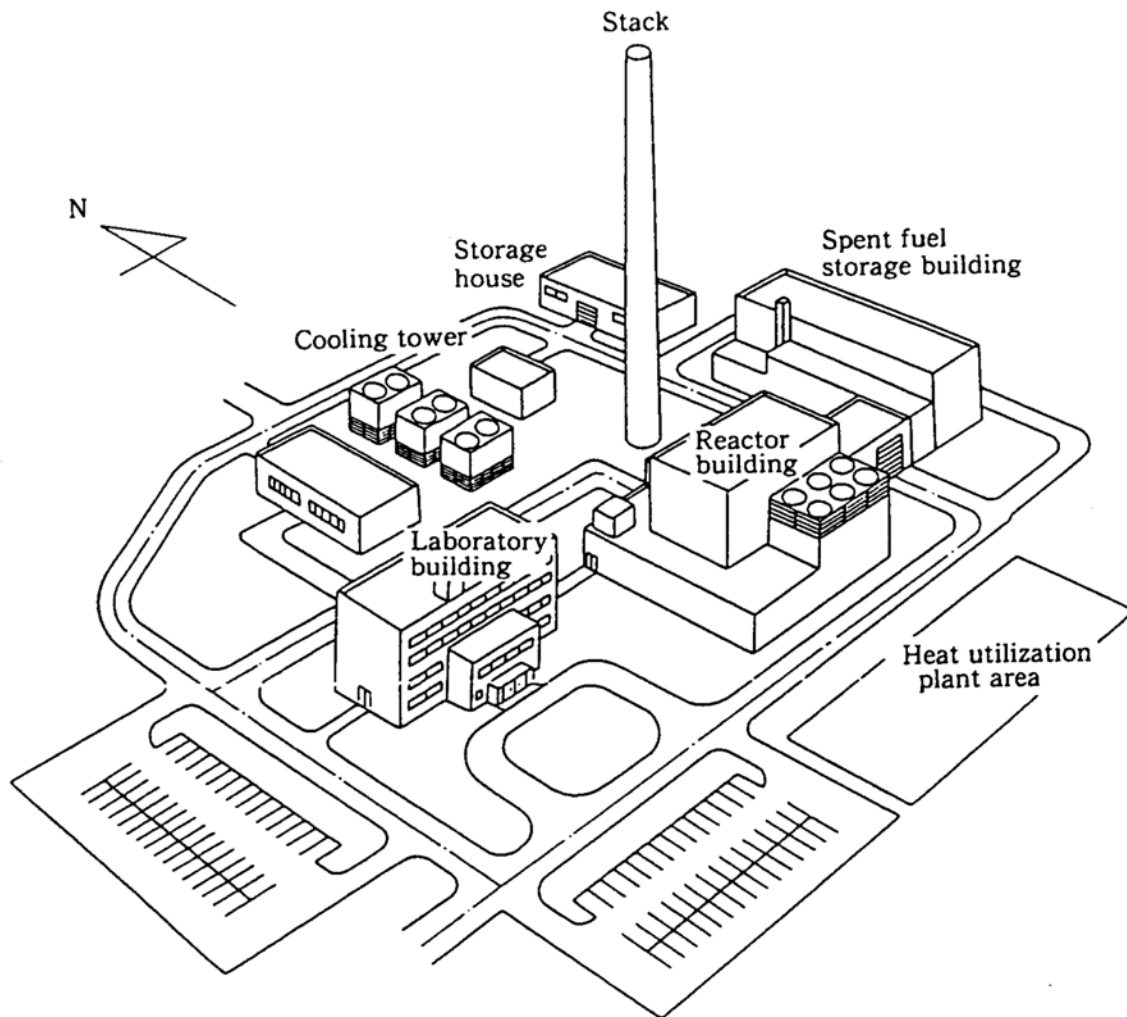


Figure 1.6. Isomeric View of HTTR Plant ([Ref. 15, p. 10](#)).

Figure 1.7. HTTR Building Exterior (<http://httr.jaea.go.jp>).

ITEM \ FY*	1990	1991	1992	1993	1994	1995	1996	1997	1998
MILESTONE	Construction start ▽				Electricity receive ▽			Fueling ▽ Criticality ▽	
Safety review	□								
Approval of design and construction method	□	□	□	□	□				
Site renovation	□								
Excavation of reactor building		□							
Reactor building			□	□	□	□	□	□	□
Containment vessel			□						
Cooling system					□	□	□	□	□
Reactor pressure vessel and core internals					□	□	□	□	□
Fuel fabrication				□	□	□	□	□	□

\*1 Fiscal year of Japan starts in April and ends in next March

Figure 1.8. Construction Schedule of the HTTR (Ref. 15, p. 6).



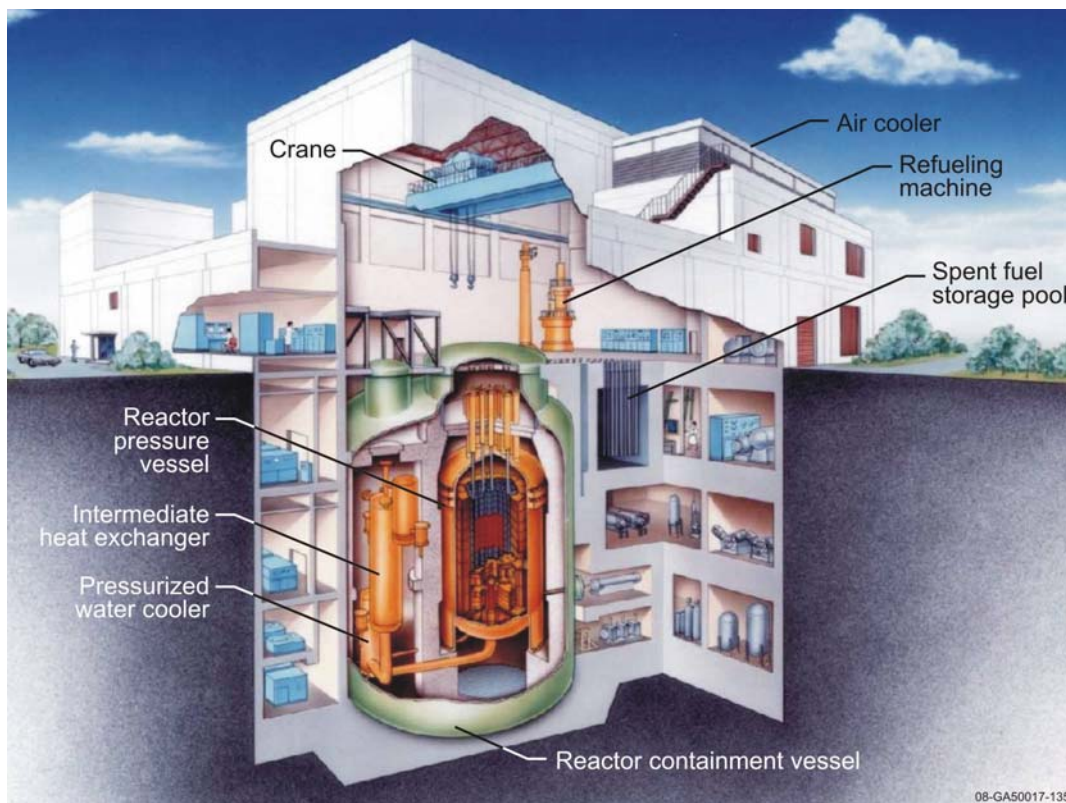


Figure 1.9. HTTR Building Interior.

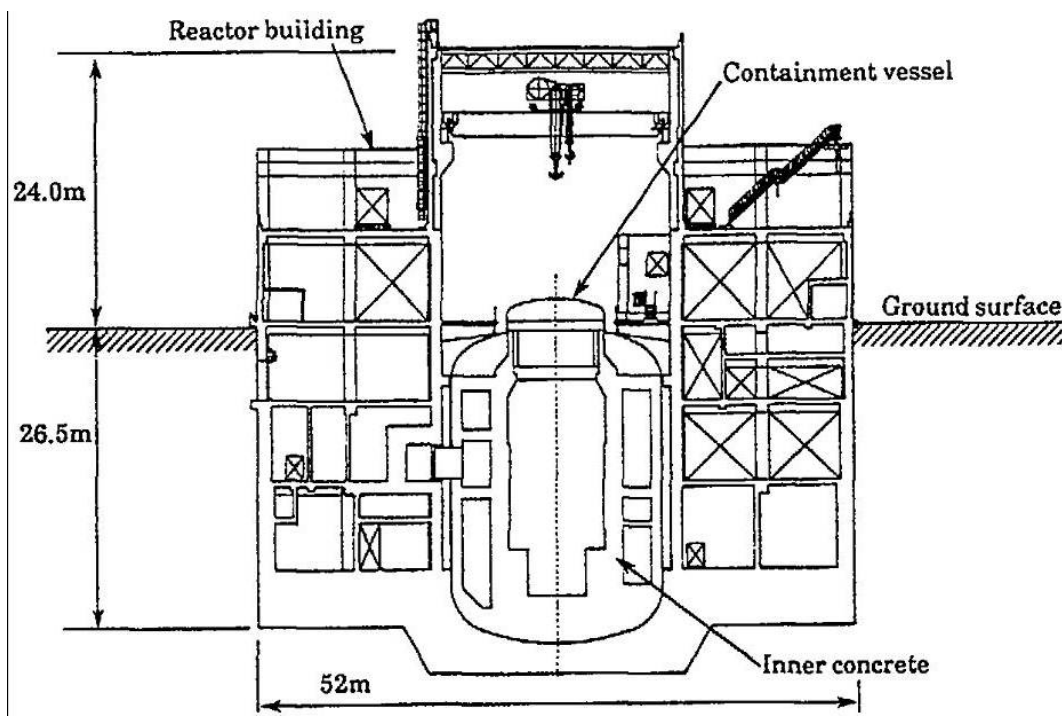


Figure 1.10. The HTTR Structure.<sup>a</sup>

<sup>a</sup> F. Yagishita, H. Suzuki, and Y. Yamagishi, "Soil-Structure Interaction Analysis of HTTR Building by a Simplified Model," *Proc. Int. Working Group on Gas-Cooled Reactors (IWGGCR-22)*, Gif-sur-Yvette, France (November 14-16, 1989).

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Table 1.2. Major Design Specifications of the HTTR (Ref. 1, p. 311, and Ref. 2, p. 9).

Thermal Power	30 MW
Outlet Coolant Temperature	850/950 °C
Inlet Coolant Temperature	395 °C
Primary Coolant Pressure	4 MPa
Core Structure	Graphite
Equivalent Core Diameter	2.3 m
Effective Core Height	2.9 m
Average Power Density	2.5 W/cm <sup>3</sup>
Fuel	UO <sub>2</sub>
Enrichment	3 to 10 wt. %
	6 wt. % (average)
Fuel Type	Pin-in-Block Type
	Coated Fuel Particles
Burn-Up Period (EFPD)	660 days
Fuel Block	Graphite Block
Coolant Material	Helium Gas
Flow of Direction in Core	Downward
Reflector Thickness	
Top	1.16 m
Side	0.99 m
Bottom	1.16 m
Number of Fuel Assemblies	150
Number of Fuel Columns	30
Number of Pairs of Control Rods	
In Core	7
In Reflector	9
Plant Lifetime	20 years

## **Containment Structure**

The containment structures of the HTTR consist of the reactor containment vessel (C/V), the service area, and the emergency air purification system. The reactor containment vessel is relatively small, so as to minimize air ingress and reactivity with graphite in the event of a rupture of the primary pressure boundary. The carbon steel vessel has 2800 m<sup>3</sup> of free volume, with a height of 30.3 m and a diameter of 18.5 m. It has a personnel air lock, maintenance hatch, and refueling hatch, as shown in Figure 1.11. Elastic plugging material is installed between the reactor containment vessel and the base mat of the reactor building so as to absorb any thermal expansion in the event of an accident. Other penetrations in the C/V are shown in Figure 1.12. The C/V is designed to be leak tight with a leakage rate of less than 0.1 % of the total free volume per day at room temperature, air atmosphere, and 0.9 times its maximum pressure. Specifications of the C/V are tabulated in Table 1.3 ([Ref. 10, pp. 135-137](#)).

The containment vessel is installed in the center of the reactor building. Although the containment vessel is not needed for retention of an accidental release of fission products, it was required to meet Japanese safety design guidelines for light water nuclear power plants, as HTGR safety features had yet to be developed. Above the reactor pressure vessel (RPV) is a fueling hatch attached to the C/V. Some of the compartments surrounding the C/V function as confinement or serve areas. Major components such as the primary cooling system and RPV are contained within the C/V ([Ref. 2, p. 6](#)).

Mitsubishi Heavy Industries was in charge of the design, manufacture, and installation of the main HTTR equipment, such as the reactor containment vessel and the main cooling system ([Ref. 14, p. 386](#)).

A service area is provided (Figure 1.13) that surrounds the reactor C/V and includes the fuel handling and storage systems and the primary helium purification system. The service area is kept at a negative pressure during accidents by an emergency air purification system, which operates under the isolated reactor containment vessel condition. During normal operations, the service area is kept at a negative pressure by the ventilation and air conditioning systems ([Ref. 10, pp. 140-141](#)).

Because helium gas is employed as the coolant in the HTTR, a shielding effect is not expected as would be in a conventional LWR. Therefore, a thick shielding structure is necessary for the HTTR. A schematic of the shielding arrangement is shown in Figure 1.14.

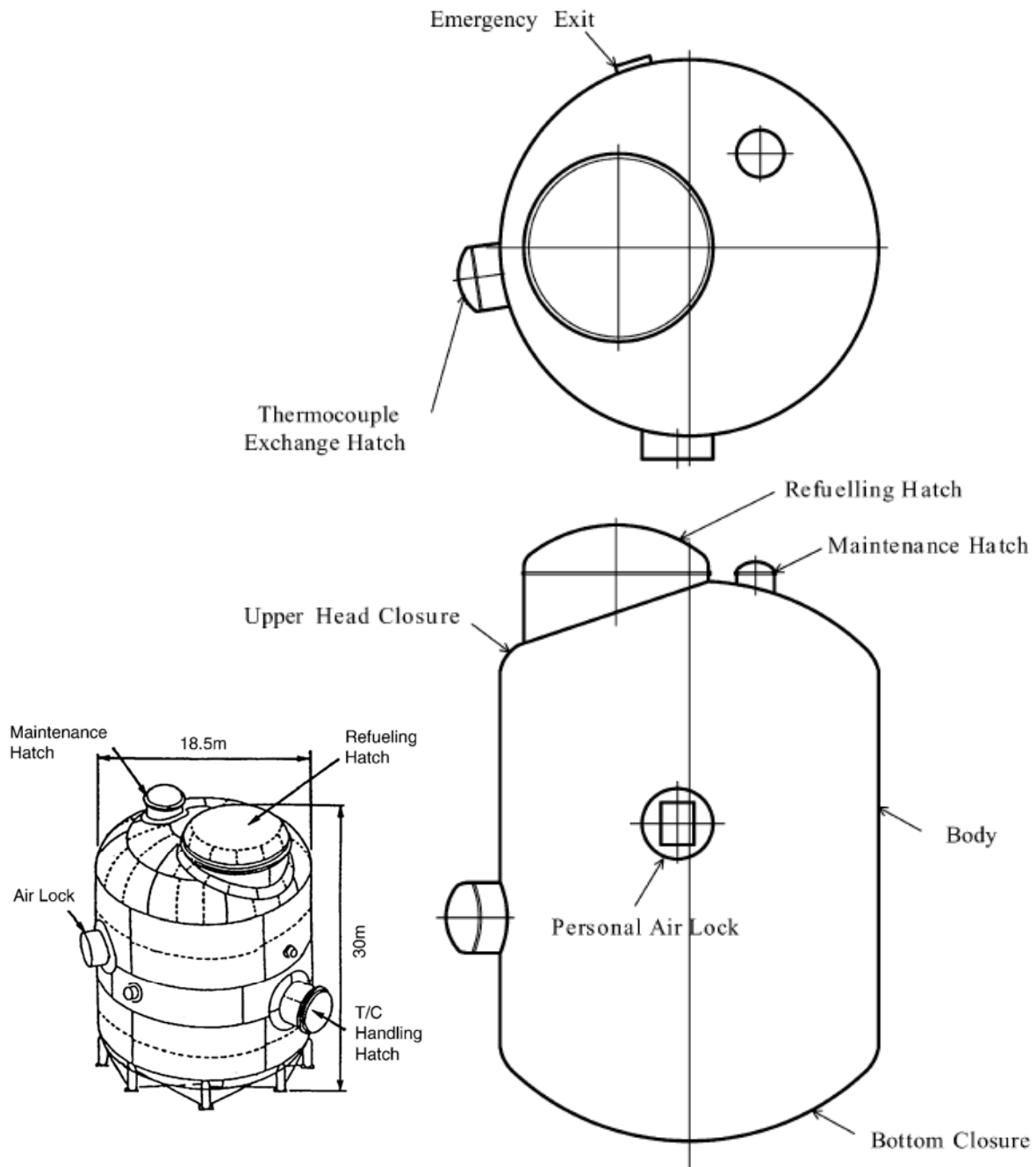


Figure 1.11. Reactor Containment Vessel Major Penetrations (Ref. 10, p. 136, and Ref. 14, p. 387).

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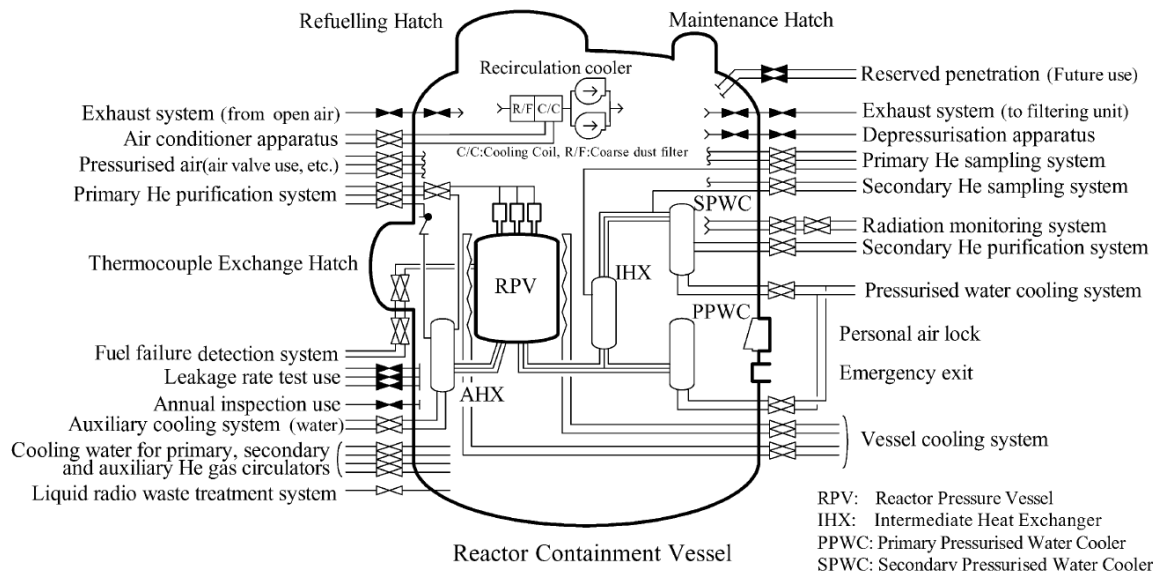


Figure 1.12. Reactor Containment Vessel Penetrations (Ref. 10, p. 137).

Table 1.3. Specifications of the Reactor Containment Vessel (Ref. 10, p. 137).

Containment Type	Steel
Maximum Pressure in Service (MPa)	0.4
Maximum Temperature in Service (°C)	150
Major Size	
Inner Diameter (m)	18.5
Overall Height (m)	30.3
Body Thickness (mm)	30
Upper Head Closure Thickness (mm)	38
Refueling Hatch Diameter (m)	8.5
Maintenance Hatch Diameter (m)	2.4
Personal Air Lock Diameter (m)	2.5
Free Volume (m <sup>3</sup> )	2800
Material	Carbon Steel
Leakage Rate	Less than 0.1 %/d at the room temperature and 0.9 times the maximum pressure of 0.4 MPa



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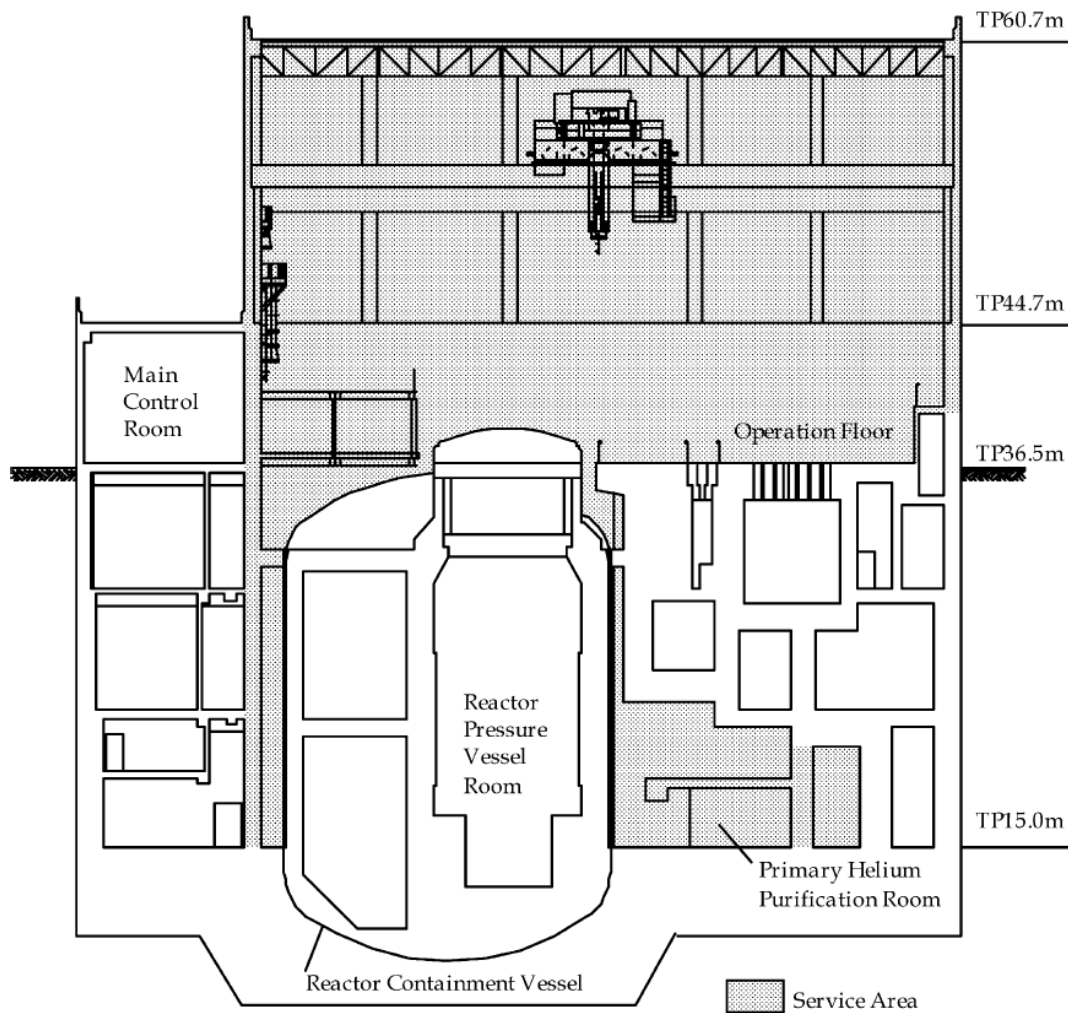


Figure 1.13. HTTR Service Area (Ref. 10, p. 142).

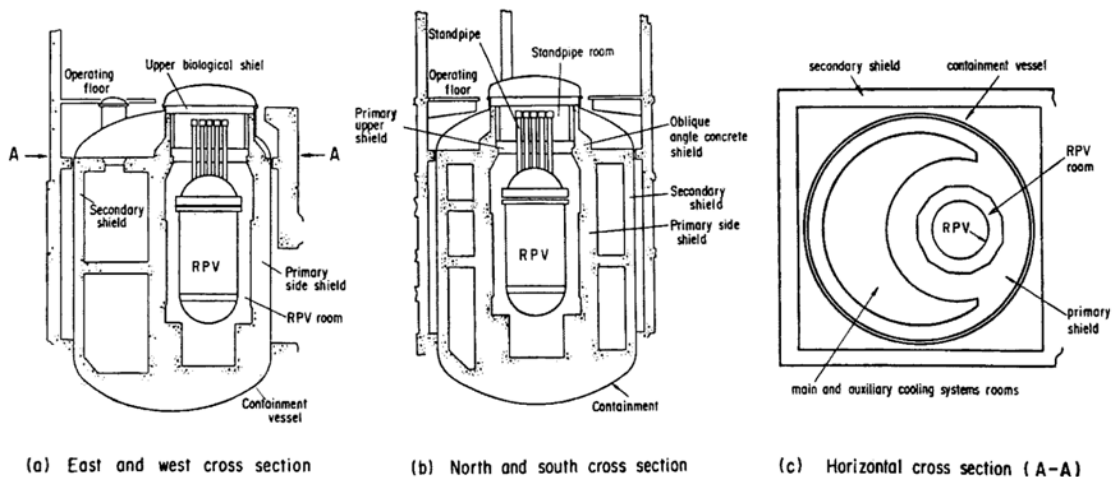


Figure 1.14. Schematic Arrangement of Shields in HTTR.<sup>a</sup>

<sup>a</sup> I. Murata, R. Shindo, and S. Shozawa, "Importance Determination Method for Geometry Splitting with Russian Roulette in Monte Carlo Calculations of Thick and Complicated Core Shielding Structure," *J. Nucl. Sci. Tech.*, 32(10): 971-980 (October 1995).

**Reactor Pressure Vessel**

The reactor pressure vessel is formed as a vertical cylinder with hemispherical top and bottom head closures, and 31 standpipes. The top head closure is bolted to a flange of the vessel cylinder. The standpipes are used for manipulation of control rods, irradiation experiments, and instrumentation. A diagram of the RPV and various reactor internals is shown in Figure 1.15 ([Ref. 2, p. 6](#)). The RPV is 13.2 m in height and 5.5 m in diameter ([Ref. 3, p. 17](#)).

A schematic of just the RPV is shown in Figure 1.16, with a close-up view of the upper structure in Figure 1.17. The RPV contains thirty-one standpipes that are welded to the top head dome. A thermal shield is installed on the inner surface of the top head closure to prevent high temperature exposure in accidents. The RPV is supported by a skirt, stabilizers, and a standpipe support beam. The skirt is welded to the outside of the bottom dome. The stabilizers surround the outside of the RPV cylinder and are supported by the side concrete. The standpipe support beam is located near the top of the standpipes. A special standpipe-fixing device is located at the top of some of the standpipes to prevent the standpipe internal structure from being ejected in the event of a standpipe break ([Ref. 8, pp. 103-104](#)).

Major specifications of the RPV are shown in Table 1.4. The RPV has a top head which includes standpipes, a top head dome, a top head flange, thermal shields, etc., and a RPV body comprised of a shell flange, shell, three standpipes, a bottom head petal, skirt, a bottom head dome, a support ring, and radial keys. The RPV top head is bolted to the RPV body by 72 stud bolts. There are 31 standpipes: 16 for control rods, five for irradiation tests, three for surveillance tests, three for neutron detection, two for in-service inspection of reactor internals, and two for measurement of temperatures and core differential pressure. These are welded to nozzles on the top head dome. The control standpipes constrain the control rod drive mechanism. The irradiation standpipes are used to install specimens and experimental equipment into the core. Figure 1.18 shows the arrangement of the standpipes on the top head dome and Table 1.5 provides the diameter of the standpipes and penetrating holes. The largest seven standpipes, N1-N7, are also used for refueling. Three other standpipes for hot plenum temperature measurement and fuel failure detection are welded to nozzles on the shell. The skirt supports weight and seismic load of the reactor. Horizontal seismic load is sustained by six stabilizers and a standpipe support beam, as well as the skirt. The support ring supports the vertical load of the core, and the radial keys hold horizontal movement of the core through a core restraint mechanism ([Ref. 8, p. 105-106](#)).

The standpipe closure, which can be handled with the control rod handling machine, is plugged into the standpipe with double rubber D-rings to provide leak tightness. Latches prevent the closure from being pushed out by the coolant pressure. The standpipe-fixing device is made up of snubbers, a support plate, and other devices. If a standpipe breaks off in an accident, the fixing device limits displacement of the top of the standpipe by the snubbers, to prevent the internal structures from being pulled out and to limit the outflow of primary coolant. In normal operation, a standpipe support beam is not in contact with the standpipes, and is free of thermal expansion of the RPV. It restrains an excessive displacement of standpipes induced by an earthquake ([Ref. 8, p. 107](#)).

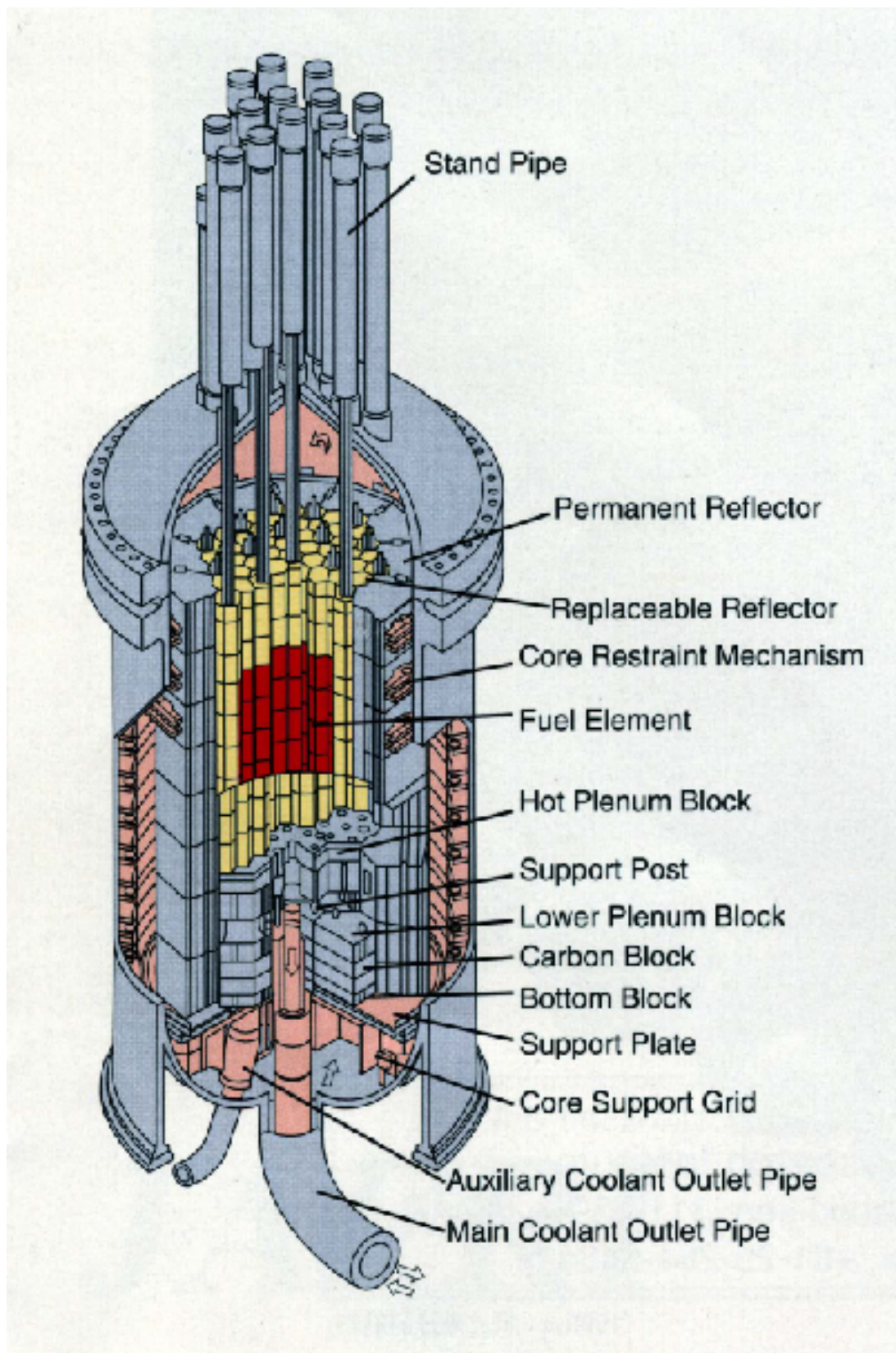


Figure 1.15. HTTR Pressure Vessel and Internals (<http://httr.jaea.go.jp>).



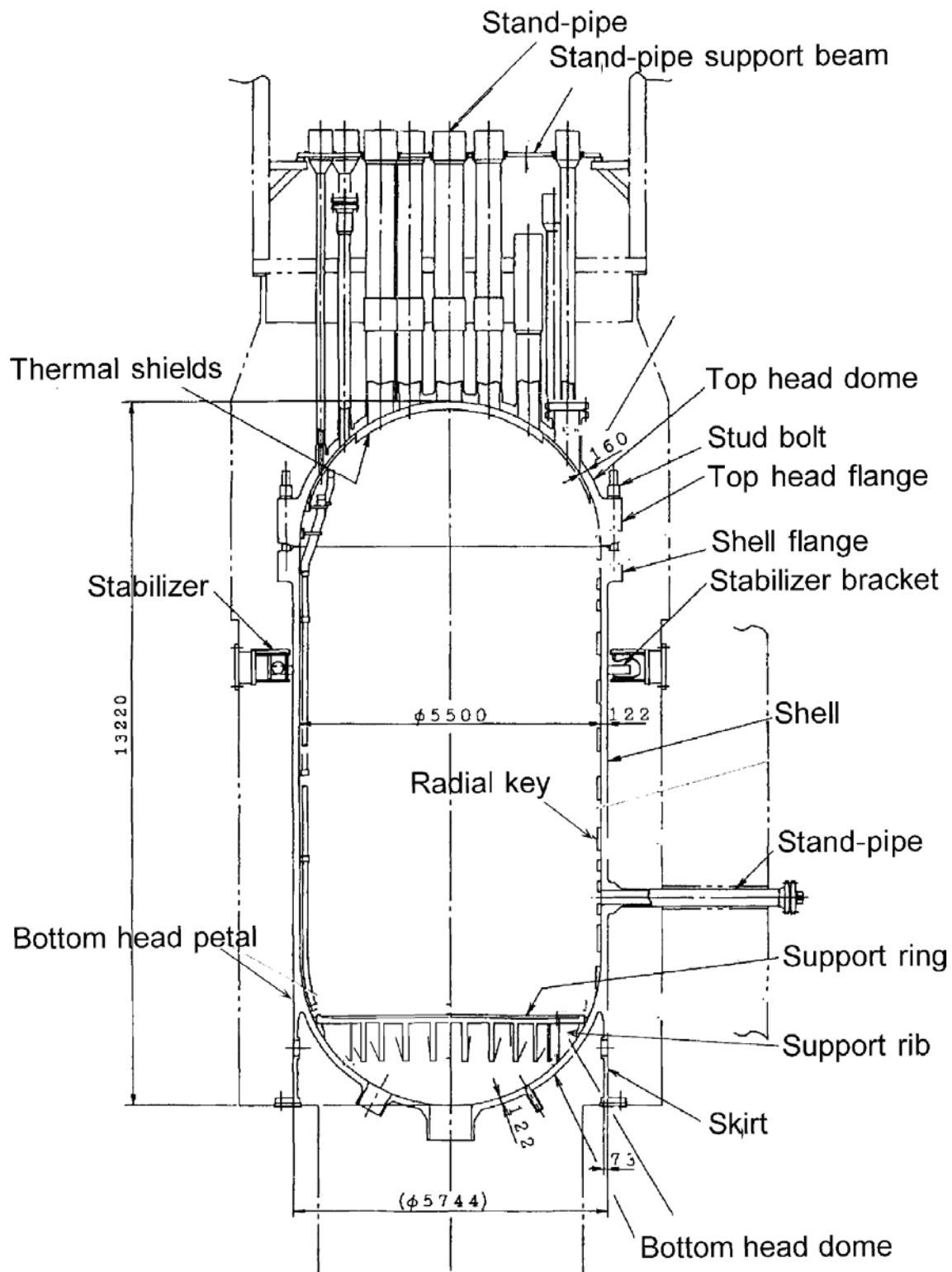
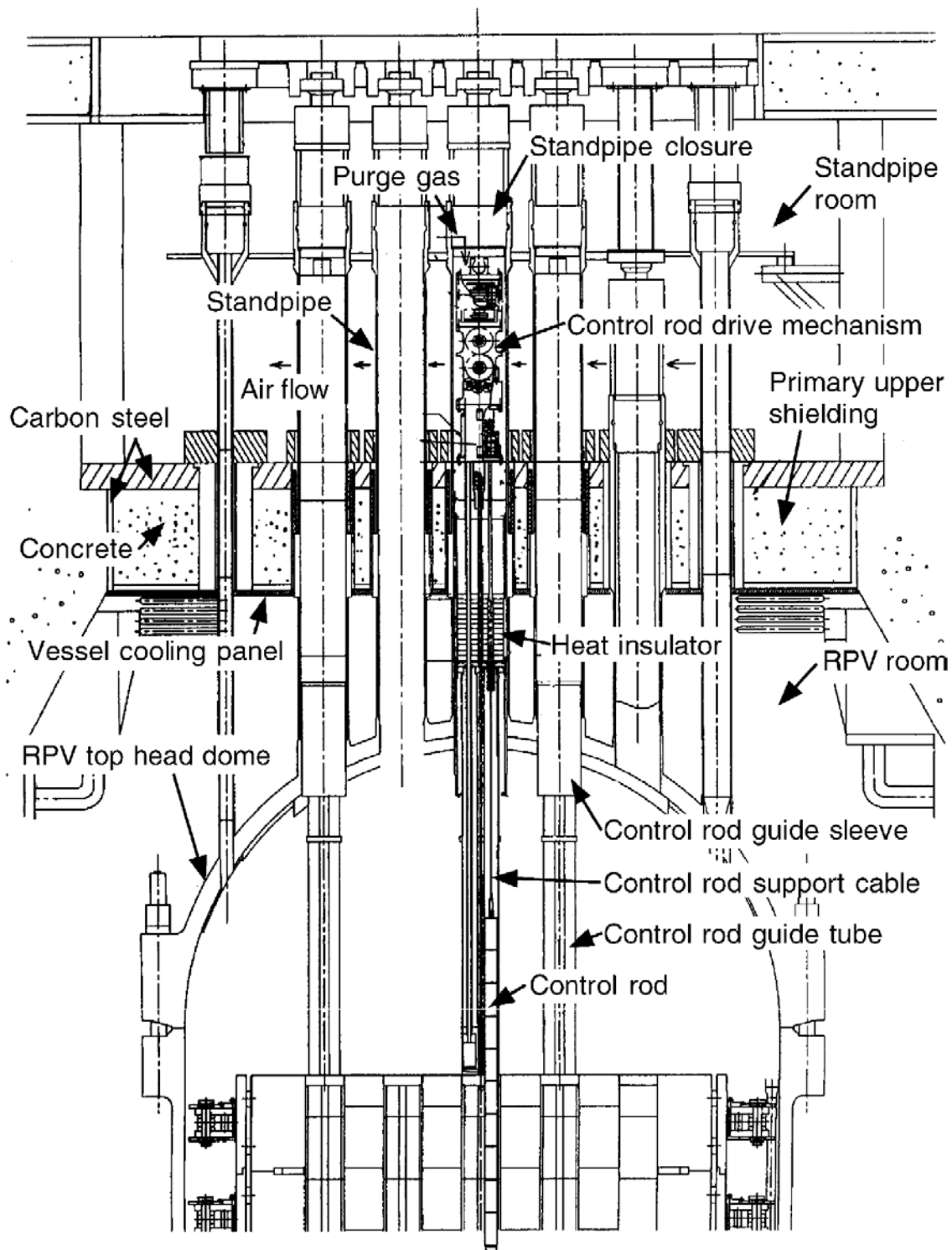


Figure 1.16. Schematic Design of the Reactor Pressure Vessel. (Ref. 8, p. 104).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATEFigure 1.17. Upper HTTR Structure.<sup>a</sup>

<sup>a</sup> N. Fujimoto, Y. Tachibana, A. Saikusa, M. Shinozaki, M. Isozaki, and T. Iyoku, "Experience of HTTR Construction and Operation – Unexpected Incidents," *Nucl. Eng. Des.*, **233**: 273-281 (2004).

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Table 1.4. Major Specifications of the HTTR RPV (Ref. 8, p. 105).

Design Pressure (MPa)	4.7 [gauge]
Design Temperature (°C)	440
Normal Operating Pressure (MPa)	3.9 [gauge]
Inlet Coolant Temperature (°C)	395
Inside Diameter (m)	5.5
Height (m)	13.2
Thickness of Cylindrical Shell and Bottom Head Dome (mm)	122 (minimum)
Thickness of Top Head Dome (mm)	160 (minimum)
Number of Standpipes	34
Material	2-1/4Cr-1Mo steel (normalized and tempered)

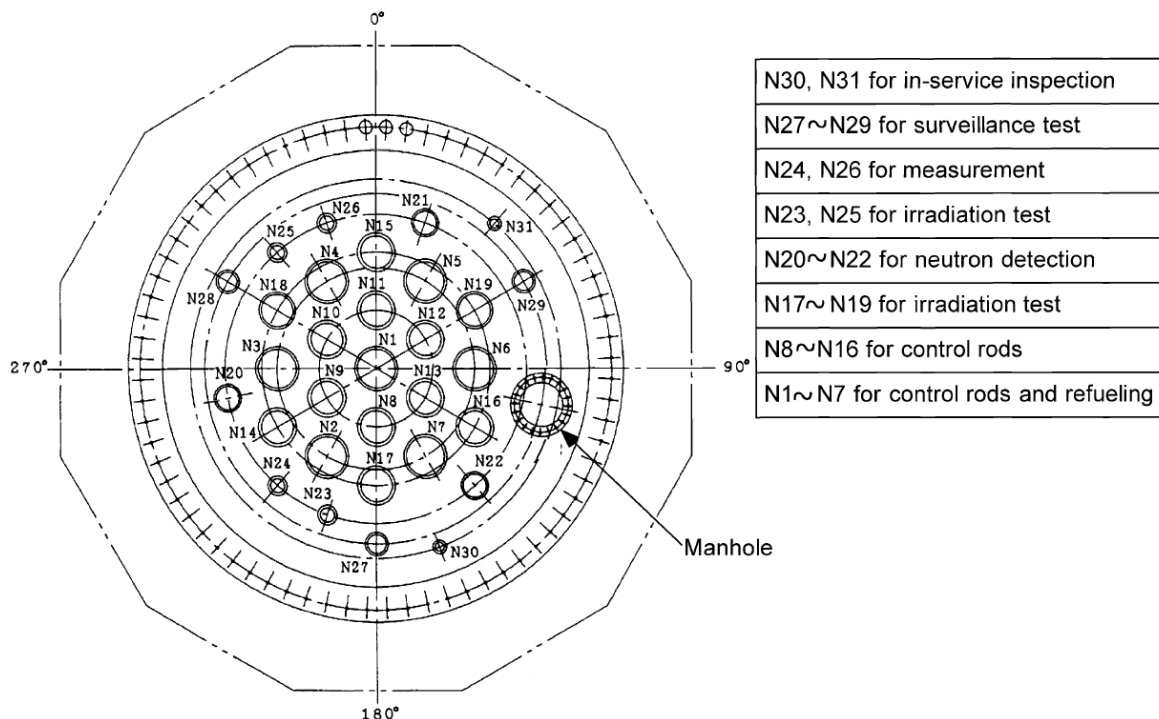


Figure 1.18. Arrangement of Standpipes of the HTTR (Ref. 8, p. 105).

Table 1.5. Diameter of Standpipes and Penetrating Holes in Support Plate. <sup>(a)</sup>

Standpipe No.	Use	Outer Diameter of Standpipe (mm)	Inner Diameter of Standpipe (mm)	Diameter of Penetrating Hole (mm)
1-7	Control Rods	525	485	525
8-16	Control Rods	460	425	460
17-19	Irradiation Test	512 <sup>(b)</sup>	405	640
20-22	Neutron Detection	319 <sup>(b)</sup>	298	614
23, 25	Irradiation Test	216	200	216
24, 26	Reactor Instrumentation	216 <sup>(b)</sup>	200	271
27-29	Surveillance Test	267	249	267
30, 31	In-Service Inspection	140	121	140

(a) T. Takeda and Y. Tachibana, "Indirect Air Cooling Techniques for Control Rod Drives in the High Temperature Engineering Test Reactor," *Nucl. Eng. Des.*, **223**: 25-40 (2003).

(b) Diameter of the standpipe fixing device.

### **Reactor Cooling System**

Hitachi was responsible for the design, manufacture, and installation of key HTTR components such as the reactor pressure vessel, the vessel cooling system, and the auxiliary cooling water system (Ref. 14, p. 380).

The reactor cooling system is comprised of a main cooling system (MCS), an auxiliary cooling system (ACS) and two reactor vessel cooling systems (VCSs), as shown in Figure 1.19. The MCS removes the heat energy from the reactor core during normal operations, while the ACS and VCSs function as engineered safety features to remove residual heat after a reactor scram. The ACS provides force-cooling of the core restraint mechanism to protect it against thermal damage from reactor heat during anticipated operational occurrences (AOOs) and normal operation (Ref. 2, p. 7).

Approximately 97 % of the reactor thermal power is removed by the MCS with a helium flow rate of 12.4 kg/s at rated operation and 10.2 kg/s at the high-temperature test operation. The remaining heat is removed mostly by the VCS. After a reactor scram, the residual core heat is removed by the ACS and/or the VCS.<sup>a</sup>

<sup>a</sup> E. Takada, S. Nakagawa, N. Fujimoto, and D. Tochio, "Core Thermal-Hydraulic Design," *Nucl. Eng. Des.*, **233**: 37-43 (2004).

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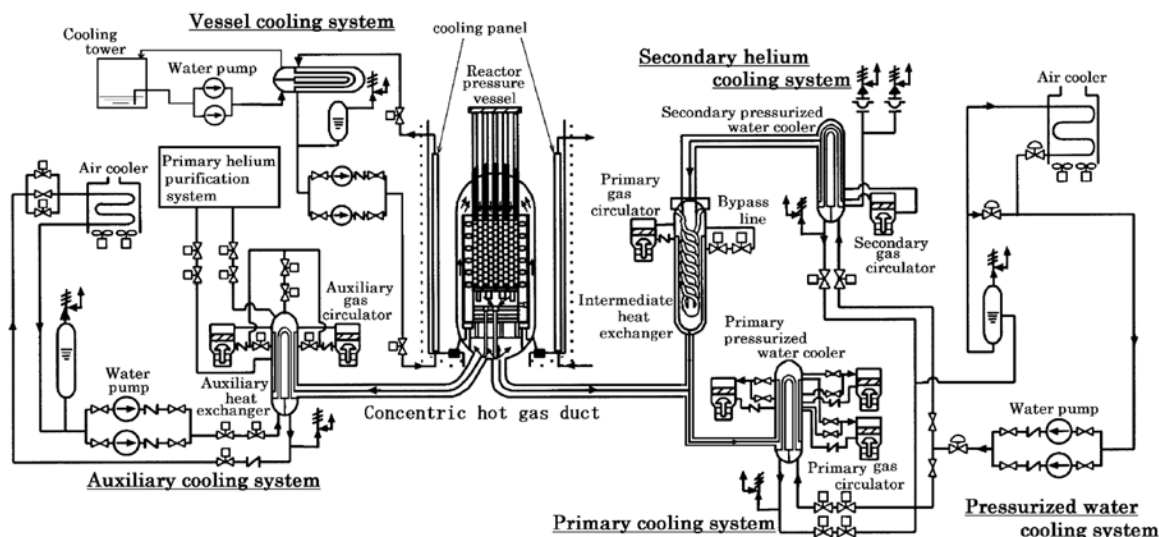


Figure 1.19. Schematic Diagram of HTTR Cooling System (Ref. 9 p. 114).

The MCS consists of an intermediate heat exchanger (IHX), a primary pressurized water cooler (PPWC), a secondary pressurized water cooler (SPWC), and a pressurized water/air cooler. The MCS has two operational modes: single-loaded operation and parallel-loaded operation. The PPWC functions to remove the reactor heat of 30 MW during the single-loaded operation, while during parallel-loaded operation the IHX removes 10 MW and the PPWC removes the other 20 MW. The SPWC serves the function of removing the heat from the IHX. The heat removed by the PPWC and the SPWC is transported through the pressurized water at 2.5 MPa. The pressurized water is then cooled down by the air cooler. In the HTTR reactor plant, the reactor heat is eventually transferred to the atmosphere by the pressurized water/air cooler. During normal operation, the pressure of the secondary helium is controlled to always be 0.1 MPa higher than that of the primary helium at the IHX heat transfer tubes in order to reduce the pressure load on the tubes and to protect against accidental leakage of radioactive materials into the secondary coolant. The water pressure is controlled so that a large amount of water cannot ingress into the core (Ref. 2, p. 8).

The IHX development was led by Toshiba and Ishikawahima-Harima Heavy Industries. Fabrication of the IHX was completed in September 1994 and installed at the HTTR site in February 1995 (Ref. 14, pp. 380-382).

The IHX is a vertical helically-coiled counter-flow type heat exchanger in which primary helium gas flows on the shell side and secondary helium gas in the tube side as shown in Figure 1.20. Table 1.6 has the major specifications of the IHX. Primary helium gas enters the IHX through the inner pipe of the primary concentric hot gas duct. It is deflected under a hot header and discharged around the heat transfer tubes to transfer the heat to the secondary helium cooling system. It flows to the primary gas circulator via the upper outlet nozzle and flows back to the annular space between the inner and outer shells. Secondary helium gas flows downwards in the heat transfer tubes and upwards in the central hot gas duct through the hot header. The inner insulation is installed inside the inner shell to maintain its temperature below 440 °C. The insulation outside and inside the central gas duct restrain the heat transfer so that high efficiency can be obtained. In addition, it also keeps the temperature of the central duct below 940 °C. Primary helium gas is contained only in the primary cooling system because the pressure in the secondary helium cooling system is adjusted somewhat higher than that in the primary cooling system (Ref. 9, pp. 116-117).

A floating hot header with a combination of a central hot gas duct passes through the central space inside the helix bundle so as to minimize constraints of axial and radial thermal expansion of the helically-coiled heat transfer tubes. An assembled tube support allows free thermal expansion of a helix in the



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radial direction. The primary helium enters the IHX through the inner pipe of the primary concentric hot gas duct attached to the bottom of the IHX. It flows up outside the tubes, transferring 10 MW of nuclear heat to the secondary helium, and then flows back to the annular space between the inner and outer shells. A double-walled shell with thermal insulation attached to the inside surface of the inner shell provides reliable separation of the heat resisting and pressure retaining functions. Cold helium flowing through the annulus brings uniform temperature distribution throughout the outer shell, which serves the function of being the pressure retaining member (Ref. 2, pp. 8-9).

The tube support assemblies hold the heat transfer tubes. Both the central hot gas duct and the heat transfer tube support assemblies are hung from the vessel top so that the thermal expansion is not constrained (Ref. 9, p. 117). A photograph of the IHX tube bundles is shown in Figure 1.21.

The IHX has a bypass line, which prevents natural circulation from the reactor core from entering IHX during the single-loaded operation. The forced circulation from the PPWC through the bypass line occurs and keeps the temperature of the outer shell below 430 °C. Primary helium gas flows from the PPWC and enters into the annulus space between the inner and outer shells before flowing inside the inner shell through the bypass line. It then returns to the PPWC through the IHX and the primary concentric hot gas duct. The shutoff valve stops this forced circulation during the parallel-loaded operation and when the ACS is activated for a scram. The inner structures, such as the heat transfer tubes, central hot gas duct, and the hot header, are operated beyond 900 °C (Ref. 9, pp. 117-118).

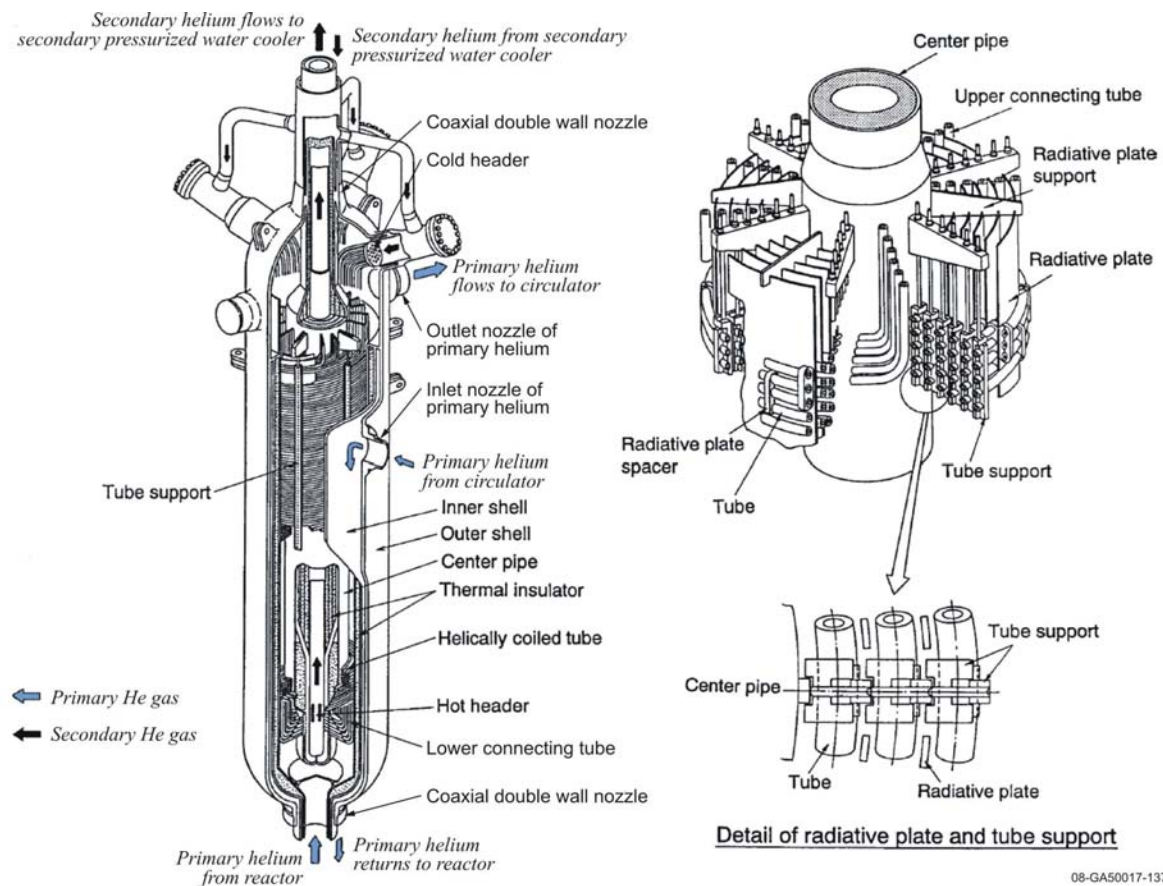


Figure 1.20. The Intermediate Heat Exchanger.

08-GA50017-137

Table 1.6. Major Specifications of the Intermediate Heat Exchanger ([Ref. 9 p. 118](#) and [Ref. 14, p. 382](#)).

Type	Vertical Helically-Coiled Counter Flow	
Design Pressure		
Outer Shell (MPa)	4.7	
Heat Transfer Tube (MPa)	0.29 (differential pressure)	
Design Temperature		
Outer Shell (°C)	430	
Heat Transfer Tube (°C)	955	
Operating Condition	Rated Operation	High Temperature
Flow Rate of Primary Helium Gas (t/h)	14.9 (max)	12.2 (max)
Inlet Temperature of Primary Helium Gas (°C)	850	950
Outlet Temperature of Primary Helium Gas (°C)	395	395
Flow Rate of Secondary Helium Gas (t/h)	12.8	10.8
Inlet Temperature of Secondary Helium Gas (°C)	244	237
Outlet Temperature of Secondary Helium Gas (°C)	175	174
Heat Capacity (MW)	10	
Heat Transfer Tube		
Number	96	
Number of Coil Layers	6	
Pitch (mm)	47	
Outer Diameter (mm)	31.8	
Thickness (mm)	3.5	
Length (m)	30	
Outer Diameter of Shell (m)	1.9	
Total Height (m)	10	
Radiation Plate Thickness (mm)	5	
Material		
Outer and Inner Shell	SCMV4-2NT(2-1/4Cr-1Mo Steel)	
Heat Transfer Tube	Hastelloy XR	
Hot Header and Center Pipe	Hastelloy XR	



Figure 1.21. Completed Tube Bundles of the IHX (Ref. 14 p. 383).

The PPWC is a vertical U-tube heat exchanger (Figure 1.22 and Table 1.7). Hot primary helium gas from the inlet nozzle flows horizontally between the baffle plates, and cools the outside surface of the heat transfer tubes. It flows upwards and turns backwards several times. Then it flows out via the upper or lower outlet nozzles to the primary gas circulators and flows back to the annular space between the inner and outer shell to cool them. Pressurized water of 3.5 MPa is led to each heat transfer tube and heated up by the primary helium gas. Thermal insulation is installed inside the inner shell to maintain its temperature lower than 440 °C. A tubesheet supports the heat transfer tubes (Ref. 9, p. 115).

The heat capacity of the PPWC can be changed from 30 to 20 MW by changing the helium gas flow paths according to the loop operational modes. Primary helium gas flows out through three lower outlet nozzles during parallel-loaded operation and the three upper nozzles during single-loaded operation (Ref. 9, p. 115).

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The maximum temperature difference between the helium gas and pressurized water is approximately 800 °C. To prevent burn out, the flow velocity of the primary helium gas in the high temperature region is maintained lower than the average flow velocity in the PPWC. The flow velocity in the lower temperature region is maintained higher than the average to promote heat transfer. The size of the PPWC thus remains compact ([Ref. 9, pp. 115-116](#)).

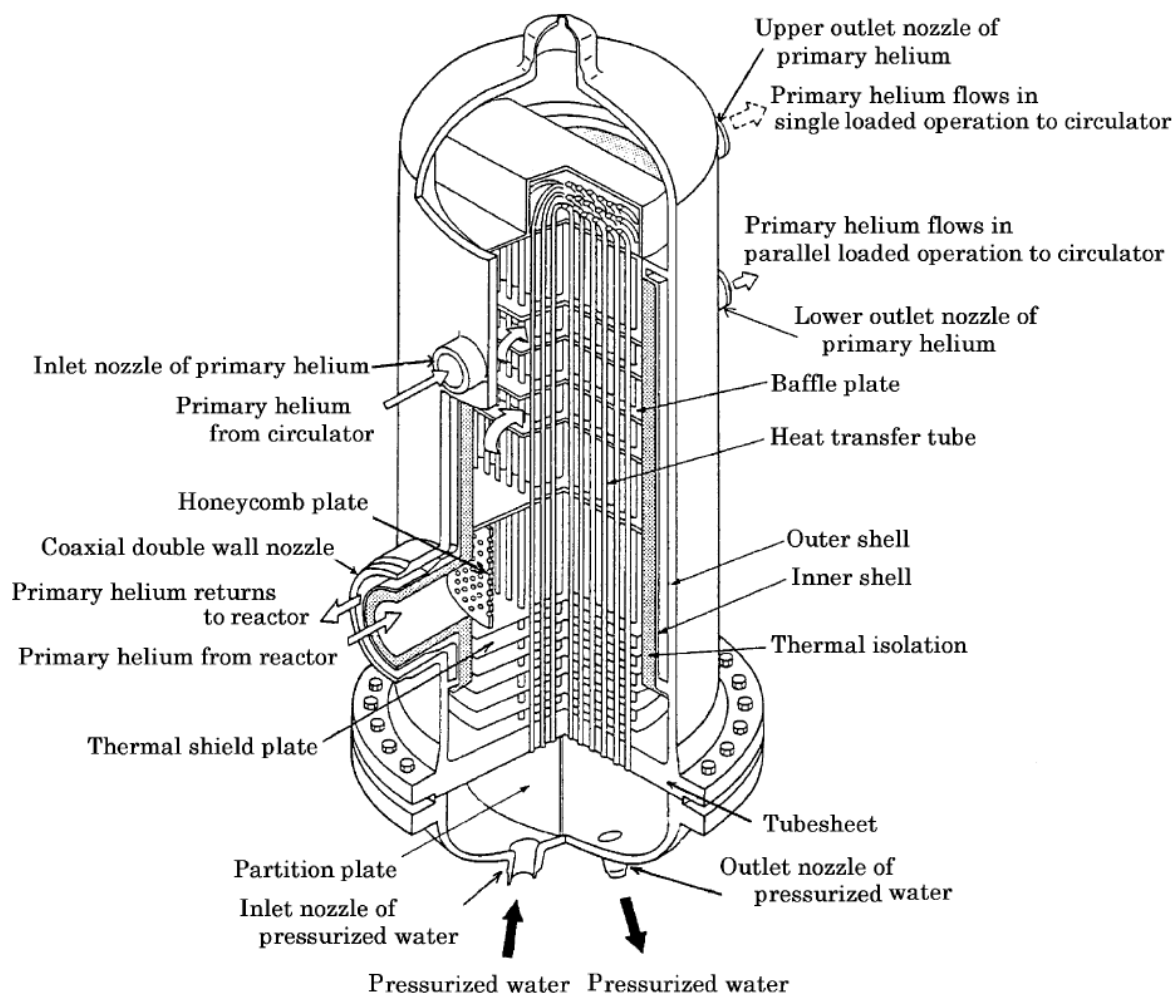


Figure 1.22. The Primary Pressurized Water Cooler ([Ref. 9 p. 115](#)).

Table 1.7. Major Specifications of the Primary Pressurized Water Cooler ([Ref. 9 p. 116](#)).

Type	Vertical U-Bent Tube	
Design Pressure		
Outer Shell (MPa)	4.7	
Heat Transfer Tube (MPa)	4.7	
Design Temperature		
Outer Shell (°C)	430	
Heat Transfer Tube (°C)	380	
Operating Condition	Rated Operation	High Temperature
Flow Rate of Primary Helium Gas (t/h)		
Single Loaded Operation (max)	45.2	37.0
Parallel Loaded Operation (max)	29.7	24.3
Inlet Temperature of Primary Helium Gas (°C)	850	950
Outlet Temperature of Primary Helium Gas (°C)	395	395
Flow Rate of Pressurized Water (t/h)		
Single Loaded Operation	625	618
Parallel Loaded Operation	413	410
Inlet Temperature of Pressurized Water (°C)	135	134
Outlet Temperature of Pressurized Water (°C)	175	174
Heat Capacity (MW)		
Single Loaded Operation	30	
Parallel Loaded Operation	20	
Heat Transfer Tube		
Number	136	
Outer Diameter (mm)	25.4	
Thickness (mm)	2.6	
Length (m)	10	
Outer Diameter of Shell (m)	2.1	
Total Height (m)	7.5	
Material		
Outer and Inner Shell	SCMV4-2NT(2-1/4Cr-1Mo Steel)	
Heat Transfer Tube	SUS321TB	
Tubesheet	SFVA F22B(2-1/4Cr-1Mo Steel)	

The primary gas circulator is a centrifugal, dynamics gas bearing type circulator (Figure 1.23). Table 1.8 contains the major specifications of the circulator. There are three circulators for the PPWC and one for the IHX in the primary cooling system. The former circulators are operated during both parallel-loaded operation and single-loaded operation while the latter is only operated during parallel-loaded operation. During operation the rotating assembly is fully floating on a dynamic gas bearing system. The circulator consists of the following components: electric stator and rotor assembly, internal structure supports, thrust bearings and journal bearings, impeller unit, and a filter unit. These internal structures are contained within a casing that is cooled by a water jacket. The casing prevents primary helium gas from leaking into the atmosphere. The flow rate of the gas is controlled by a variable speed motor using a frequency converter. The filter unit, which is on the top of the circulator, protects the impeller and rotating shaft from dust ([Ref. 9, p. 118](#)).



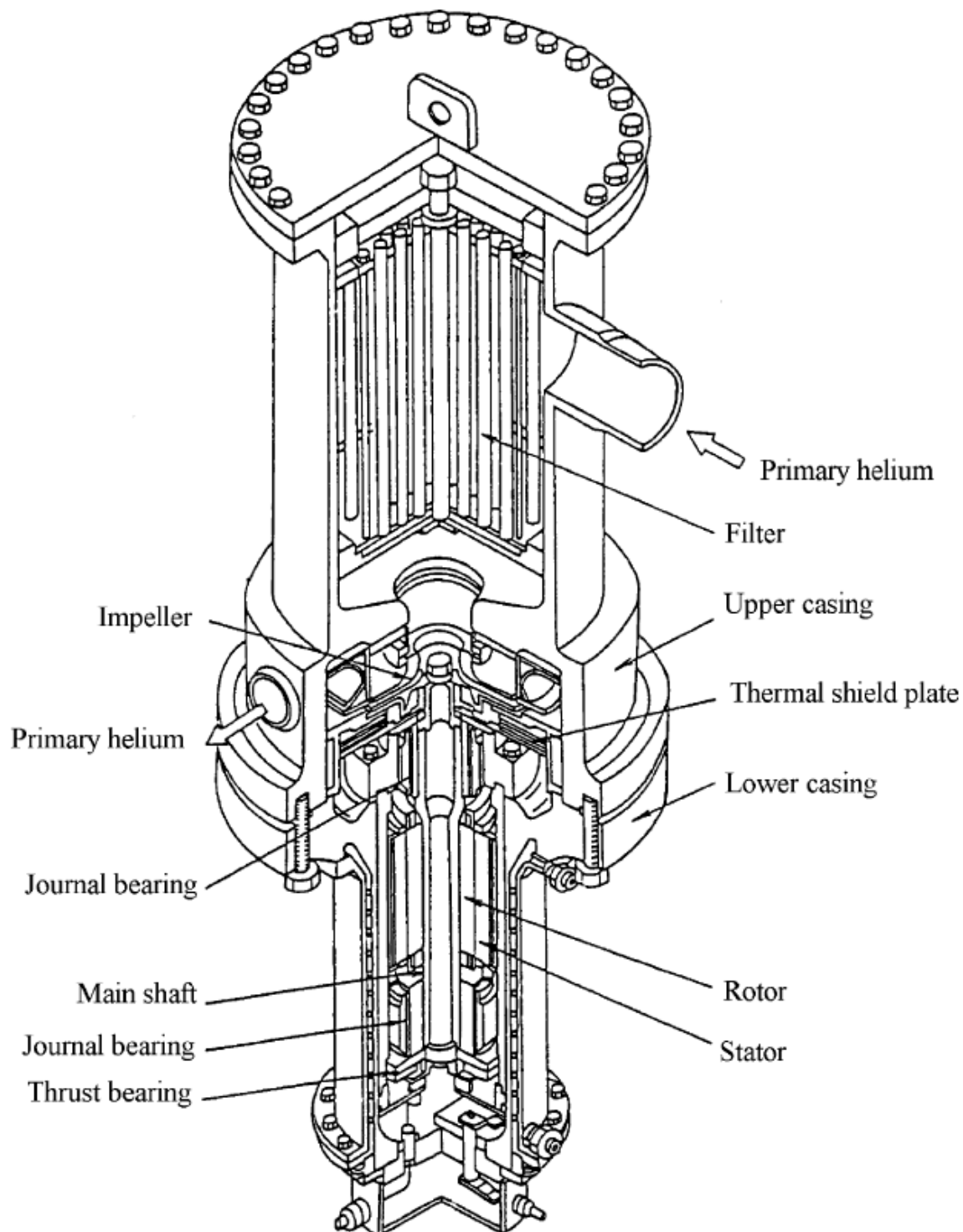


Figure 1.23. The Primary Gas Circulator (Ref. 9 p. 120).

Table 1.8. Major Specifications of the Primary Gas Circulator ([Ref. 9 p. 119](#)).

Type	For the IHX	For the PPWC
Centrifugal Gas Bearing		
Flow Rate (t/h)	14.9 (max)	15.1 (max)
Head (kPa)	79.4 (max)	107.9 (max)
Design Pressure (MPa)	4.7	4.7
Design Temperature (°C)	430	430
Material		
Casing	SCMV4-2NT SFVA F22B (2-1/4Cr-1Mo Steel)	
Motor		
Type	Cage-Type Induction Motor	
Power (kW)	260	
Number of Revolutions (rpm)	3000 – 12000	
Type of Frequency Converter	Thyristor-Converter	
Filter		
Type	Sintering Metal	
Material	SUS316	

The primary concentric hot gas duct consists of an outer pipe, an inner pipe, and a thermal insulator (Figure 1.24 and Table 1.9). The cold helium gas of 400 °C flows in an annular path inside the inner pipe, which has a temperature of 950 °C. The outer pipe can contain high pressure helium gas of 4.0 MPa. The inner pipe, which separates the high and low temperature helium gas paths, supports the pressure difference between the high and low gas. The pressure difference is about 0.1 MPa. The liner forms a high temperature helium gas boundary and reinforces the ceramic fiber insulator. The internal insulator between the liner and inner pipe minimizes heat loss from the high to low temperature helium gases and maintains the temperature of the inner pipe below 440 °C ([Ref. 9, p.119](#)).



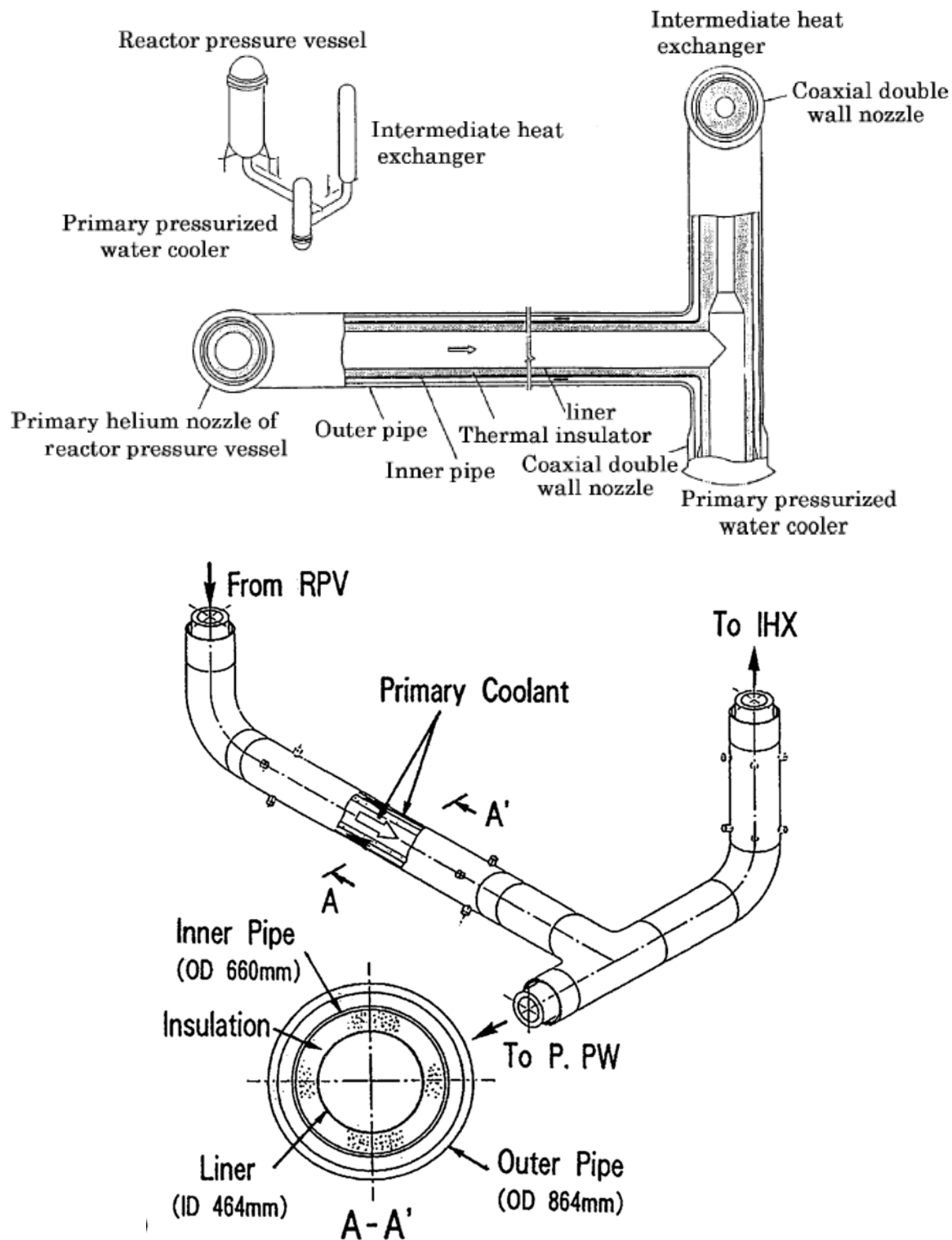


Figure 1.24. The Primary Concentric Hot Gas Duct (Ref. 9 p. 120 and Ref. 14, p. 387).

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Table 1.9. Major Specifications of the Primary Concentric Hot Gas Duct (Ref. 9 p. 119).

Design Pressure (MPa)	
Outer Pipe	4.7
Design Temperature (°C)	
Outer Pipe	430
Dimension of Outer Pipe (mm)	
Outer Diameter	863.6
Thickness	42
Dimension of Inner Pipe (mm)	
Outer Diameter	660.4
Thickness	15
Thickness of Thermal Insulator (mm)	90
Material	
Outer Pipe	SCMV4-2NT (2-1/4Cr-1Mo Steel) SFVA F22B (2-1/4Cr-1Mo Steel)
Inner Pipe	SCMV4-2NT (2-1/4Cr-1Mo Steel)
Liner	Hastelloy XR

The secondary helium cooling system consists of a secondary gas circulator, the secondary pressurized water cooler, and a secondary helium piping. This system is operated during the parallel-loaded operation. The heat is transferred from the primary to the secondary helium gas through the IHX. This heat is sent to the pressurized water cooling system through the secondary pressurized water cooler. In the future, the hydrogen production system will be connected to the secondary helium cooling system (Ref. 9, p. 119).

The structure of the secondary pressurized water cooler is fundamentally the same as of the PPWC. The secondary pressurized water cooler is a vertical U-bent type heat exchanger with double shells. The inner shell is thermally insulated on the inside. The structure of the secondary gas circulator is nearly the same as that of the primary gas circulator. The secondary gas circulator has one vertical centrifugal type gas circulator that is composed of the casing, an impeller, a rotor, bearings, and a motor. The secondary helium piping consists of a concentric hot gas duct connecting the IHX and the secondary pressurized water cooler, and a single wall piping connecting the secondary pressurized water cooler and the secondary gas circulator. Thermal insulation is attached on the inside of the inner pipe of the concentric hot gas duct. The hot helium gas from the IHX flows inside the inner pipe. The helium gas flows in the annular space between the inner and outer pipes (Ref. 9, pp. 120-121).

The pressurized water-cooling system is installed to cool primary and secondary helium gas in the PPWC and secondary pressured water cooler, respectively. The heat is finally released to the atmosphere via an air cooler. This system consists of a pressurized water pump, an air cooler, and piping. The pressure of the water is controlled to be lower than that of the primary cooling system so as to minimize the amount of water ingress into the primary cooling system in the event of a tube rupture accident, but the pressure is still high enough to prevent boiling. Two pressurized water pumps (including a spare) of horizontal centrifugal type are installed. They have a capacity of 640 t/h flow rate at 90 m delivery head. The air cooler consists of finned heat transfer tubes and blowers. It has a 30 MW cooling capacity and an air flow rate of 2600 t/h (Ref. 9, p. 121).

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The main cooling system is also used to remove decay heat of the core at normal reactor shutdown conditions. Besides the main cooling system, the HTTR has two other residual heat removal systems, which are the auxiliary cooling system and the vessel cooling system. The auxiliary cooling system removes the residual heat during anticipated operational occurrences and for accidents such as reactivity insertion or pipe rupture in the secondary cooling system. The vessel cooling system removes the residual heat during a loss of coolant accident in the primary cooling system ([Ref. 9, p. 121](#)).

The ACS consists of an auxiliary heat exchanger (AHX), two auxiliary helium circulators and an air cooler. At the AHX, the auxiliary helium is cooled by water. During normal operation, a small flow of auxiliary helium (~200 kg/h) passes through the AHX to the primary helium purification system so as to remove impurities contained within the reactor coolant. With a reactor scram, while the reactor coolant pressure boundary remains intact, the auxiliary helium cooling system automatically starts and transfers the residual heat from the core to the auxiliary air cooler. The AHX has heat transfer capacity of approximately 3.5 MW ([Ref. 2, p. 8 and Ref. 9, p. 121](#)).

The two VCSs (Figure 1.25) provide protection of the reactor core and RPV against thermal damage by residual heat after a scram when the ACS fails to cool the core. Each of these systems is capable of controlling temperatures of the core and RPV within safe limits and consists of water-cooled panels surrounding the RPV with two cooling water systems. Cooling tubes with fins form the panels and are arranged so that adjacent tubes do not belong to the same system, such that a tube failure will not endanger the RPV and core. The heat removal rate from the RPV to the panels is designed as 0.6 MW. The VCS is also an engineered safety feature equipped with two independent complete sets which are backed up with an emergency power supply. It is operated during normal operation to cool the biological shielding concrete wall ([Ref. 2, p.8 and Ref. 9, p. 121](#)).

The structure of the VCS cooling water panel is designed based on heat distribution calculations of the concrete wall of the RPV cavity and the analysis of heat removal values. Four heat reflectors were added between the RPV and the cooling water panel in order to decrease the heat removal values below pre-established limits at higher thermal loads. Cooling water pipes were added to the innermost reflector to increase heat removal values for low thermal loads. The pitch between two cooling water pipes is 120 mm. After the VCS was manufactured and installed, it had component testing completed by 1997 ([Ref. 14, p. 380](#)).

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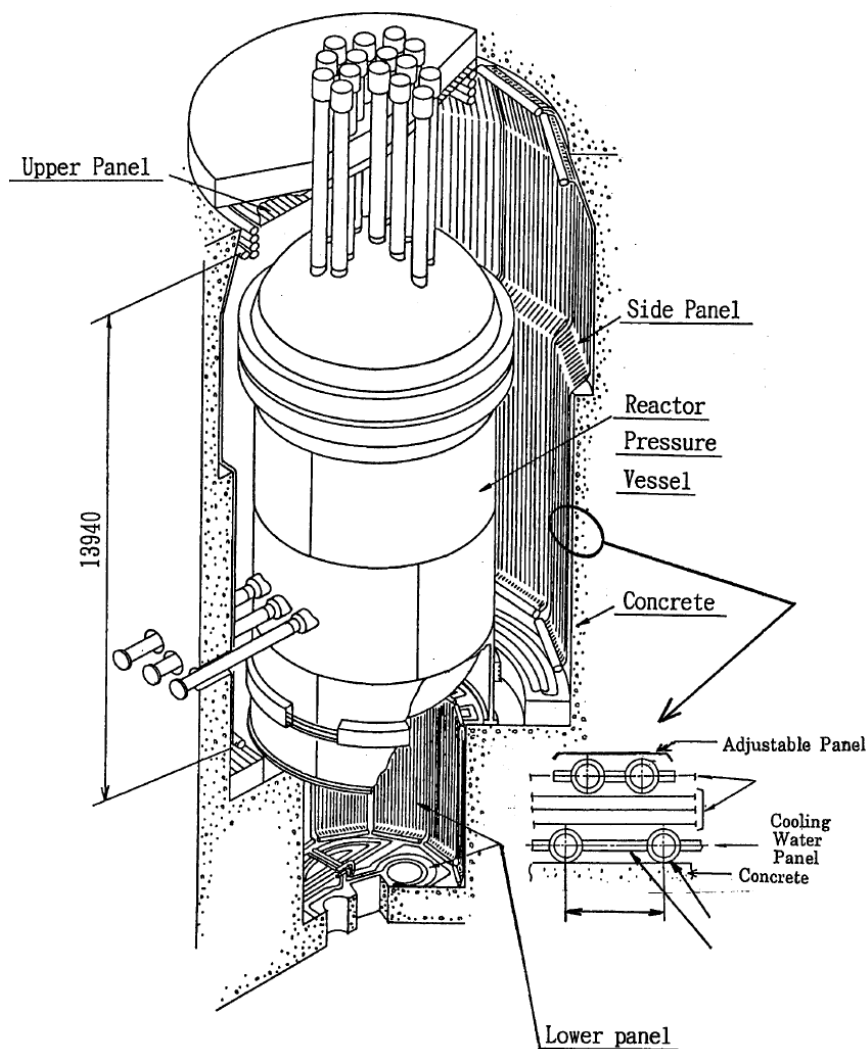
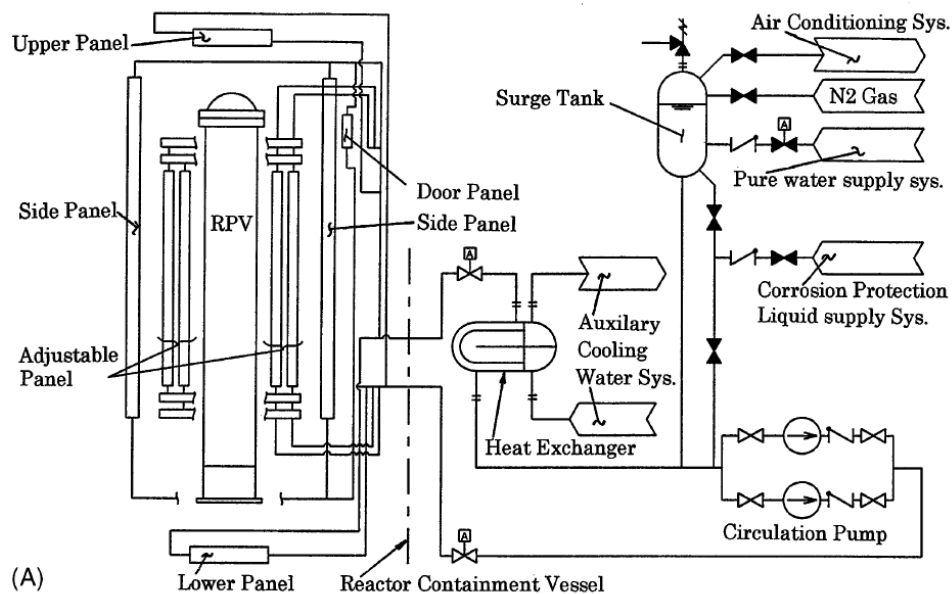


Figure 1.25. Structural View of the Water Cooling Panel of the Vessel Cooling System (Ref. 14 p. 381).

## Utility Systems

The utility systems of the HTTR include the auxiliary helium systems (helium purification, helium sampling, and helium storage and supply) and fuel handling and storage. The helium purification system is installed in the primary and secondary helium cooling systems to reduce the quantity of chemical impurities that could induce corrosion of the reactor and its components. The helium sampling systems monitor the concentration of impurities and the helium storage and supply systems provide a steady pressure supply of helium during normal operation. The fuel handling and storage system is used to manage new and spent fuels. Refuelling occurs every three reactor years (Ref. 11, pp. 147-148).

The helium purification systems reduce the quantity of chemical impurities such as hydrogen, carbon monoxide, water vapor, carbon dioxide, methane, oxygen, and nitrogen to limits below those shown in Table 1.10. The primary helium purification system is mainly composed of a pre-charcoal trap, an inlet heater, two copper oxide fixed beds, coolers, two molecular sieve traps, two cold charcoal traps, and gas circulators. A flow diagram is shown in Figure 1.26. The secondary helium purification system is identical to the primary one, except that it does not have a pre-charcoal trap. The helium flow rate is 200 kg/h. The primary helium gas is introduced into the helium purification system through the auxiliary cooling system and the purified gas returns to the ACS and standpipes (Ref. 11, p. 148).

Table 1.10. Upper Impurity Limit in the Primary Coolant at 4 MPa and a Reactor Outlet Coolant Temperature between 800 to 950 °C (Ref. 11 p. 148).

Impurity	Concentration (ppm)
H <sub>2</sub>	3.0
CO	3.0
H <sub>2</sub> O	0.2
CO <sub>2</sub>	0.6
CH <sub>4</sub>	0.5
N <sub>2</sub>	0.2
O <sub>2</sub>	0.04

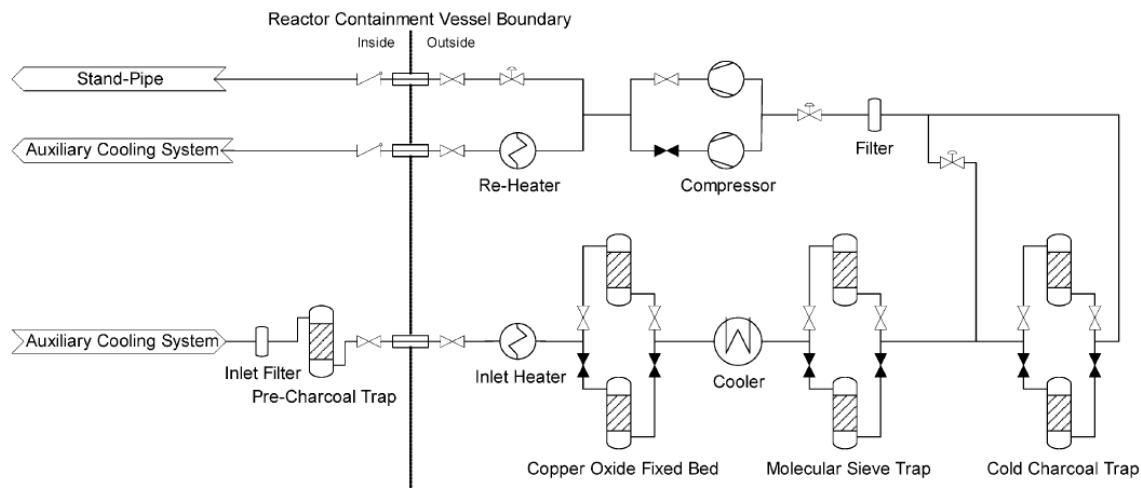


Figure 1.26. Flow Diagram of the Primary Helium Purification System (Ref. 11 p. 148).

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HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

The helium sampling system detects chemical and radioactive impurities in the primary cooling system and secondary helium cooling system using gas chromatograph mass spectrometers. Both systems transmit impurity concentration measurements to the main control room. The helium storage and supply systems are installed for the primary cooling system and the secondary helium cooling system. The primary helium storage and supply system is composed of storage tanks, a supply tank, and helium compressor. The primary coolant is kept at a fixed pressure of about 4.0 MPa during normal operation. The secondary helium storage and supply system stores helium gas at a pressure higher than that of the primary coolant. The primary helium system has six storage tanks with a capacity of 220 kg each and a supply tank with a capacity of 110 kg. Two helium gas compressors are installed with one compressor on stand-by. The supply valve for the primary coolant opens and closes at 3.92 to 3.99 MPa, respectively, and the exhaust valve opens and closes at a pressure of 4.00 and 3.96 MPa, respectively. The supply rate of the helium gas is used for detecting the leakage rate of primary coolant during operation. The secondary helium system has a storage tank capacity of 25 kg, a supply tank capacity of 10 kg, and two compressors. The supply valve for the secondary coolant opens and closes at a pressure difference of -29.4 and -9.8 kPa, respectively, and the exhaust valve opens and closes at a pressure difference of 29.4 and 19.5 kPa, respectively. The design pressure difference between the primary and the secondary coolant is 73.5 kPa (Ref. 11, pp. 149-150).

The fuel handling system, utilized to install and remove fuel elements, replaceable reflector blocks, top shielding blocks, control rod guide blocks, and control rods, consists of a fuel handling machine, attached equipment, and auxiliary equipment. The fuel handling machine consists mainly of a shielded cask, a gripper, a fuel handling unit drive system, a rotating rack, and a door valve. The machine has a shield sufficient to protect fuel handling personnel, has a gas-tight boundary, and maintains subcriticality when fully loaded. A diagram and photo of the fuel handling and storage systems are shown in Figures 1.27 and 1.28, respectively (Ref. 11, p 151).

The fuel handling machine (Figure 1.29) is 11 m in height and weighs 150 tons. It has a capacity to store 10 fuel blocks at a time. The fuel handling mechanism that grips the fuel block is hung down by wire rope and chain, and can access 13 columns through each standpipe. The fuel handling mechanism consists of a gripper that grips fuel blocks and a support link with a pantograph-type arm that guides the gripper movement. The gripper has latch pawls on the underside that are inserted into the fuel block and opened to hook and lift the block that weighs about 200 kg (Ref. 14, p. 385).

The HTTR has two fuel storage systems, one for new fuel and the other for spent fuel. The storage system for the fresh fuel consists of fuel assembly and testing equipment, a new-fuel storage cell, and inert gas replacement equipment. Up to 1.5 times the core inventory can be stored in new-fuel storage. A vertical, cylindrical vessel provides sufficient distance between storage racks to ensure subcriticality. The fuel elements are maintained in a dry, inert environment. The spent-fuel storage system is located inside the reactor building, and consists of a spent-fuel storage pool, water cooling and purification system, and irradiated material storage pits for spent fuel elements, control rod guide blocks, and replaceable reflector blocks (Figure 1.30). The spent-fuel storage has sufficient shielding and capacity to store about two core inventories. The inside of the pool is lined with stainless steel to prevent leakage of the pool water. After two years of cooling, the spent fuel is transferred to another spent-fuel storage system located in a neighboring building to the HTTR facility (Ref. 11, pp. 153-154).



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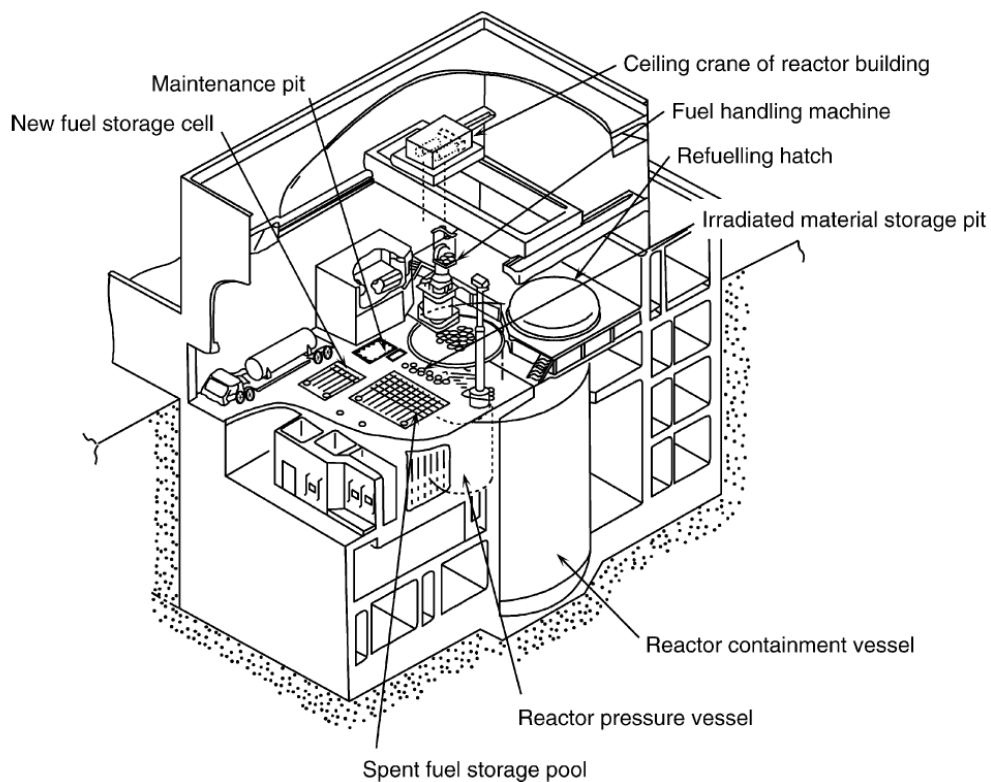


Figure 1.27. The Fuel Handling and Storage System in the Reactor Building (Ref. 11 p. 151).

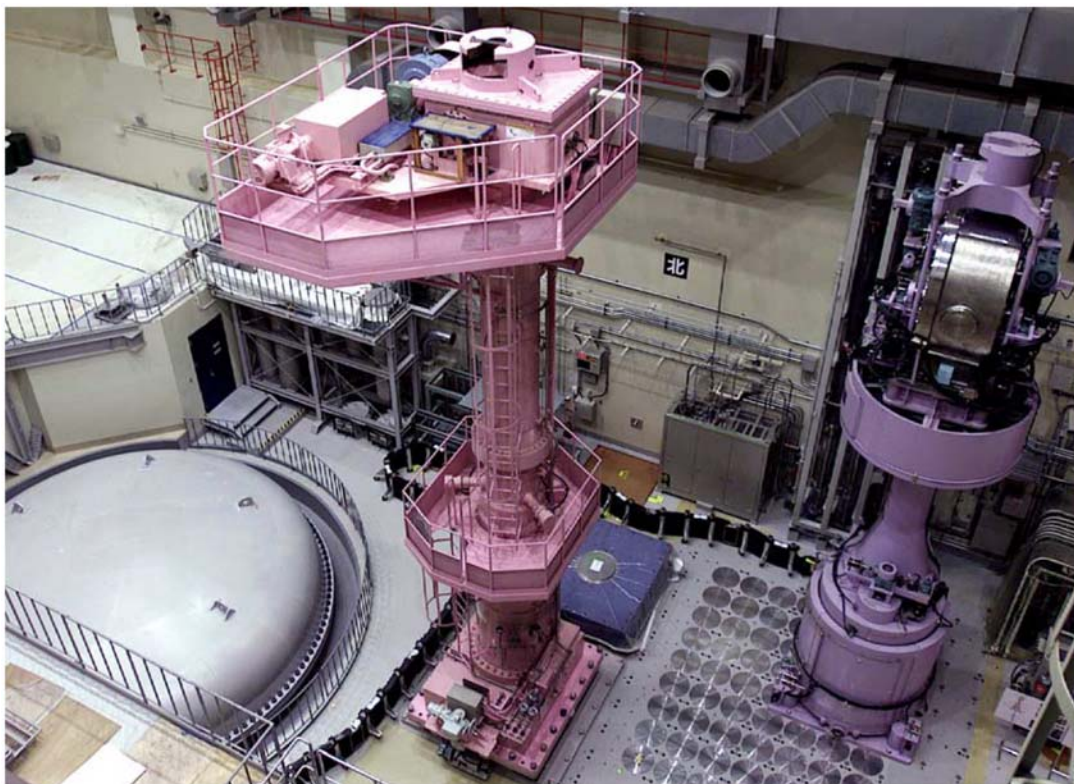


Figure 1.28. Photo of the Operation Floor in the Reactor Building. (Ref. 11 p. 152).

The fuel handling machine is above the spent-fuel storage pool on the right.

The control-rod handling machine and refuelling hatch of the reactor containment vessel are seen at the center and the left side, respectively.

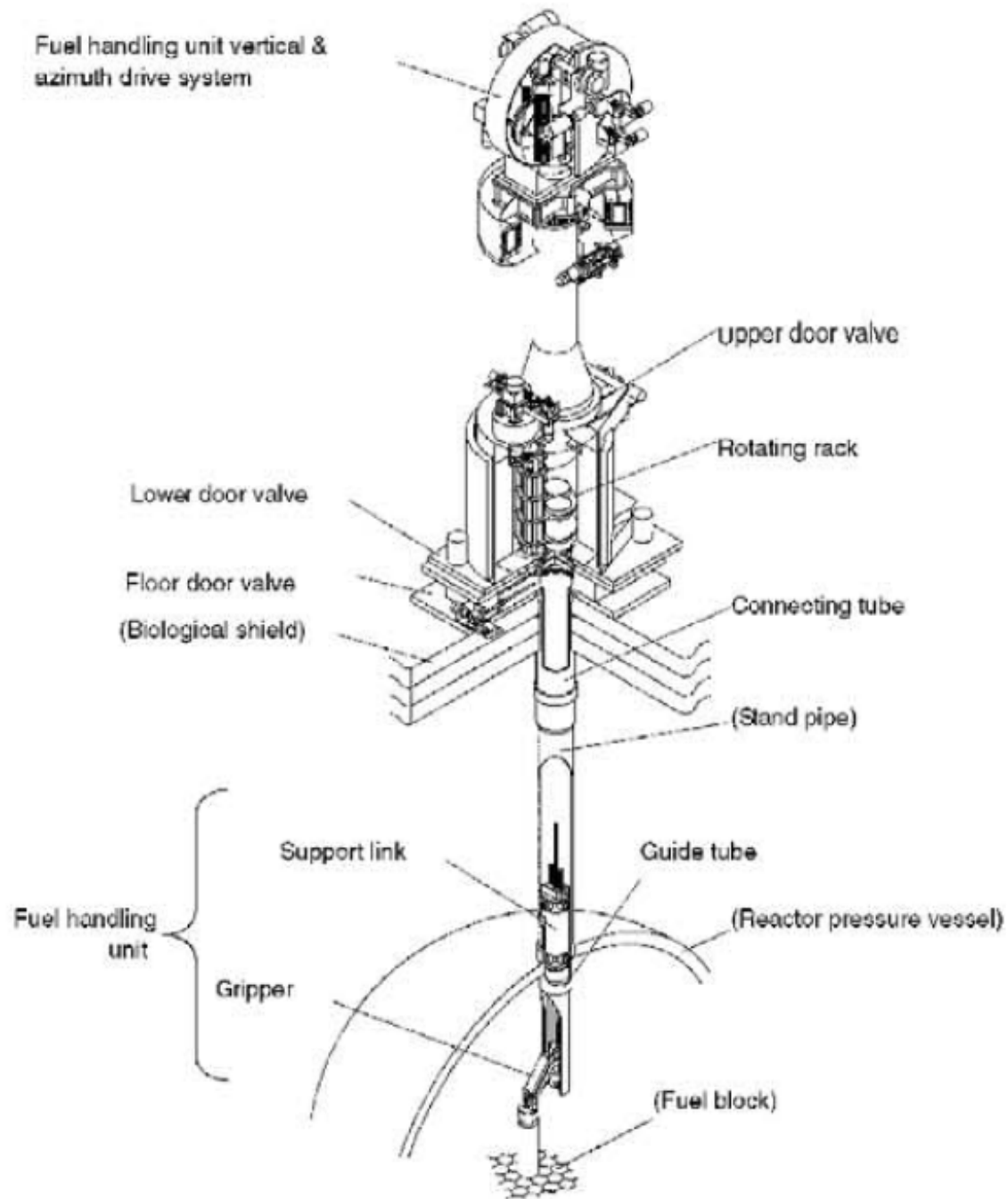
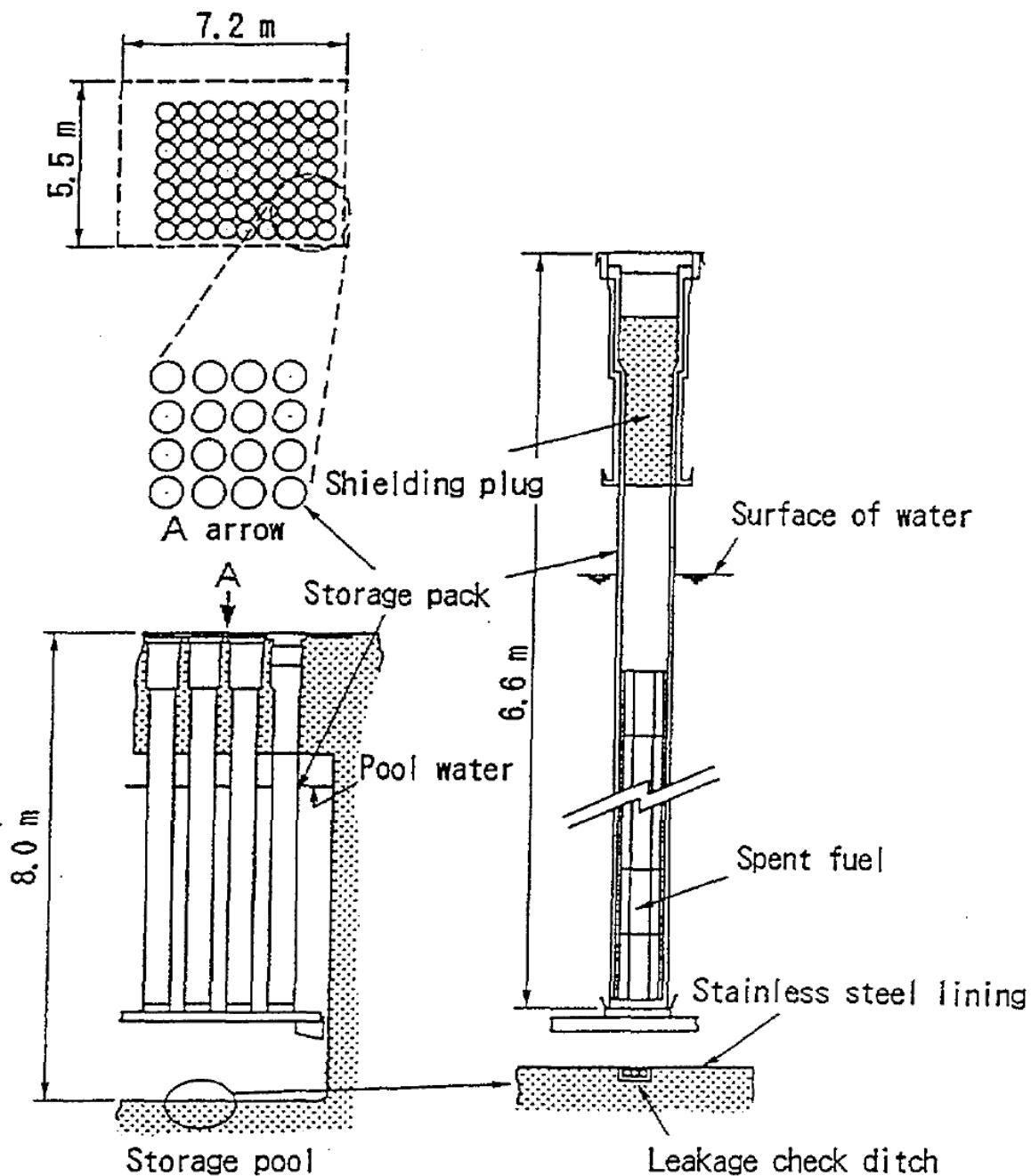


Figure 1.29. HTTR Fuel Handling Machine (Ref. 14, p. 384).

Figure 1.30. Spent Fuel Storage System in the HTTR Reactor Building.<sup>a</sup>

<sup>a</sup> K. Sawa, S. Yoshimuta, S. Shiozawa, S. Fujikawa, T. Tanaka, K. Watarumi, K. Deushi, and F. Koya, "Study on Storage and Reprocessing Concept of the High Temperature Engineering Test Reactor (HTTR) Fuel," IAEA-TECDOC-1043, International Atomic Energy Agency, Vienna, pp. 177-189 (September 1998).

### *1.1.2.2 Reactor Internals*

Fuji Electric collaborated with JAEA for much of the thermal-hydraulic, nuclear, and safety design analysis of the HTTR. They supplied major equipment such as the reactor internals, the fuel handling and storage facilities, and the radiation monitoring system ([Ref. 14, p. 384](#)).

The reactor internals consist of graphite core support structures, metallic core support structures, and the other components shown in Figure 1.31. The graphite support structures include hot plenum blocks, core bottom structures, and core support posts. The hot plenum blocks provide lateral and vertical positioning and support of the core array. The blocks contain flow paths that guide the primary coolant from the outlet of the fuel columns and distribute it into the hot plenum beneath the hot plenum blocks. The core support posts are designed such that they support the core and hot plenum block arrays, which form the hot plenum. The permanent reflector block is a graphite structure surrounding the replaceable reflector blocks and control rod guide blocks located in the circumference of the active core. The metallic core support structures are composed of core support plates, a core support grid and the core restraint mechanisms. The core support plate and the core support grid are placed below the thermal insulation layers. The core restraint mechanism surrounds the permanent reflector blocks ([Ref. 3, pp. 15-16, and Ref. 6, p 82](#)).



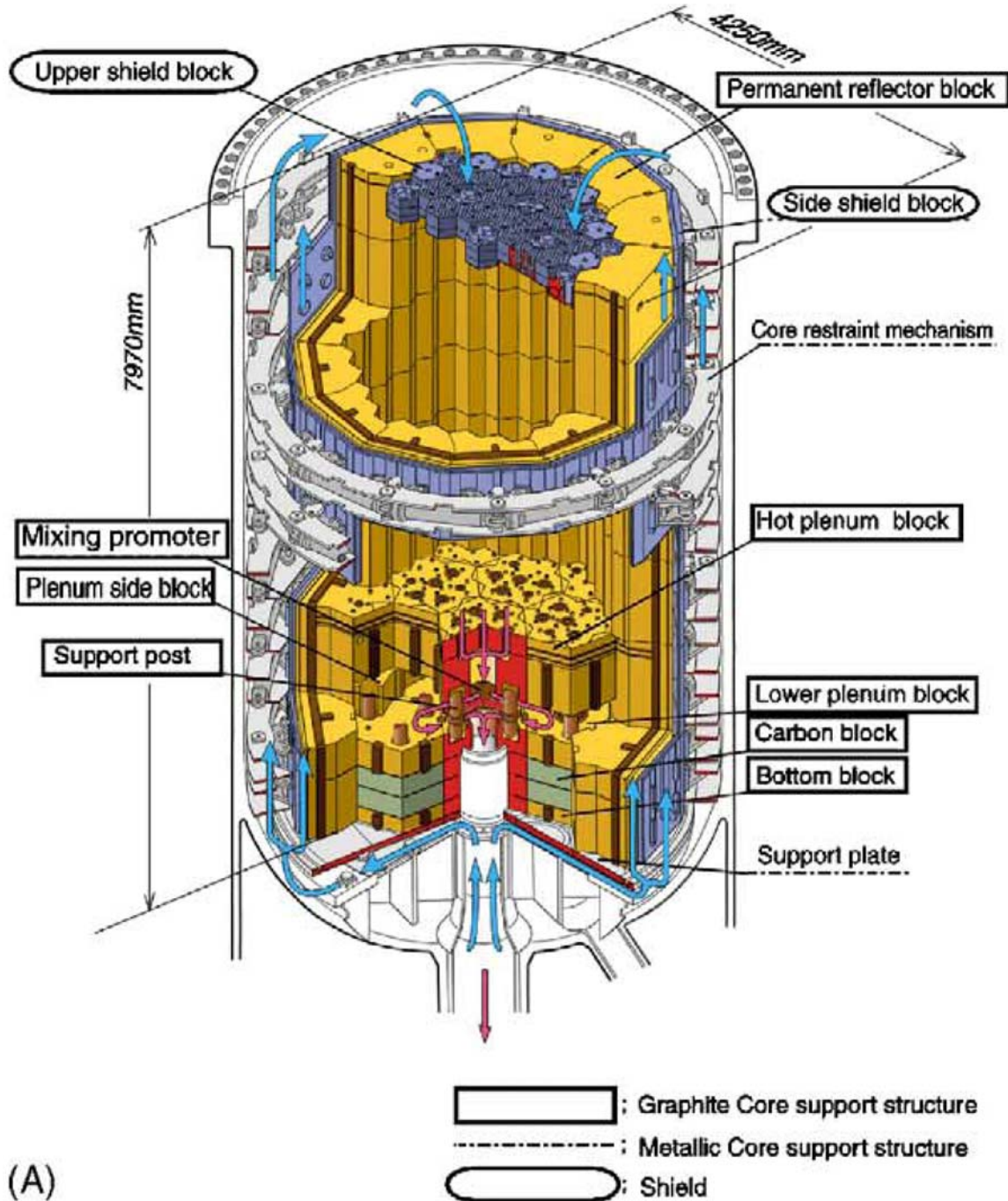


Figure 1.31. Reactor Core Internals of the HTTR (Ref. 14, p. 384).

**Graphite Support Structure**

The structures at the core bottom consist of three types of blocks: lower plenum blocks, carbon blocks, and bottom blocks. These blocks function as thermal insulation between the hot plenum and metallic core support structure (Ref. 6, p. 83).

The hexagonal hot plenum block array is made up of two axial layers. The upper layers consist of sealed plenum blocks, in which leakage flow between these blocks is sealed by the triangular seal elements.

The lower layer is composed of keyed plenum blocks, which provide lateral and vertical positioning and support of the core array. The hot plenum block assembly contains passages for the primary coolant flow from the columns in the core to the hot plenum (Ref. 6, p. 82).

The hot plenum block is 0.9 m in height and about 1 m across flats. They are combined with each other by key-keyway systems and are supported separately by core support posts that are 0.6 m in length and 0.15 m in diameter. Both ends of a core support post are spherical and fit into similarly shaped recesses in the post seats. The curvature radii of the ends of the core support post differ from those of the post seats to accommodate the relative motion between the hot plenum block and the lower plenum block.<sup>a</sup>

The support posts and seats are designed to support the core and hot plenum blocks while providing the hot plenum for the primary coolant flow. The hot plenum space is where the hot core outlet helium gas can be mixed uniformly (Ref. 6, pp. 82-83).

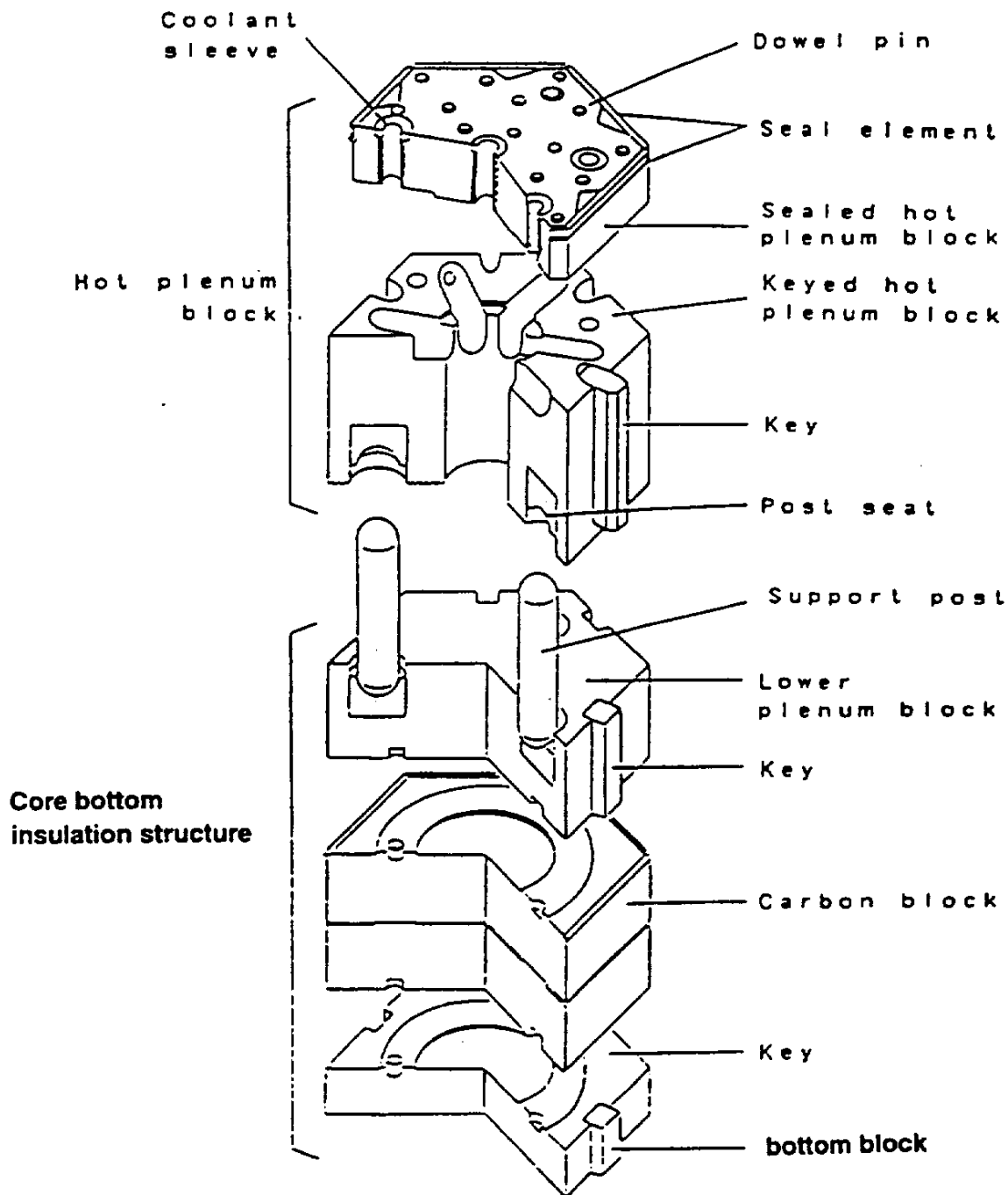
A diagram depicting the hot plenum blocks and support post is shown in Figure 1.32. A close-up of the reactor core internals at the bottom of the HTTR is shown in Figure 1.33. A diagram of the high-performance triangular shape seal used to minimize leakage in the hot plenum is depicted in Figure 1.34. These elements provide superior stability (Ref. 15, p. 41) and increase the effective core cooling by reducing gap-flow between the blocks.<sup>b</sup>

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<sup>a</sup> T. Iyoku, J. Smita, M. Ishihara, and S. Ueta, "R&D on Core Seismic Design," *Nucl. Eng. Des.*, **233**: 225-234 (2004).

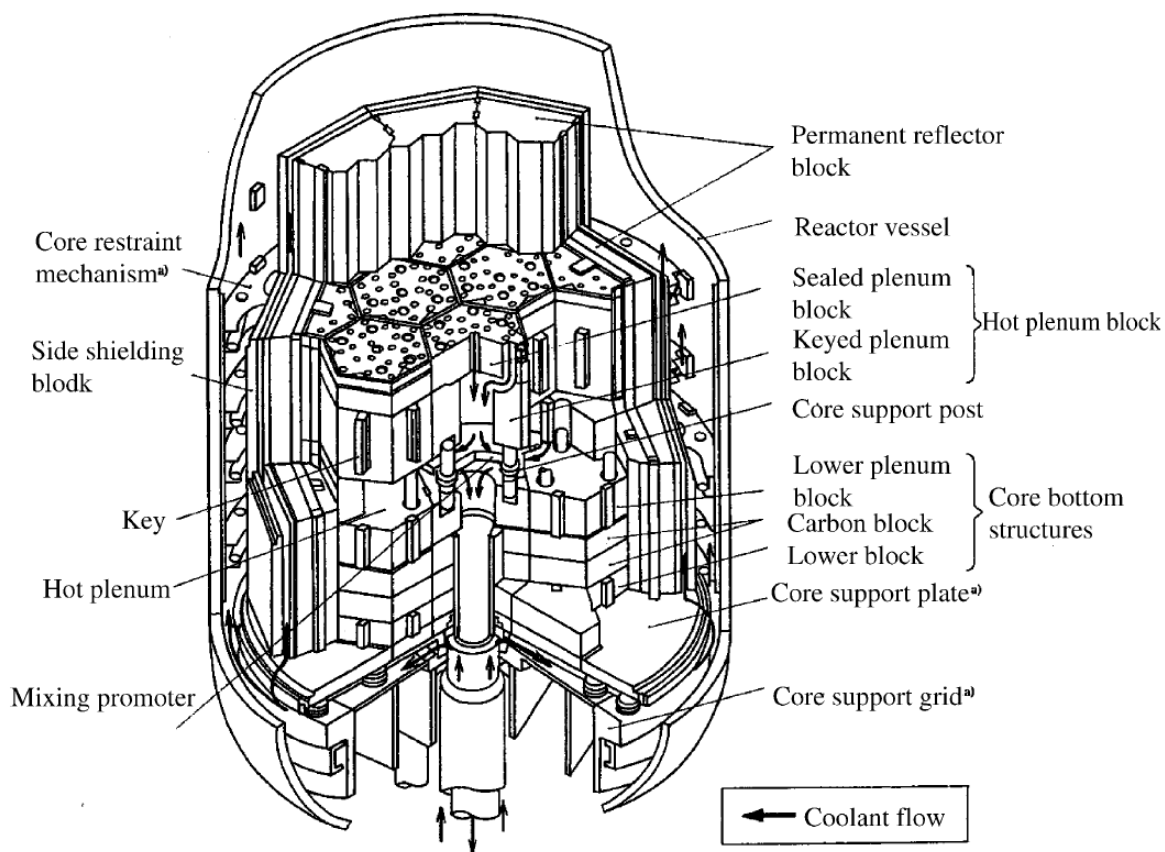
<sup>b</sup> S. Maruyama, A. Saikusa, S. Shiozawa, N. Tsuji, and T. Miki, "Results of Assembly Test of HTTR Reactor Internals," IAEA-TECDOC-901, International Atomic Energy Agency, Vienna, pp. 105-116 (August 1996).



Figure 1.32. Keyed Plenum Block and Core Bottom Graphite Structure.<sup>a</sup>

<sup>a</sup> S. Maruyama, A. Saikusa, S. Shiozawa, N. Tsuji, and T. Miki, "Results of Assembly Test of HTTR Reactor Internals," IAEA-TECDOC-901, International Atomic Energy Agency, Vienna, pp. 105-116 (August 1996).

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(a) metallic core support structures

Figure 1.33. Reactor Core Internals of the HTTR (Ref. 3, p. 17).

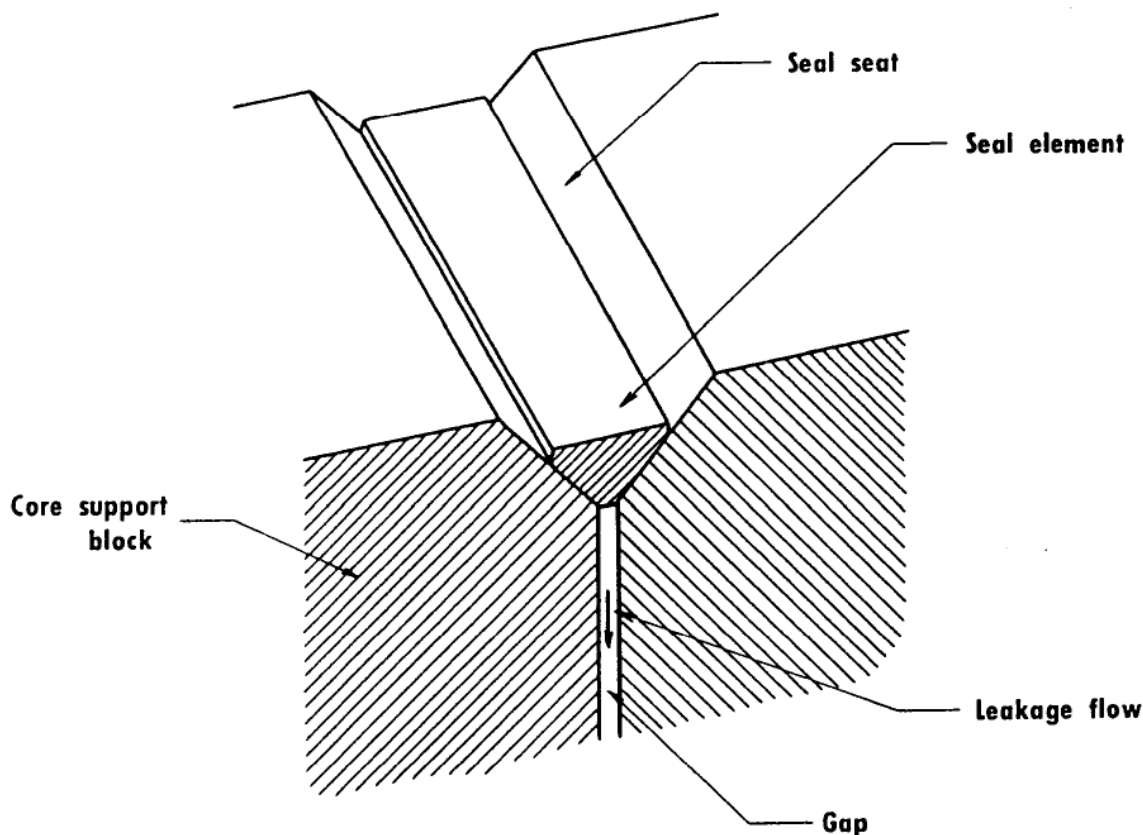


Figure 1.34. New Seal Mechanism for Hot Plenum Block ([Ref. 15, p. 41](#)).

### **Metallic Support Structure**

The metallic core support structures include a support plate, the core support grid, and the core restraint mechanism, as shown in Figure 1.33. The support plate and core support grid are located under the core-bottom insulation layer. The core restraint mechanism surrounds the permanent reflector blocks. The support plate and core support grid form a statically stable foundation for the core components and reactor internals under seismic loads and elevated temperatures ([Ref. 6, p. 84](#)).

The support plate forms a plain foundation surface for the core components and reactor internals, and is composed of steel plates with a thickness of 89 mm. The support plates are set on steel support posts, guiding the core weight to the core support grid below. The core support grid transfers the total weight of the core to the RPV through the support ribs welded onto the inner surface of the hemispherical bottom head closure. The grid is reinforced with welded diamond-shaped plate structures. The support plate and the core support grid are cooled by the primary coolant flowing into the RPV ([Ref. 6, p. 85](#)).

The core restraint mechanism surrounds and stresses the permanent reflector blocks with 10 axially distributed units. Each unit consists of 12 restraint bands, 12 band supports, and 2 restraint rings and radial keys, as shown in Figure 1.35. The restraint bands are stressed to produce radial force on the band supports that is transmitted to the permanent reflector blocks via the side shield blocks and support legs. The restraint rings, located above and below the restraint bands, are normally stress free. They limit radial displacement of the bands during seismic events. Normally there is no contact between the restraint rings and the radial keys. The radial keys are individually adjusted before being attached to the RPV and are provided to compensate for circumferential fabrication tolerances. The restraint bands are composed of a central rod and six concentric pipes, as shown in Figure 1.36, except for the 9<sup>th</sup> and 10<sup>th</sup> units. These latter units have eight concentric pipes each. The pipes are connected in series with the

pipes mutually acting as compression and tensile elements. The whole arrangement forms a tensile spring with the bolt eyes connected to the central bar on one end and to the outer pipe on the other end of the band (Ref. 6, p. 85).

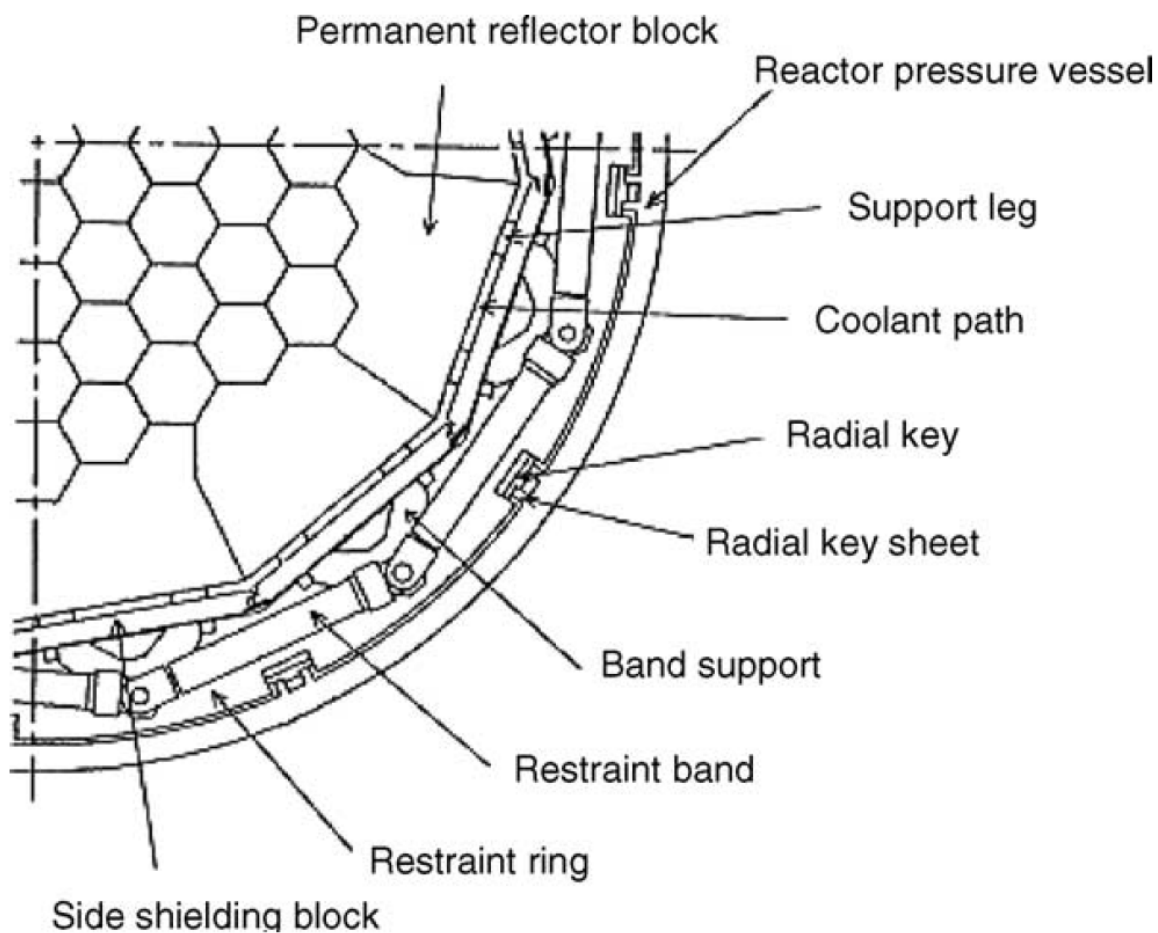


Figure 1.35. Core Restraint Mechanism (Ref. 6, p. 85).

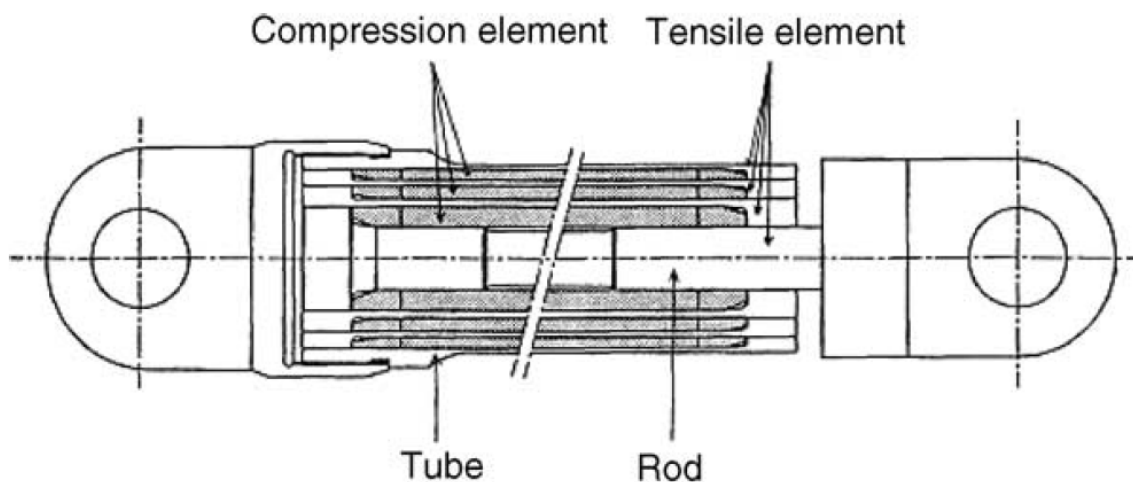


Figure 1.36. Core Restraint Band (Ref. 6, p. 85).

### **Reactor Core Shielding**

Shielding consists of top and side shielding blocks composed of neutron absorbing material and casing. The top shielding block is installed at the top of each column, and has coolant channels matching the channels in the column below. The side shielding blocks are installed outside of the permanent reflector blocks and are compressed inward by the core restraint mechanism. The top shielding block is hexagonal and composed of sintered  $B_4C/C$ , a casing of SUS316, and dowel pins. The top shielding blocks for the fuel columns have an internal cavity for coolant in order to prevent neutron streaming (Figure 1.37). Dowel pins on the top are used for positioning of the handling head of the fuel handling machine. The side shielding block is also composed of  $B_4C/C$  and a SUS316 casing, as shown in Figure 1.38. Four support legs are connected directly to the thick SUS316 plate, which locates the RPV side of the blocks and transfers the compressive forces of the restraint band to the permanent reflector blocks (Ref. 6, p. 86).

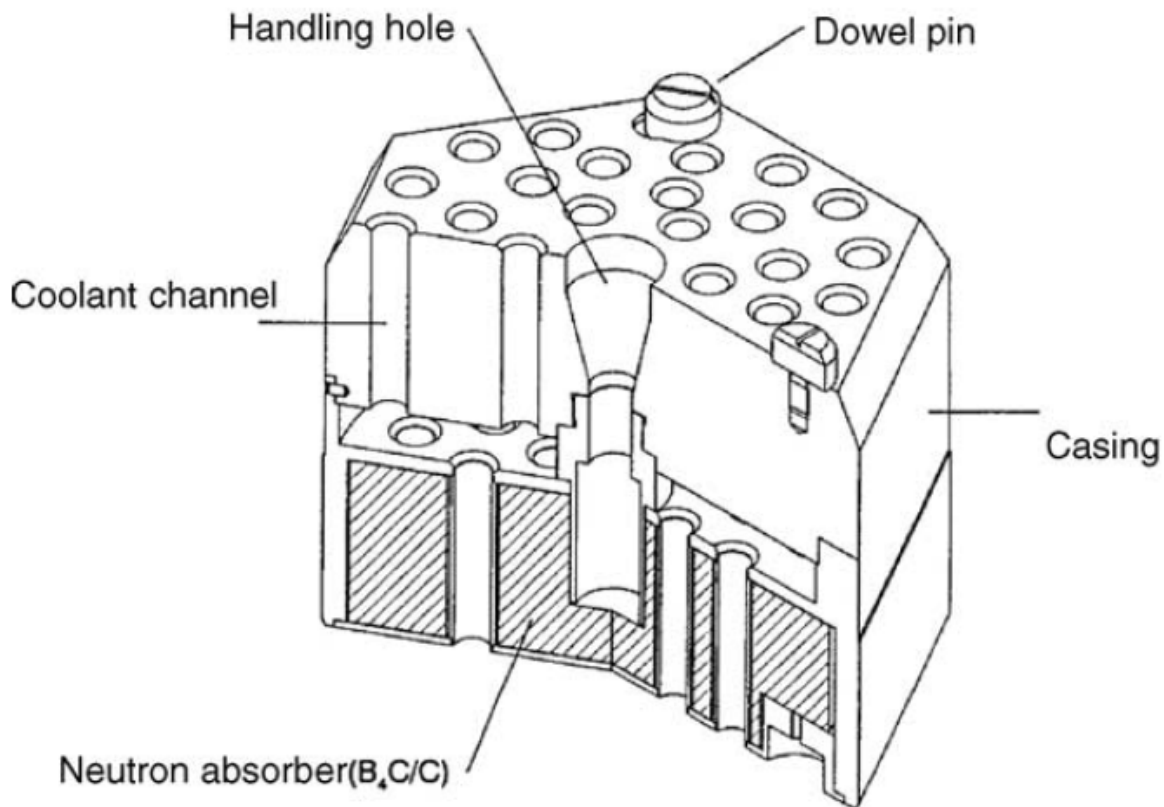


Figure 1.37. Top Shielding Block (Ref. 6, p. 86).

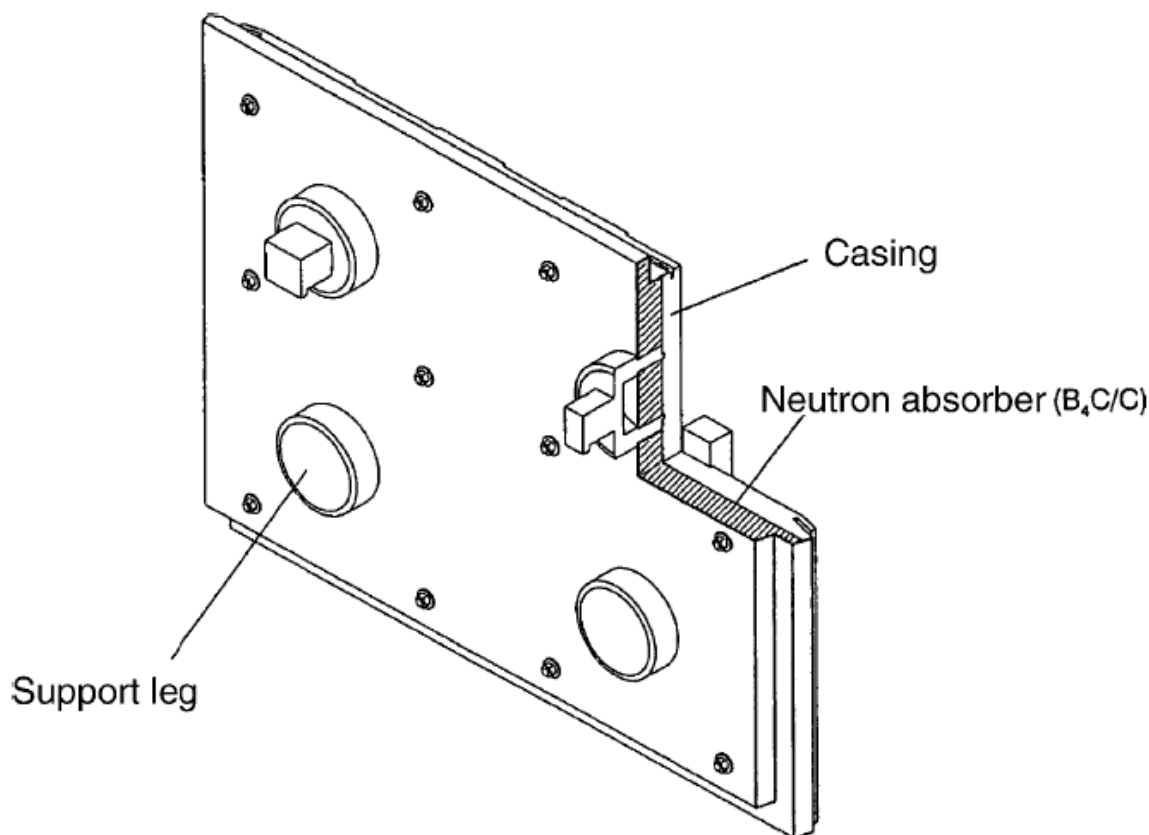


Figure 1.38. Side Shielding Block (Ref. 6, p. 86).

### **General Instrumentation and Control Systems**

The nuclear instrumentation for the HTTR is comprised of a wide range monitoring system (WRMSS) and a power range monitoring system (PRMS), shown in Figure 1.39. The wide range monitoring system is available as a post accident monitor under accident conditions such as rupture of the primary concentric hot gas duct. The wide range monitoring system is used to measure the neutron flux from  $10^{-8}$  to 30% of rated power. Three fission chambers are installed in the permanent reflector blocks through the stand-pipes. The power range monitoring system is used to measure the neutron flux from 0.1 to 120% of rated power. The power range monitoring system is also used as the sensor for the reactor power control system. The detectors of the power range monitoring system are located outside the reactor pressure vessel. The detectors have high sensitivities so as to detect the neutron flux at very low levels. In order to monitor the core outlet temperature of the primary coolant, seven thermocouples are arranged in the hot plenum blocks below the reactor core. N-type thermocouples (Nicrosil-Nisil) are selected because the temperature of the primary coolant in the hot plenum reaches about 1000 °C at rated power operation. The fuel failure detection system is composed of precipitators, a pre-amp, and a control box. The precipitator is used to detect  $\beta$ -rays radiated from short-lived gaseous fission products such as Kr-88, Kr-89, and Xe-138 (Ref. 3, p. 18). Additional details regarding the reactor instrumentation is summarized elsewhere.<sup>a</sup>

The maximum temperature of the blocks has also been reported of reaching about 1100 °C during high temperature operation (Ref. 7, p. 90).

<sup>a</sup> K. Saito, H. Sawahata, F. Homma, M. Kondo, and T. Mizushima, "Instrumentation and Control System Design," *Nucl. Eng. Des.*, **233**: 125-133 (2004).



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The WRMS fission counter is used under a high temperature environment at the top of a permanent reflector. The PRMS uncompensated ionization chamber can detect a low neutron flux level outside the RPV. In the HTTR, the temperature around the wide range detector becomes about 600 °C and the neutron flux level around the power range detector becomes about  $10^7$  n/cm<sup>2</sup>-s during the rated power operation of 30 MW (Ref. 15, pp. 111-112).

The control system of the HTTR consists of an operational mode selector, a reactor power control system, and a plant control system. Microcomputers are used for the plant control system and the reactor power control system. An operational mode selector supervises them. A schematic of the HTTR control system is shown in Figure 1.40. Additional information regarding reactor control design is summarized elsewhere.<sup>a</sup>

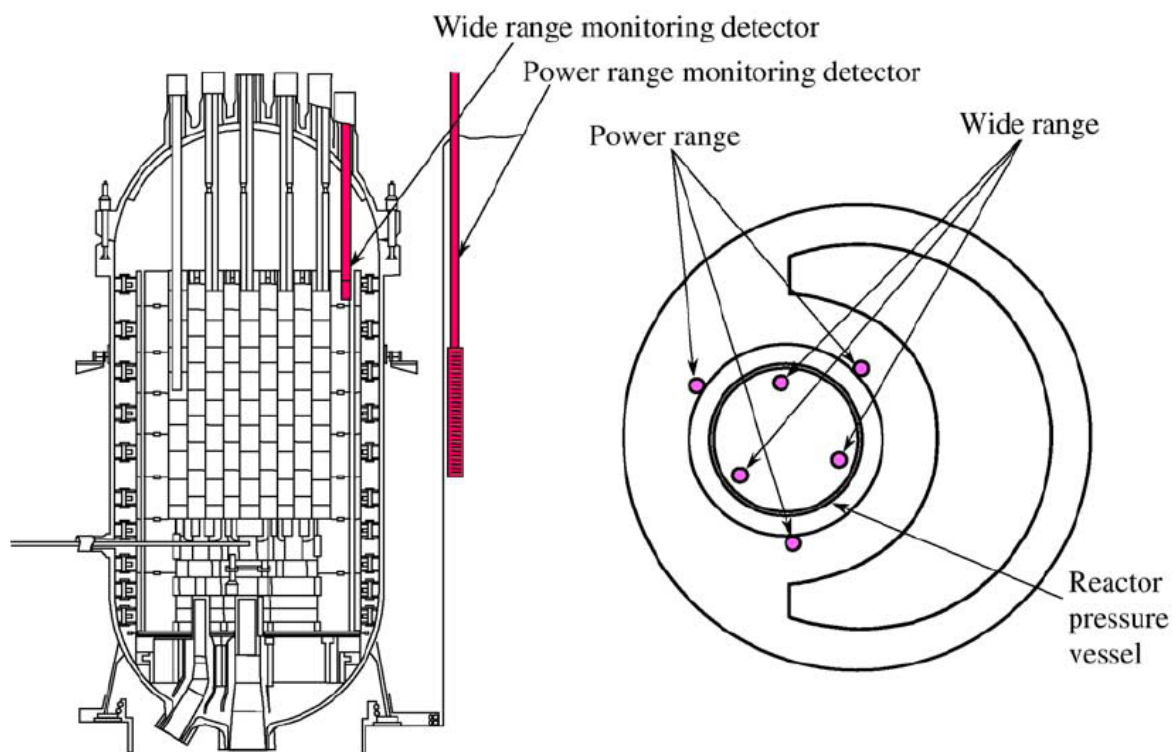
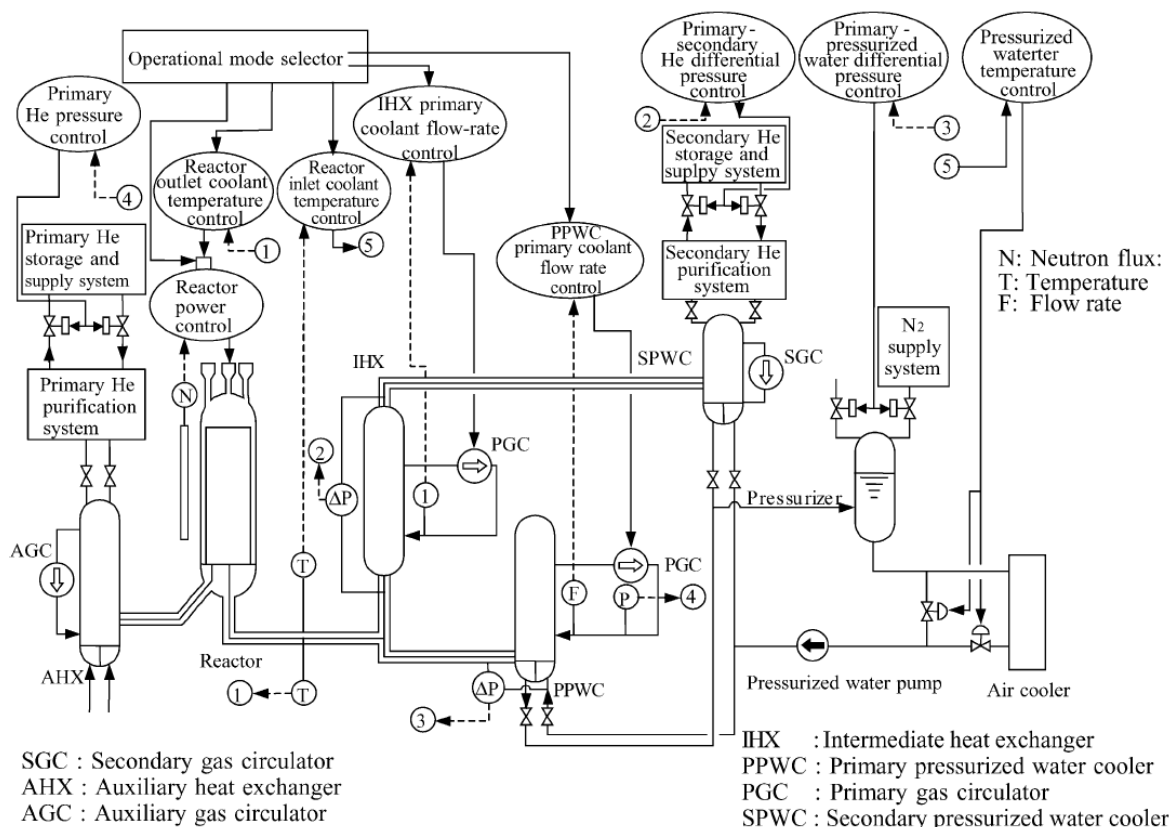


Figure 1.39. Neutron Detection Arrangement.<sup>b</sup>

<sup>a</sup> K. Saito, H. Sawahata, F. Homma, M. Kondo, and T. Mizushima, "Instrumentation and Control System Design," *Nucl. Eng. Des.*, **233**: 125-133 (2004).

<sup>b</sup> K. Saito, H. Sawahata, F. Homma, M. Kondo, and T. Mizushima, "Instrumentation and Control System Design," *Nucl. Eng. Des.*, **233**: 125-133 (2004).

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HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATEFigure 1.40. Schematic Diagram of the Control System of the HTTR.<sup>a</sup>**Irradiation Equipment**

The HTTR has a unique and superior capability to irradiate large-sized specimens at high temperatures with a uniform neutron flux in irradiation tests. The I-I type irradiation equipment (Figure 1.41) is the first irradiation rig in the HTTR and is served for an in-pile creep test for a metallic material. The equipment can have specimens up to 6 mm in diameter.<sup>b</sup>

<sup>a</sup> K. Saito, H. Sawahata, F. Homma, M. Kondo, and T. Mizushima, "Instrumentation and Control System Design," *Nucl. Eng. Des.*, **233**: 125-133 (2004).

<sup>b</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

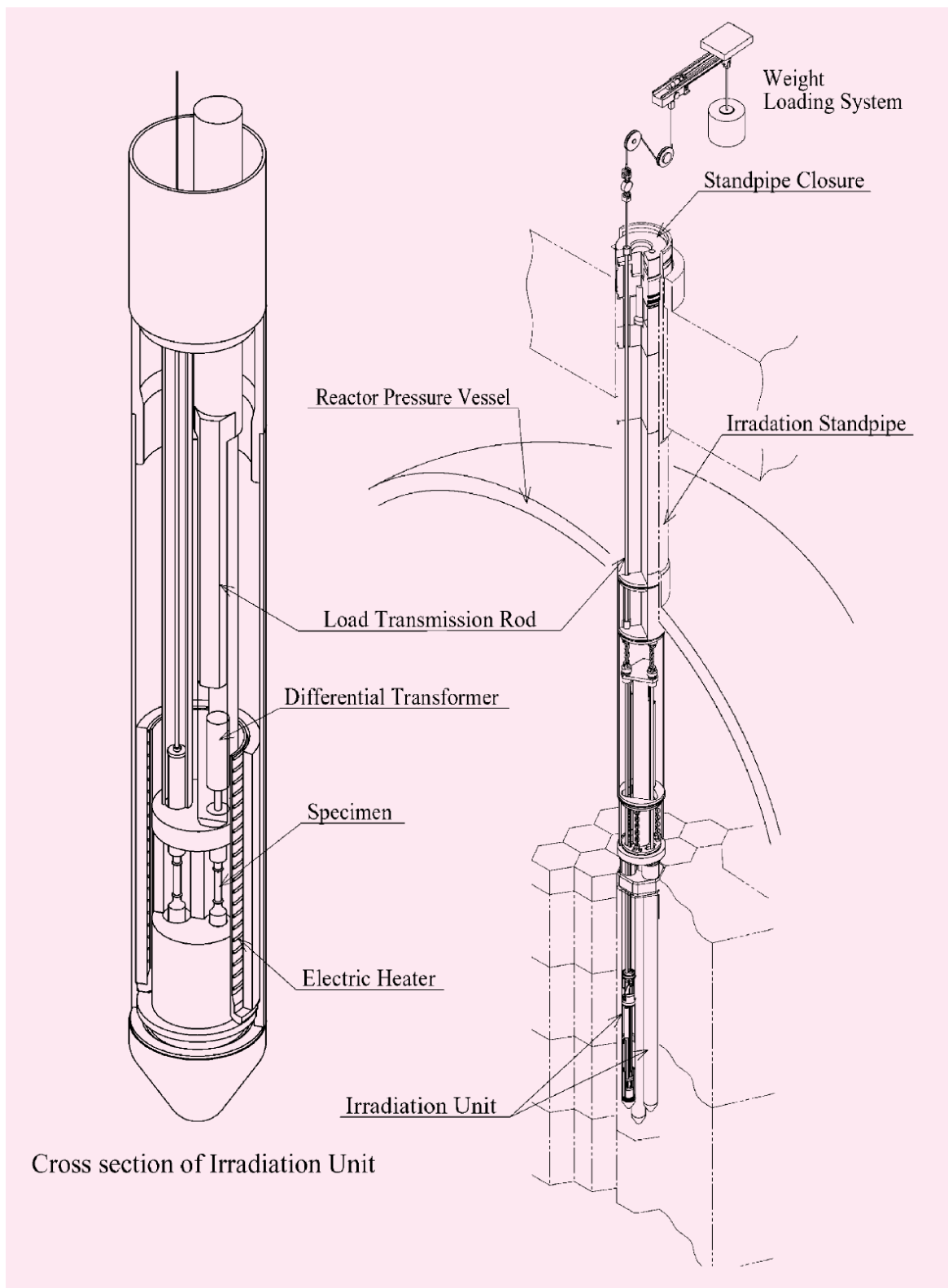


Figure 1.41. I-I Type Irradiation Equipment.<sup>a</sup>

<sup>a</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

### 1.1.2.3 Reactor Core Components

#### **Reactor Core**

The active core has a height of 290 cm and effective diameter of 230 cm (Figures 1.42 and 1.43). The core consists of hexagonal graphite blocks 580 mm high and 360 mm across flats. These fuel blocks, control rod (CR) guide blocks, replaceable reflector blocks, and irradiation blocks are stacked vertically into columns. The active core contains 30 fuel columns. One column contains five fuel blocks. Each fuel block has 31 or 33 coolant channels, into which fuel rods are inserted. Fuel rods consist of a graphite sleeve containing 14 fuel compacts. Each fuel compact contains about 13,000 coated fuel particles (CFPs) randomly embedded in a graphite matrix (elsewhere is reported that each fuel compact contains about 13,500 particles<sup>a</sup>). Fuel-block assembly is depicted in Figure 1.44. A CFP is comprised of a spherical fuel kernel of low-enriched  $\text{UO}_2$  with a coating of four layers. The core has 12 different uranium enrichments between 3.4 and 9.9 wt.% (as shown in Table 1.11), to reduce the maximum fuel temperature and increase the outlet temperature of the gas. Fuel blocks of more highly enriched uranium are placed in the upper- and outer-core regions. Burnable poisons (BPs), made of boron carbide and carbon, are inserted into two of three holes below the dowel pins in the fuel blocks. The coolant gas flow is downward through annular channels between the graphite blocks and fuel rods. Sixteen pairs of CRs are used for reactivity control. A pair of CRs is individually moved by a CR driving mechanism located in standpipes above the core. The CRs are inserted into two of three channels in the CR guide columns (Ref. 1, pp. 311-312).

The position of blocks in the core is described by a vertical position number and column number. The vertical number ranges from 1 through 9, where the top of the blocks is the 1<sup>st</sup> layer, and the bottom of the blocks is the 9<sup>th</sup> layer. The column number is named according to Figure 1.45. An example is position 4C05, which is the 4<sup>th</sup> block from the top, the 2<sup>nd</sup> ring of fuel from the center of the core, and the 5<sup>th</sup> block from the north in clockwise direction. All 30 fuel columns are grouped concentrically into 4 fuel zones (Ref. 2, p. 14).

The core consists of vertical columns of hexagonal blocks arranged on a uniform triangular pitch (Ref. 15, p. 52). The triangular pitch of the columns on each support block is 362 mm at cold conditions (Ref. 5, p. 75). The effective diameter of the core, including removable reflectors, is 3258 mm. The overall dimensions of the primary components of the core, including permanent reflectors, are a diameter of 4250 mm and a height of 5250 mm (Ref. 2, p. 10).

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<sup>a</sup> K. Minato, H. Kikuchi, T. Tobita, K. Fukuda, M. Kaneko, N. Suzuki, S. Yoshimuta, and H. Tomimoto, "Improvements in Quality of As-Manufactured Fuels for High-Temperature Gas-Cooled Reactors," *J. Nucl. Sci. Tech.*, **34**(3): 325-333 (March 1997).

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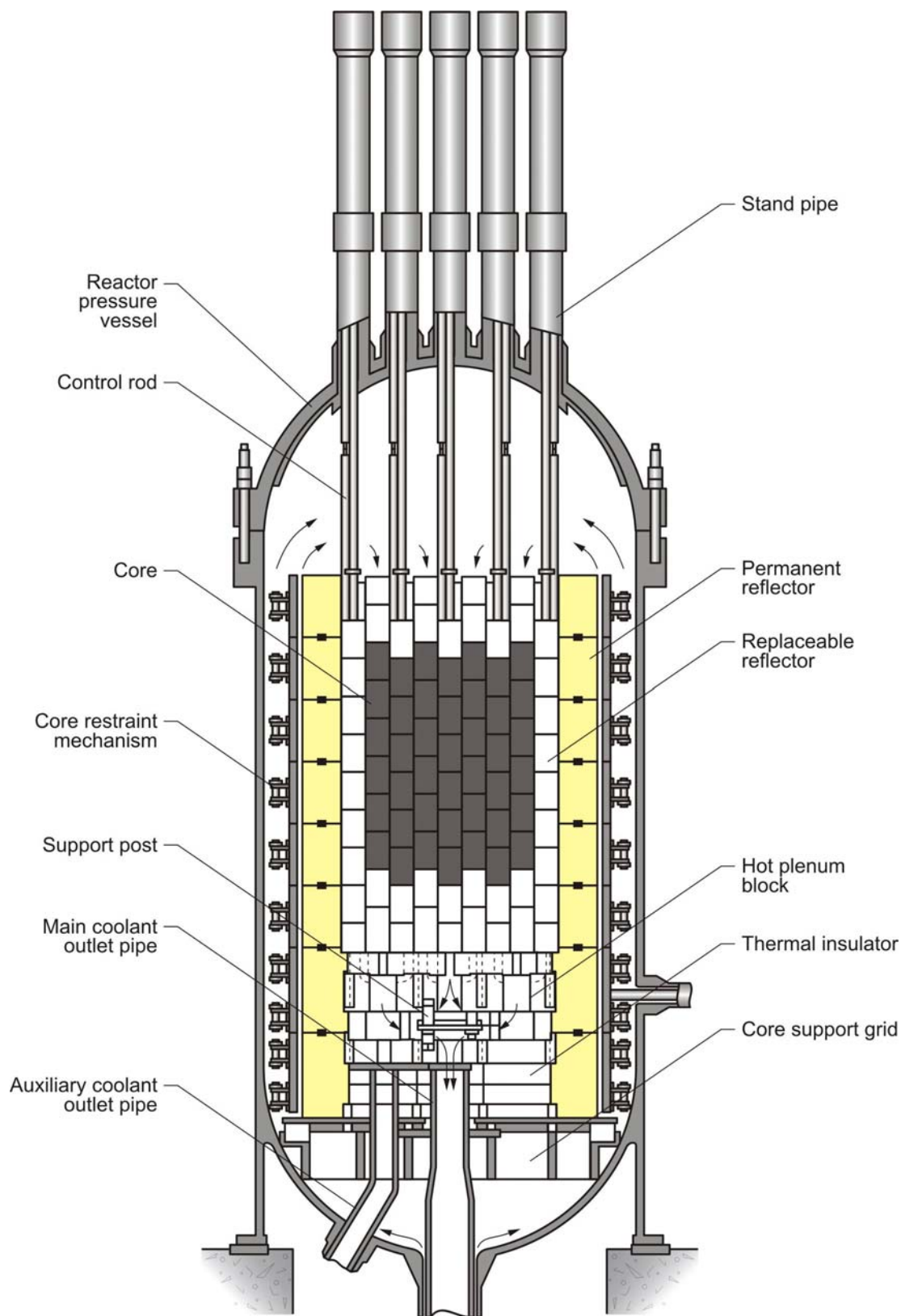
HTTR-GCR-RESR-001  
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Figure 1.42. Vertical Cross Section of the HTTR.

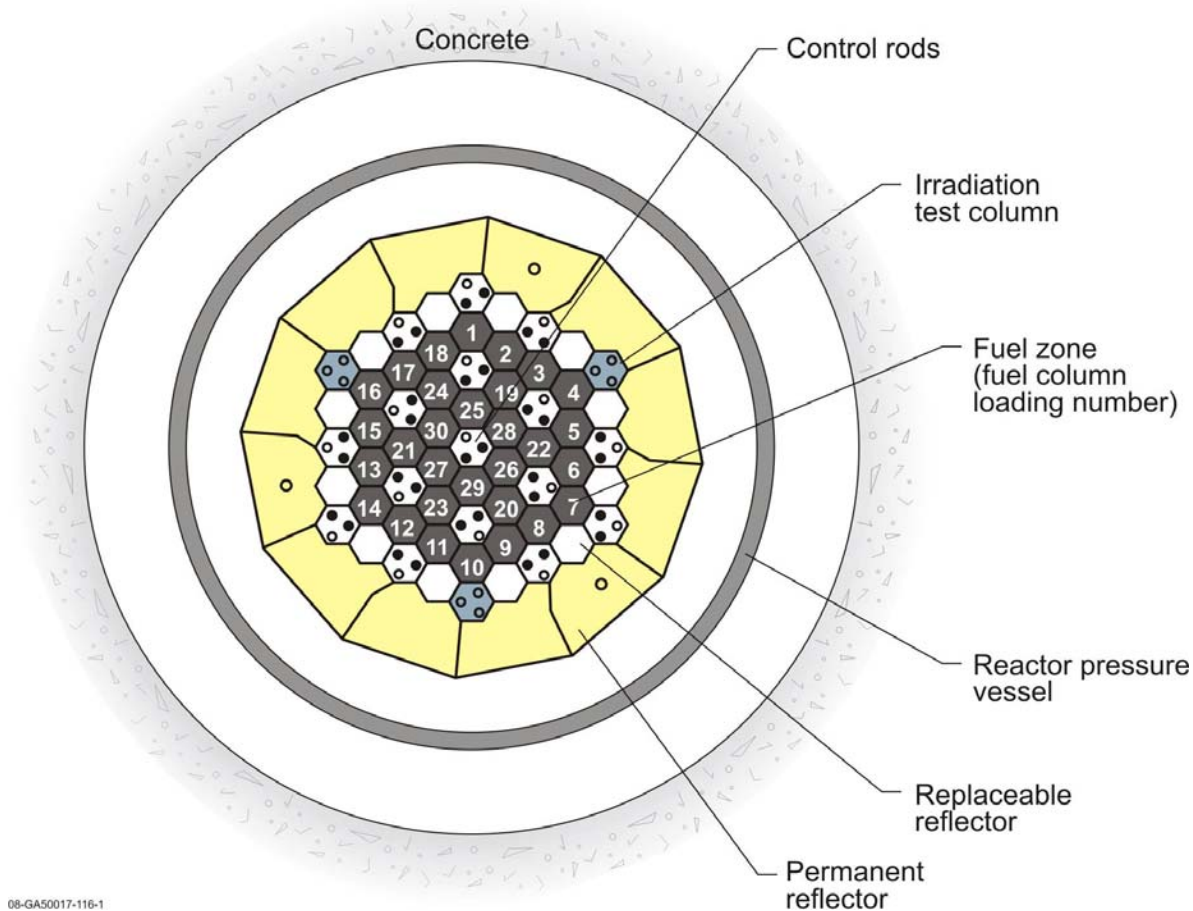


Figure 1.43. Horizontal Cross Section of the HTTR.



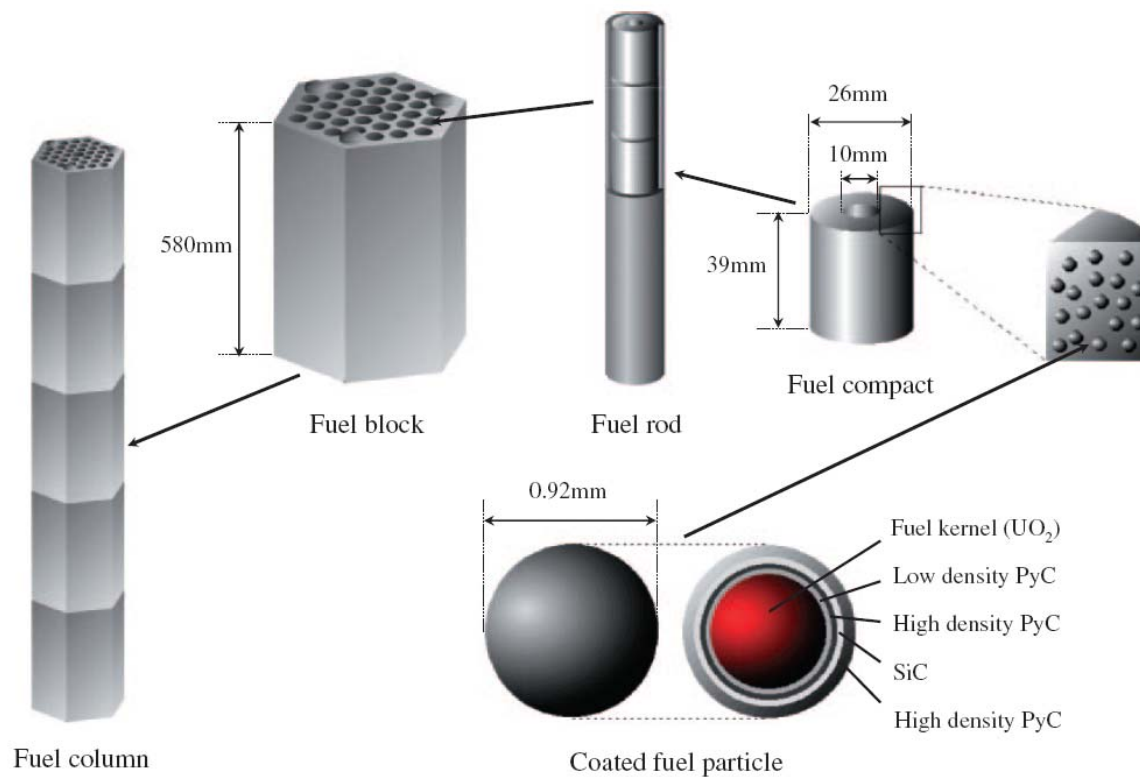
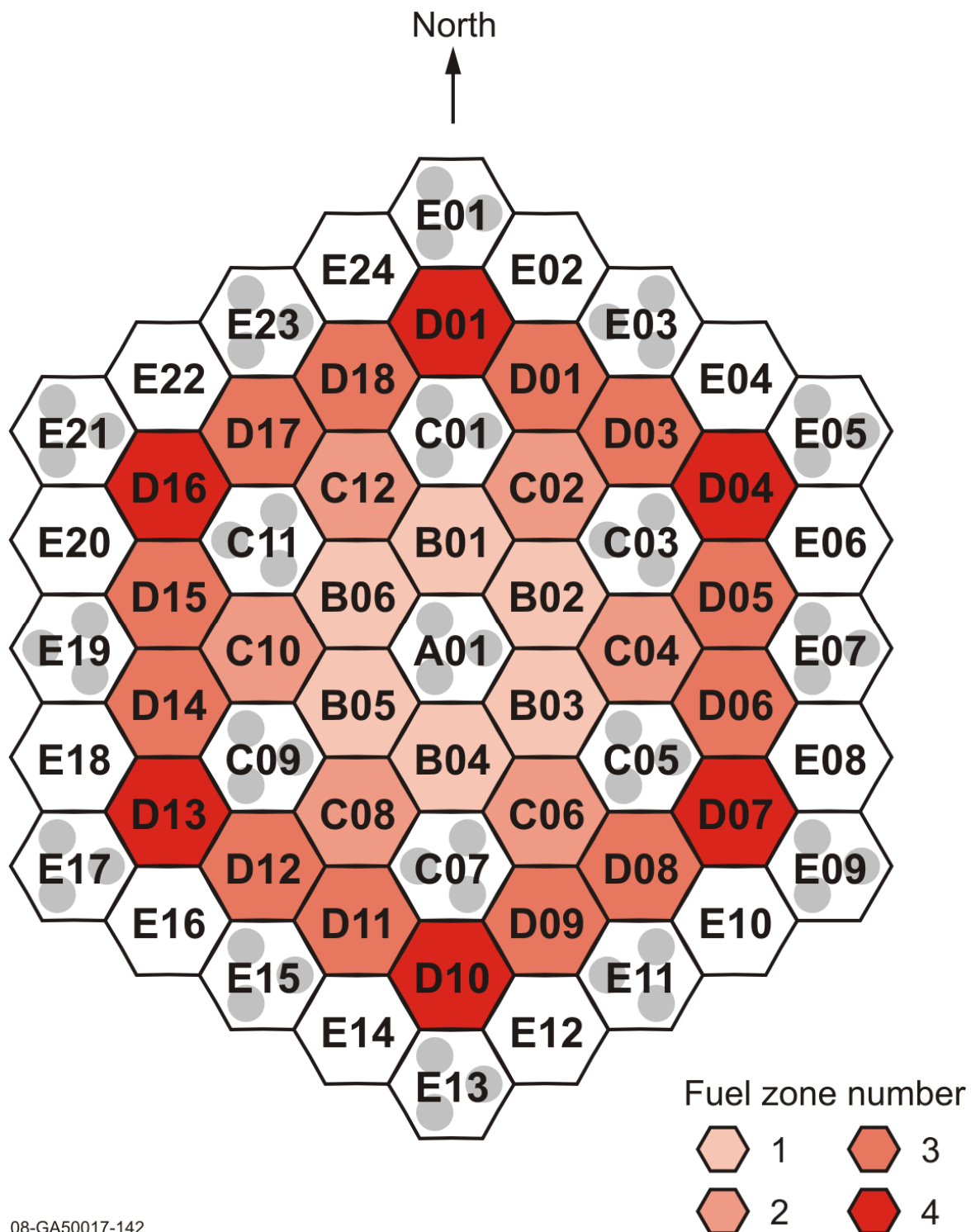


Figure 1.44. HTTR's Fuel Column, Fuel Block, Fuel Rod, Fuel Compact, and Coated Fuel Particle.<sup>a</sup>

<sup>a</sup> M. Goto, N. Nojiri, and S. Shimakawa, "Neutronics Calculations of HTTR with Several Nuclear Data Libraries," *J. Nucl. Sci. Tech.*, **43**(10): 1237-1244 (2006).



08-GA50017-142

Figure 1.45. Fuel Column Name and Zone Number in the HTTR Core (adapted from [Ref. 2, p. 12](#)).

**Prismatic Fuel**

The fuel element of the HTTR is a so-called pin-in-block type fuel element, which is composed of fuel rods in a hexagonal graphite block (Ref. 5, p. 72).

In 1992, Nuclear Fuel Industries Ltd. (NFI) started operating the HTR fuel plant in Tokai Works; based on established technologies, NFI successfully fabricated the first fuel loading of the HTTR from June 1995 to December 1997. The HTTR fuel fabrication consists mainly of the kernel, coated fuel particle, annular fuel compact, fuel rod, and fuel assembly process (Ref. 14, p. 378).

The fabrication of the first-core fuel started in June 1995 and took a total of 33 months. A total of 66,780 fuel compacts in 126 fabrication lots were produced, corresponding to 4770 fuel rods, containing 66,780 fuel compacts, using a total of 900 kg of uranium (Ref. 5, p. 75). In December 1997, 150 fuel assemblies were completely formed and stored in new-fuel storage cells of the HTTR.<sup>a</sup>

Approximately 900 million particles were fabricated for the first fuel loading of the HTTR. The average SiC defective fraction of the fuel compacts was as low as  $8 \times 10^{-5}$  (Ref. 14, p. 378).

A fuel assembly consists of fuel rods, two BP rods, and a fuel graphite block. Each fuel rod (Figure 1.44) is comprised of a graphite sleeve with 14 fuel compacts containing CFPs. The fuel rods are inserted into vertical channels of 41-mm diameter in the fuel graphite block, to form annular coolant channels between the holes and rods. There are two types of fuel graphite blocks: 31- and 33-holed. There are twelve different uranium enrichments in the core (Figure 1.46 and Table 1.11). The enrichment of all compacts in a fuel assembly does not vary. The fuel assembly is classified by the uranium enrichment, number of fuel rods, and the type of BPs (Ref. 2, p. 14).

The core was divided axially into five layers to allow for optimization of the axial power distribution. A layer corresponds to a fuel block. The same uranium enrichments were loaded into the fourth and fifth layers because the power density decreases due to neutron leakage to the bottom reflector. It corresponds to the tail of the exponential form. Therefore the axial power distribution was optimized by decreasing the uranium enrichment ratios from the first layer to the fourth. The radial power distribution was optimized by adjusting the uranium enrichments so that the maximum radial power peaking in each fuel column could be brought close to unity. The uranium enrichment was increased from the core's center to the periphery to compensate for the decrease in the power production caused by neutron leakage into the side reflector.<sup>b</sup>

The fuel blocks were to be completely replaced by fresh fuel blocks after each burnup cycle (Ref. 4, p. 23). Each fuel element in the core will be discharged every three years. The fuel elements keep their original position during their lifetime in the core (Ref. 11, p. 151). However, the HTTR hadn't been refueled for at least the first 10 years of HTTR operation.

There are four types of HTTR fuel. The A-type fuel is the primary driver fuel for the HTTR. B-type fuel rods, namely B-1, B-2, and B-3, have different coating layer specifications for the CFPs, and are used in irradiation tests of advanced fuels. Fuel specifications are in Table 1.12 (Ref. 15, p. 46). The A-type fuel is currently the only fuel in use as the B-type fuel has yet to be fabricated.

Specifications for the fuel assemblies are compiled into Table 1.13.

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<sup>a</sup> K. Sawa and S. Ueta, "Research and Development on HTGR Fuel in the HTTR Project," *Nucl. Eng. Des.*, **233**: 163-172 (2004).

<sup>b</sup> K. Yamashita, R. Shindo, I. Murata, S. Maruyama, N. Fujimoto, and T. Takeda, "Nuclear Design of the High-Temperature Engineering Test Reactor (HTTR)," *Nucl. Sci. Eng.*, **122**: 212-228 (1996).

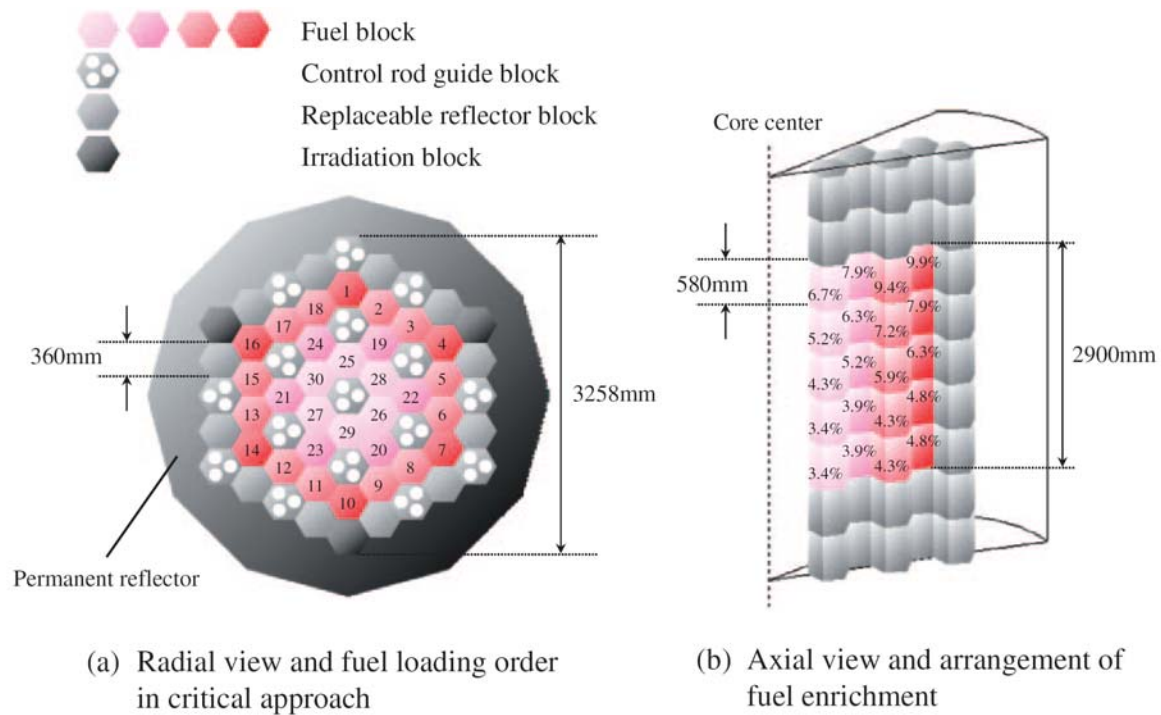
Figure 1.46. Uranium Enrichments of the HTTR Core.<sup>a</sup>

Table 1.11. Uranium Enrichments of the HTTR Core (Ref. 4, p. 31).

Layer <sup>(a)</sup>	Fuel Zone Number <sup>(b)</sup>				BP <sup>(c)</sup>
	1	2	3	4	
1	6.7	7.9	9.4	9.9	2.0
2	5.2	6.3	7.2	7.9	2.5
3	4.3	5.2	5.9	6.3	2.5
4	3.4	3.9	4.3	4.8	2.0
5	3.4	3.9	4.3	4.8	2.0

(a) Layer number from top fuel block.

(b)  $^{235}\text{U}$  enrichment (wt.%).

(c) Nat-B concentration (wt.%).

<sup>a</sup> M. Goto, N. Nojiri, and S. Shimakawa, "Neutronics Calculations of HTTR with Several Nuclear Data Libraries," *J. Nucl. Sci. Tech.*, **43**(10): 1237-1244 (2006).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE

Table 1.12. Specification of Coated Fuel Particles ([Ref. 5 p. 74](#), [Ref. 15, p. 48](#)).

A-type fuel is the primary fuel for the HTTR with B-type fuel  
 representing advanced fuel for irradiation tests.

	A-Type	B-1/B-2 Type	B-3 Type
Fuel Type	Rod	Rod	Rod
Fuel Coating Type	TRISO	TRISO	TRISO
Diameter of Particle ( $\mu\text{m}$ )	920	940	830
Fuel Kernel			
Material	UO <sub>2</sub>	UO <sub>2</sub>	(U,Th)O <sub>2</sub> (Th/U=4)
Density (% of T.D.)	95	95	95
Diameter ( $\mu\text{m}$ )	600	570	500
Materials and Thickness ( $\mu\text{m}$ ) of Coatings			
1 <sup>st</sup> Layer	Low Density PyC 60	Low Density PyC 80	Low Density PyC 60
2 <sup>nd</sup> Layer	High Density PyC 30	High Density PyC 30	High Density PyC 30
3 <sup>rd</sup> Layer	SiC 25	SiC(B-1)/ZrC(B-2) 35	SiC 30
4 <sup>th</sup> Layer	High Density PyC 45	High Density PyC 40	High Density PyC 45
Enrichment of <sup>235</sup> U (wt.%)	3-10 (Ave. 6)	5	20

Table 1.13. Main Specifications of HTTR Fuel ([Ref. 5, p. 74](#) and [Ref. 4, p. 25](#)).

Fuel Kernel	
Material	UO <sub>2</sub>
Diameter (μm)	600
Density (g/cm <sup>3</sup> )	10.41 <sup>(a)</sup>
Coated Fuel Particle	
Type / Material	TRISO
Diameter (μm)	920
Impurity (ppm)	<3 (Boron Equivalent)
Fuel Compact	
Type	Hollow Cylinder
Material	CFPs, Binder, and Graphite
Outer / Inner Diameter (cm)	2.6 / 1.0
Length (cm)	3.9
Packing Fraction of CFPs (vol. %)	30 (A and B-3) 35 (B-1 and B-2)
Density of Graphite Matrix (g/cm <sup>3</sup> )	1.7
Impurity in Graphite Matrix (ppm)	<1.2 (Boron Equivalent)

(a) The density reported in this reference is not the same as that reported in Table 1.14.



Table 1.13 (cont.). Main Specifications of HTTR Fuel (Ref. 5, p. 74 and Ref. 4, p. 25).

Fuel Rod	
Outer Diameter (cm)	3.4
Sleeve Thickness (mm)	3.75
Length (cm)	54.6
Number of Fuel Compacts	14
Number of Rods in a Block	31 / 33
Graphite Sleeve	
Type	Cylinder
Material	IG-110 Graphite
Impurity (ppm)	<1 (Boron Equivalent)
Length (cm)	58
Gap Width between Compact and Sleeve (mm)	0.25
Graphite Block	
Type / Configuration	Pin-in-Block / Hexagonal
Material	IG-110 graphite
Width across Flats (cm)	36
Height (cm)	58
Fuel Hole Diameter (cm)	4.1
Density (g/cm <sup>3</sup> )	1.75
Impurity (ppm)	<1 (Boron Equivalent)

A CFP consists of a spherical fuel kernel of low enriched  $\text{UO}_2$  (600  $\mu\text{m}$  diameter at 95 % theoretical density, TD) with a tri-isotropic (TRISO) coating: low density, porous pyrolytic carbon (PyC) buffer layer (60  $\mu\text{m}$ ), high density inner isotropic PyC layer (30  $\mu\text{m}$ ), a SiC layer (25  $\mu\text{m}$ ), and a final outer PyC layer (45  $\mu\text{m}$ ), as shown in Figure 1.47 and Table 1.12. The CFPs are embedded in graphite matrix of the fuel compact, as shown in Figure 1.44. The fuel compact is a hollow cylinder with 10-mm inner diameter, 26-mm outer diameter, and a 39-mm height (Ref. 2, p. 14-16 and Ref. 5, p. 74).

Buffer and PyC layers are obtained from the thermal decomposition of  $\text{C}_2\text{H}_2$  and  $\text{C}_3\text{H}_6$ , respectively, and the SiC layer from  $\text{SiCH}_3\text{Cl}_3$ . The defective fraction of SiC layers was minimized by reducing thermal and mechanical shocks during the process. The advanced fluidized bed for this process is able to coat about three million particles uniformly. The SiC defective fraction of CFPs is on the order of  $10^{-7}$ , less than three defective particles in one batch (Ref. 14, p. 378).

The porous PyC coating layer, called the buffer layer, attenuates fission recoils and provides void volume for gaseous fission products and carbon monoxide. The IPyC coating layer acts as a containment for the gases. The SiC coating layer provides mechanical strength for the particle and acts as a diffusion barrier to metallic fission products that diffuse easily through the IPyC layer. The OPyC coating layer protects the SiC coating layer mechanically.

Uncertainties in the fabrication of the first-loading fuel of the HTTR were assessed and characterized. Table 1.14 contains major specifications and uncertainties in the initial HTTR fuel. Figure 1.48 shows

measured dimensions and distributions of as-fabricated coated fuel particles. The thickness of the coating layers was measured by optical microscopy. Further discussion of the uncertainties in fuel fabrication and uranium enrichment is discussed elsewhere.<sup>a, b</sup>

The sphericity and deviation of kernel diameter has a profound effect on the strength and defective fraction of coated fuel particles. Therefore, much effort was focused on achieving high-quality mass-produced kernels using an ammonia diuranate dropping system capable of producing 4.3 kg of uranium per batch. The average sphericity was less than 1.05 and the standard deviation of the diameter was less than 3 % of the average diameter of the HTTR first loading fuel fabrication (Ref. 14, p. 378).

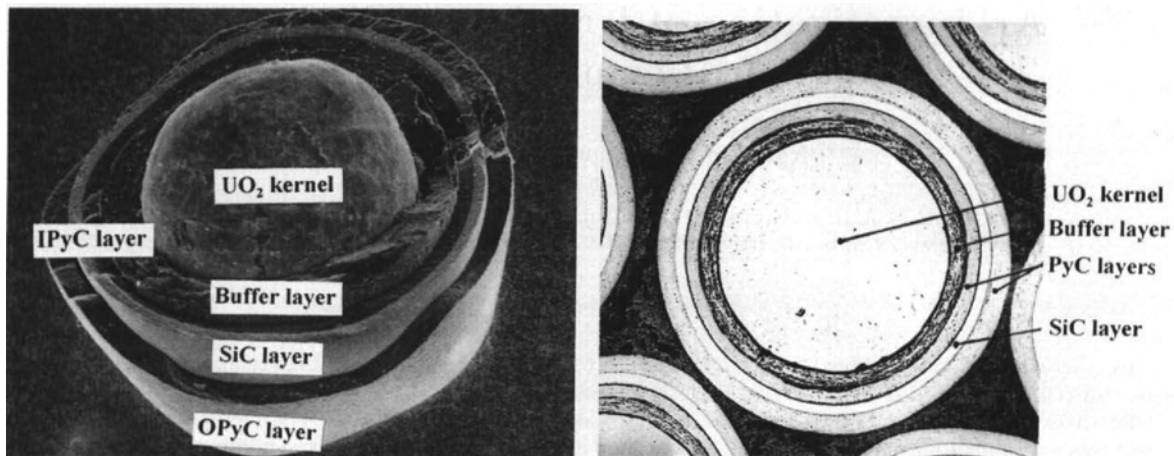


Figure 1.47. TRISO-Coated Fuel Particle.<sup>a</sup>

<sup>a</sup> K. Sawa and K. Minato, "An Investigation of Irradiation Performance of High Burnup HTGR Fuel," *J. Nucl. Sci. Tech.*, **36**(9): 781-791 (September 1999).

<sup>b</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATETable 1.14. Major Specifications of the First-Loading Fuel of the HTTR. <sup>(a)</sup>

Fuel Kernel	
Diameter ( $\mu\text{m}$ )	$600 \pm 55^{(b)}$
Density ( $\text{g}/\text{cm}^3$ )	$10.63 \pm 0.26$
Impurity (ppm)	$\leq 3 \text{ EBC}^{(c)}$
Coating Layer	
Buffer Thickness ( $\mu\text{m}$ )	$60 \pm 12$
IPyC Thickness ( $\mu\text{m}$ )	$30 \pm 6$
SiC Thickness ( $\mu\text{m}$ )	$25^{+12}_{-0}$
OPyC Thickness ( $\mu\text{m}$ )	$45 \pm 6$
Buffer Density ( $\text{g}/\text{cm}^3$ )	$1.10 \pm 0.10$
IPyC Density ( $\text{g}/\text{cm}^3$ )	$1.85^{+0.10}_{-0.05}$
SiC Density ( $\text{g}/\text{cm}^3$ )	$\geq 3.20$
OPyC Density ( $\text{g}/\text{cm}^3$ )	$1.85^{+0.10}_{-0.05}$
OPTAF <sup>(d)</sup> of PyC Layers	$\leq 1.03$
Coated Fuel Particle	
Diameter ( $\mu\text{m}$ )	$920^{+50}_{-30}$
Sphericity	$\leq 1.2$
Fuel Compact	
CFP Packing Fraction (vol. %)	$30 \pm 3$
Impurity (ppm)	$\leq 5 \text{ EBC}^{(c)}$
Exposed Uranium Fraction	$\leq 1.5 \times 10^{-4}$
SiC-Failure Fraction	$\leq 1.5 \times 10^{-3}$
Outer Diameter (mm)	$26.0 \pm 0.1$
Inner Diameter (mm)	$10.0 \pm 0.1$
Height (mm)	$39.0 \pm 0.5$
Matrix Density ( $\text{g}/\text{cm}^3$ )	$1.70 \pm 0.05$
Compressive Strength (N)	$\geq 4,900$
Fuel Rod	
Uranium Content (g)	$188.58 \pm 5.66$
Total Length (mm)	$577 \pm 0.5$
Fuel Compact Stack Length (mm)	$\geq 544$

- (a) K. Sawa, T. Tobita, H. Mogi, S. Shiozawa, S. Yoshimuta, S. Suzuki, and K. Deushi, "Fabrication of the First-Loading Fuel of the High Temperature Engineering Test Reactor," *J. Nucl. Sci. Tech.*, **36**(8): 683-690 (August 1999).
- (b) The density reported in this reference is not the same as that reported in Table 1.13.
- (c) EBC: Equivalent Boron Content
- (d) OPTAF: Optical Anisotropy Factor

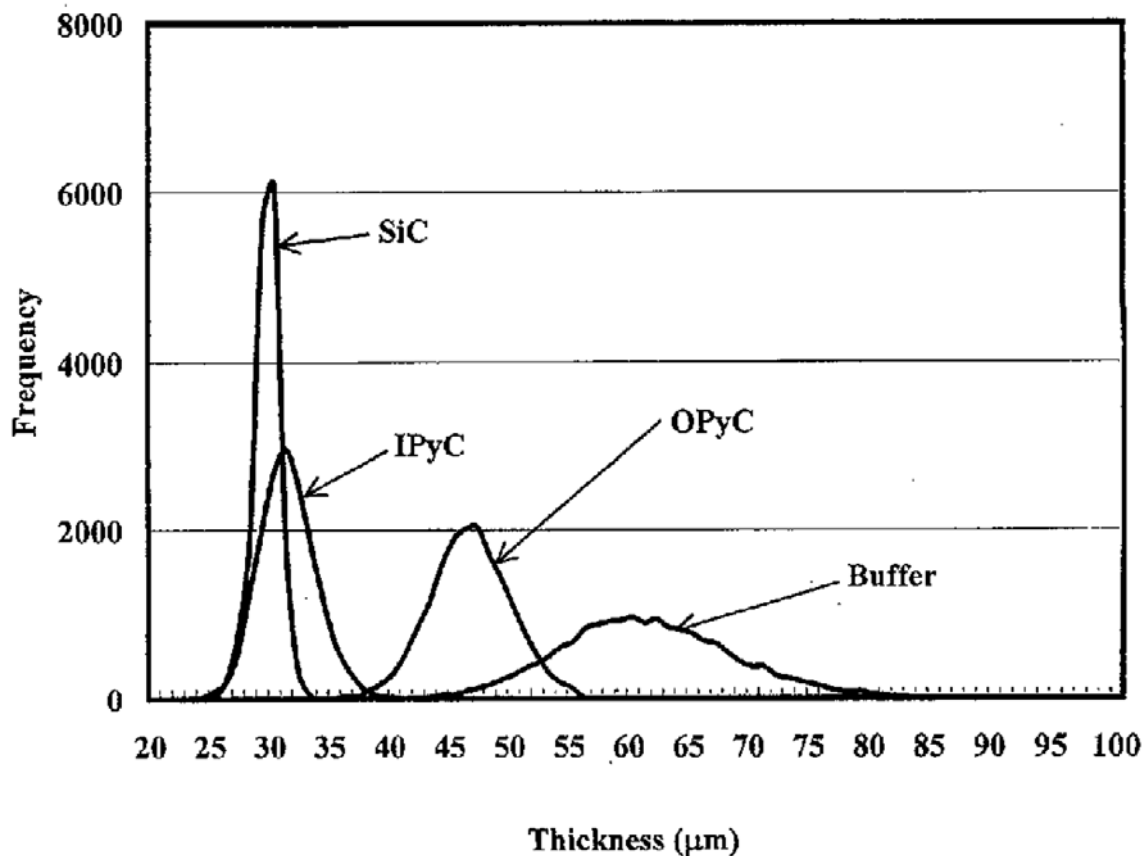


Figure 1.48. Measured Dimensions and Distributions of As-Fabricated Coating Layers.<sup>a</sup>

In order to create the fuel compact, the coated fuel particles are first overcoated with a graphite matrix. This process prevents close contact between the coated fuel particles. Hot pressing is utilized to obtain annular green compacts. The automatic hot pressing system weighs the overcoated particles, preheats the resin, charges the particles into dies, hot presses them, discharges the compacts, and then numbers each compact. This system can produce 230 green compacts in eight hours. The green compacts are carbonized in nitrogen gas and degassed in vacuum (Ref. 14, p. 378).

With about 13,000 fuel particles defined for each fuel compact, the corresponding packing density is about 30 % (Ref. 2, p. 136).

A black and white photo of some fuel compacts are shown in Figure 1.49. A horizontal cross-section view of a fuel compact with CFPs is shown in Figure 1.50.

<sup>a</sup> K. Sawa, T. Tobita, H. Mogi, S. Shiozawa, S. Yoshimuta, S. Suzuki, and K. Deushi, "Fabrication of the First-Loading Fuel of the High Temperature Engineering Test Reactor," *J. Nucl. Sci. Tech.*, **36**(8): 683-690 (August 1999).

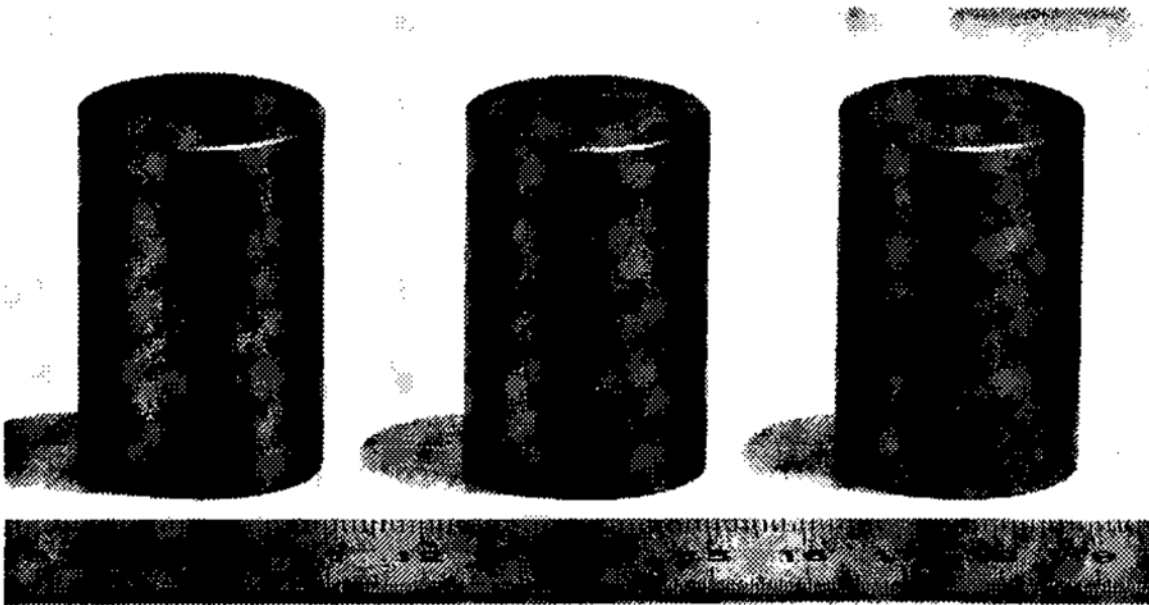


Figure 1.49. External View of Fuel Compacts.<sup>a</sup>

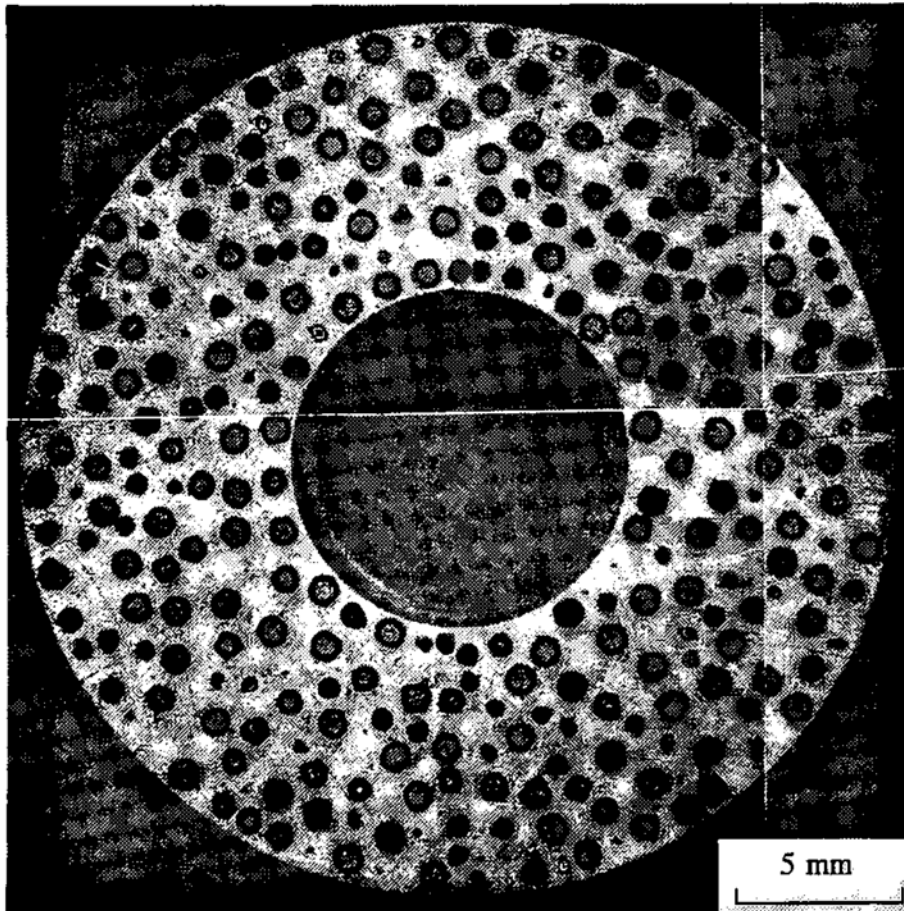


Figure 1.50. Horizontal Cross Section of a Fuel Compact.<sup>a</sup>

<sup>a</sup> K. Minato, H. Kikuchi, T. Tobita, K. Fukuda, M. Kaneko, N. Suzuki, S. Yoshimuta, and H. Tomimoto, "Improvements in Quality of As-Manufactured Fuels for High-Temperature Gas-Cooled Reactors," *J. Nucl. Sci. Tech.*, **34**(3): 325-333 (March 1997).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

The sintered fuel compacts are contained in a fuel rod that is 34 mm in outer diameter and 577 mm in length (Ref. 3, p. 15). The form of the fuel rod (graphite sleeve containing 14 fuel compacts) is shown in Figure 1.51 (Ref. 2, p. 16). The annular fuel pin has an inner diameter of 26.25 mm<sup>a</sup>. The helium gas flows downward through the 3.5 mm annular gap between the hole and the fuel rod (Ref. 5, pp. 74-75).

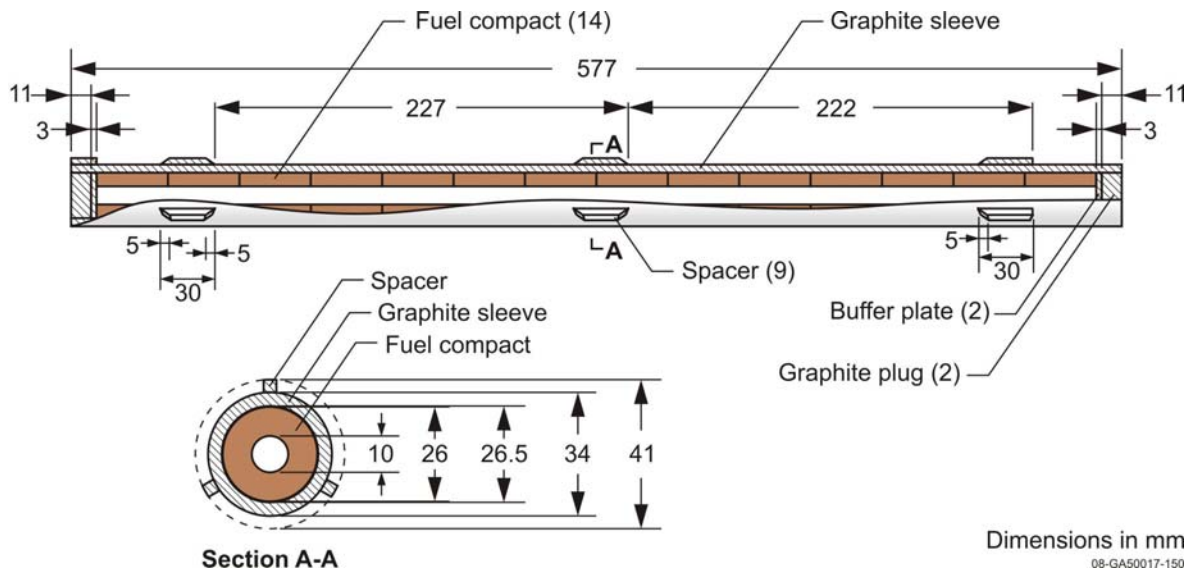


Figure 1.51. HTTR Fuel Rod.

The graphite block is a prismatic hexagonal block with a 580-mm height and 360-mm across the flats. Blocks in fuel zones 1 and 2 have 33 fuel holes, and blocks in fuel zones 3 and 4 have 31 fuel zones. The form of the block is given in Figures 1.52 and 1.53, respectively (Ref. 2, p. 18 and 19).

Ninety-nine percent of the total power is produced in the active core and the residual 1 % in the reflector region (Ref. 15, p. 14). Approximately 95 % of the thermal power in the active core is generated in the fuel elements and the rest in the graphite moderator. Thermal and irradiation conditions thus become severe for the blocks at the outermost region of the core. For this reason, the fuel blocks only contain 31 fuel rods instead of 33, maintaining their structural integrity.<sup>b</sup>

Three threaded dowels are installed on the top face of the fuel blocks that are coupled with sockets in the bottom face of the block located above it. A hole at the center of each fuel block is provided for fuel handling. The hole profile is shaped so that a lifting ledge is machined at the lower end. The grapple head of the fuel handling machine withstands against this ledge when lifting the block. Additional holes are provided in the corners of the blocks for the insertion of the BP rods (Ref. 5, p. 76 and Ref. 15, p. 53).

The arrangement pattern of the fuel blocks in the core is shown in Figure 1.54, and a photo depicting fuel loading of a graphite block is shown in Figure 1.55. There is a slight key slot in the holes of the fuel rods to align and hold in the fuel rods (Figure 1.56).

<sup>a</sup> "Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, Vienna (November 1997).

<sup>b</sup> E. Takada, S. Nakagawa, N. Fujimoto, and D. Tochio, "Core Thermal-Hydraulic Design," *Nucl. Eng. Des.*, **233**: 37-43 (2004).



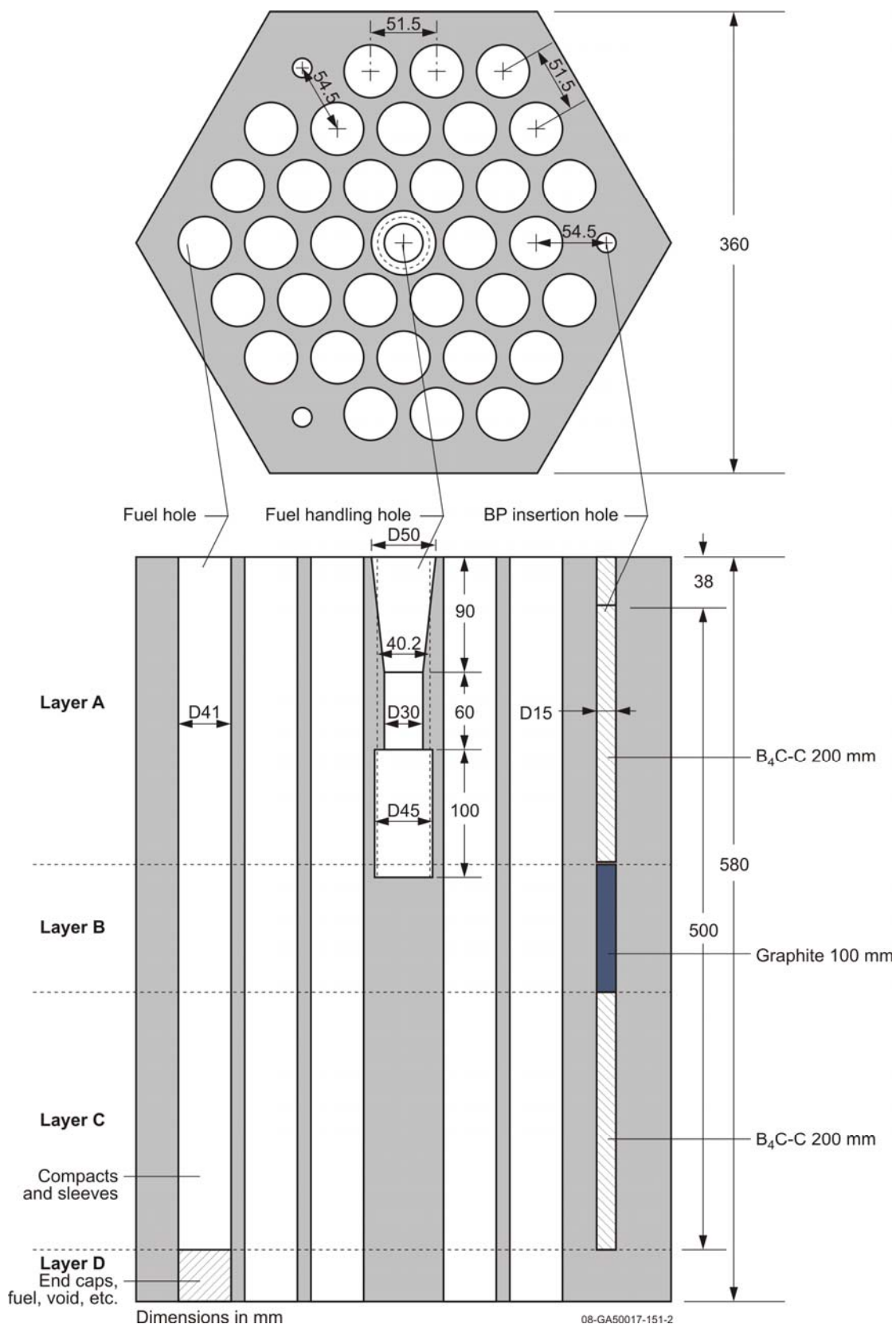


Figure 1.52. Fuel Block for 33-Pin Fuel Assembly.  
Dxx represents the diameter in xx (mm).

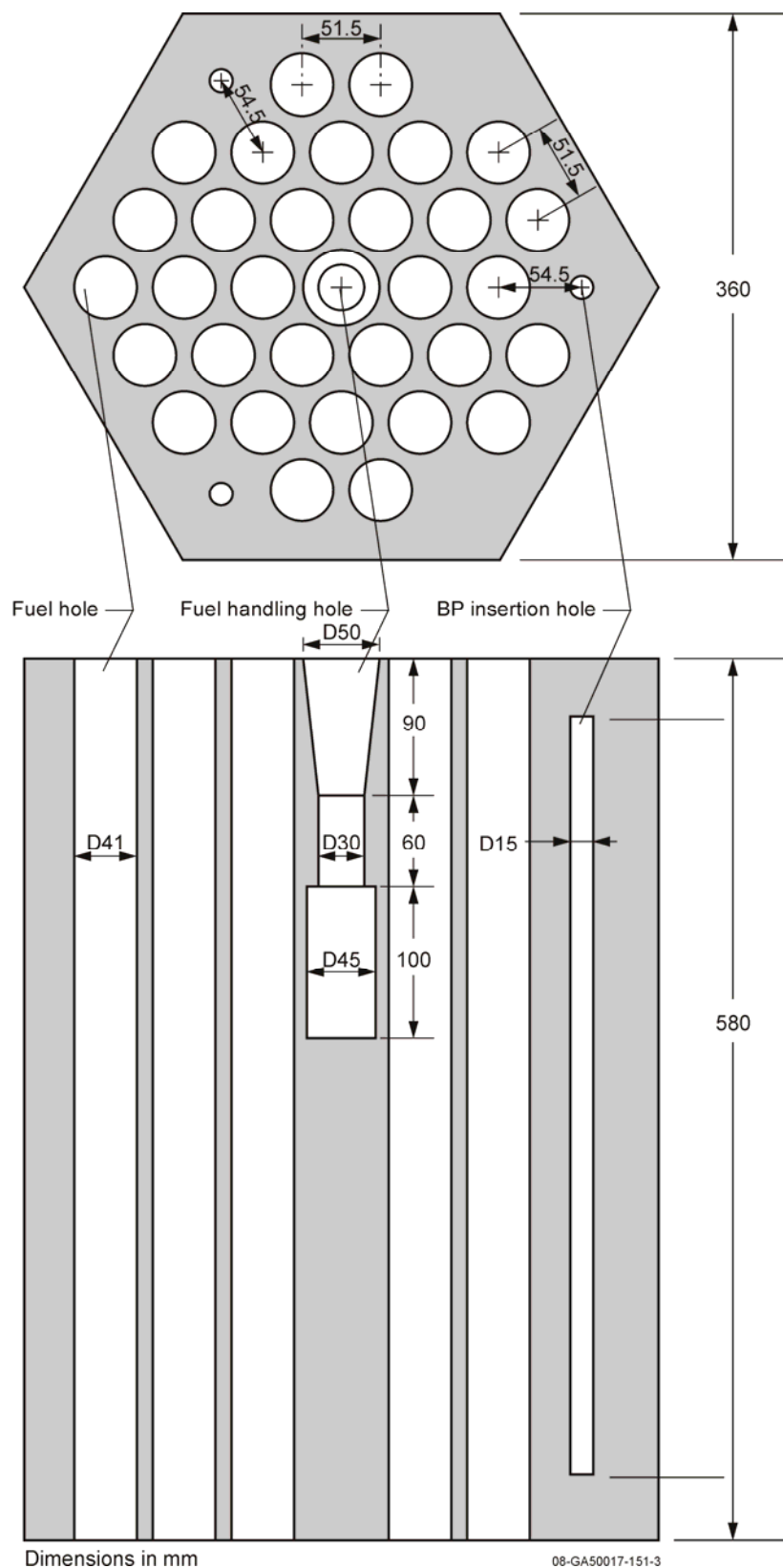
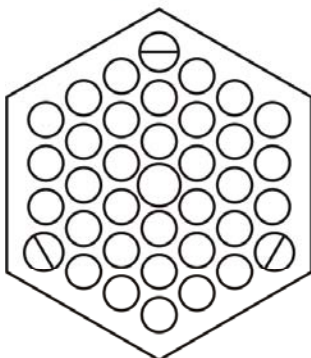
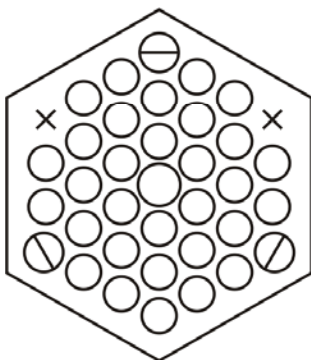
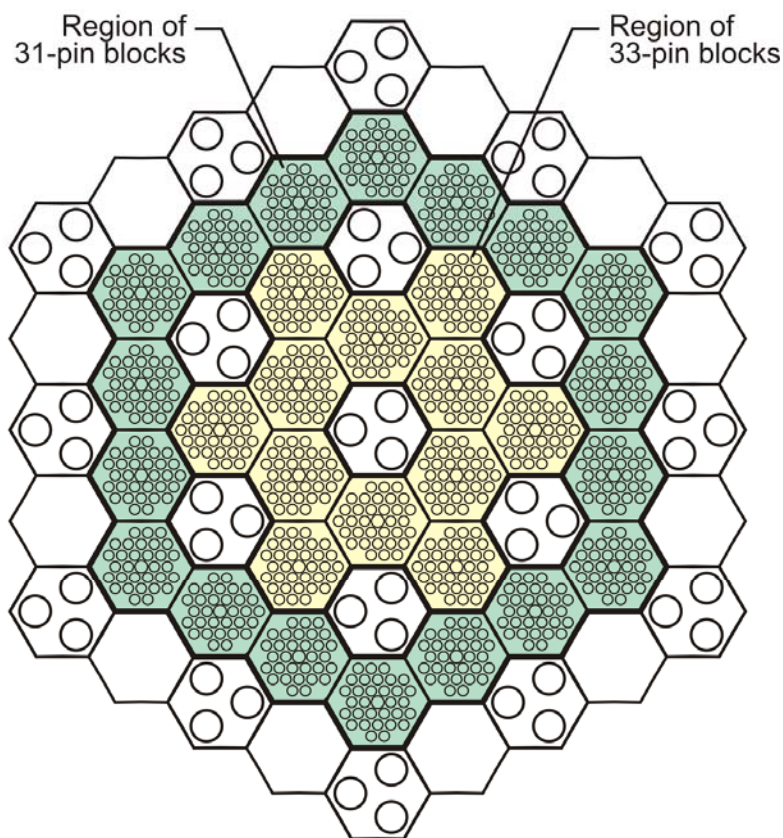


Figure 1.53. Fuel Block for 31-Pin Fuel Assembly.  
Dxx represents the diameter in xx (mm).

**Fuel rod arrangement  
for 33-pin block****Fuel rod arrangement  
for 31-pin block**

↓ Direction of  
core center

**Fuel block arrangement**

08-GA50017-233

Figure 1.54. Core Arrangement Plan of Fuel Blocks with 33 and 31 Fuel Rods.<sup>a</sup>

<sup>a</sup> Y. Sudo, S. Saito, M. Yamada, M. Uno, and T. Ohno, "Design Considerations and Improvements of the High Temperature Engineering Test Reactor (HTTR) for a Reactor Outlet Coolant Temperature of 950 °C," *Proc. Int. Working Group on Gas-Cooled Reactors (IWGGCR-19)*, San Diego, California (September 21-23, 1988).



Figure 1.55. Assembly of a Fuel Element Block.<sup>a</sup>

## HTTR Fuel Block

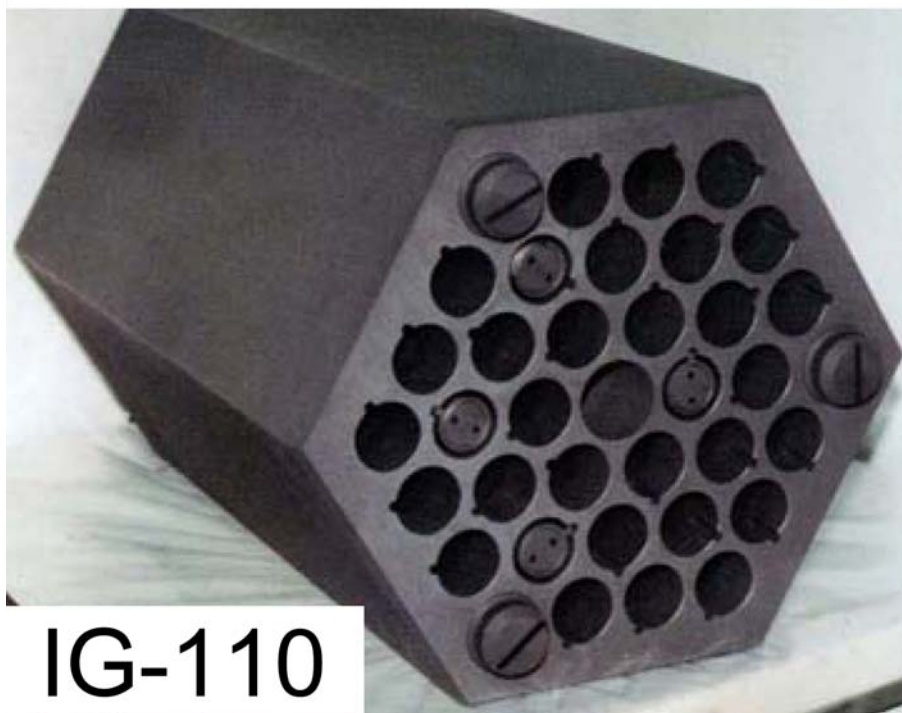


Figure 1.56. Close-Up View of a Fuel Assembly Block.<sup>b</sup>

<sup>a</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

<sup>b</sup> S. Shiozawa, "Status of Japan's HTR Program," *Key Note Presentation at the 2007 Int. Congress on Advances in Nuclear Power Plants (ICAPP 2007)*, Nice, France (May 13-18, 2007).

### **Burnable Poisons**

BP rods are inserted into two of the three BP insertion holes in the fuel graphite blocks (Figure 1.57). The BP insertion holes are under the 3 dowel pins of the fuel graphite block and are 15 mm in diameter and 500 mm long. One hole is empty, without the BP rod. Figures 1.52 and 1.53 show the fuel block positions of the holes that are filled with BPs. The BP rod has a diameter of 14 mm. The rod contains BP pellets and graphite disks, as shown in Figure 1.58. The graphite disks are placed between the BP pellets. The BP cylindrical pellets are each 20 mm in height (Ref. 2, p. 21).

The large initial excess reactivity from fresh fuel is counteracted through the use of BPs. Optimization of the specifications for the BPs reduces deviation from the optimum power distribution due to burnup effects. This is done by changing the poison atom density and radius throughout the core. It then becomes possible to operate the reactor during full power operation at 950 °C without changing the insertion position of the control rods (Ref. 3, p. 24).

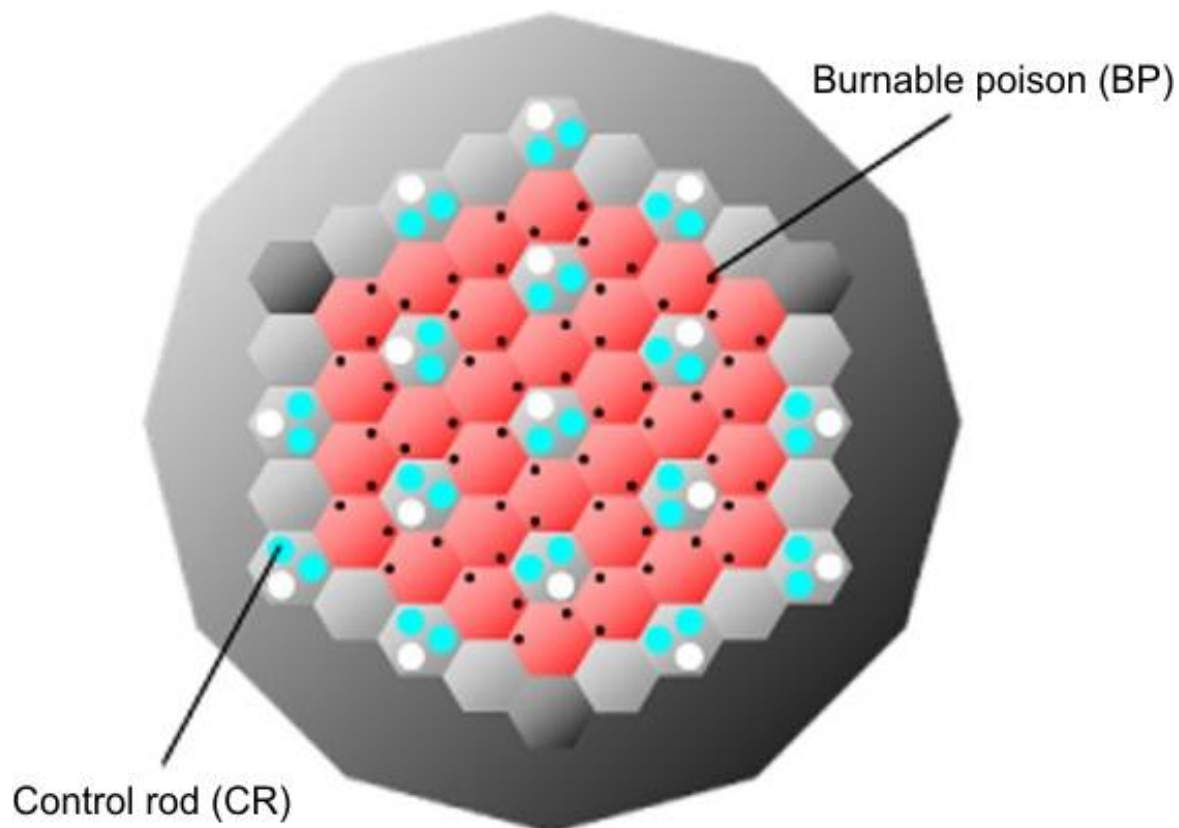


Figure 1.57. Location of BP Pellets in the HTTR.<sup>a</sup>

<sup>a</sup> S. Shimakawa and K. Kunitomi, "Validation of Reactor Physics Codes used for HTTR and Design-Phase HTGRs at JAEA," *1<sup>st</sup> Int. Verification and Validation for Nuclear Systems Analysis Workshop*, Idaho Falls, Idaho, July 21-25, 2008.



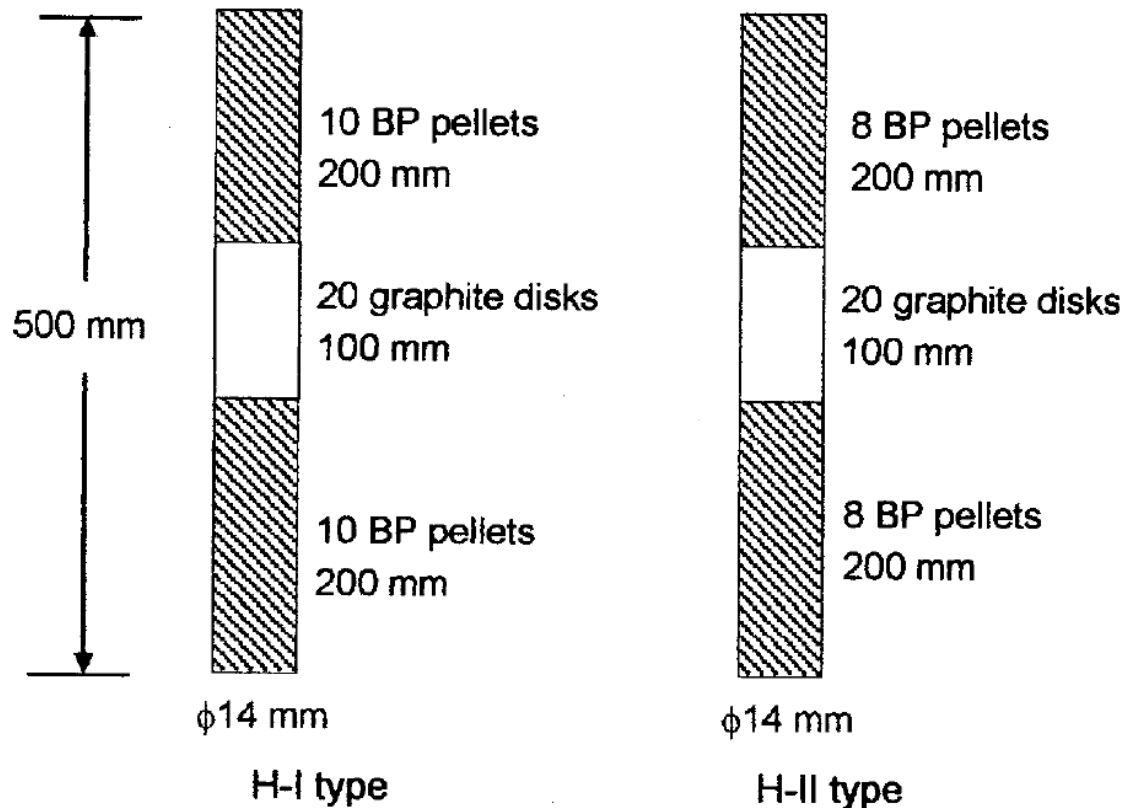


Figure 1.58. Configuration of BP Pellets and Graphite Disks in a BP Rod (Ref. 2, p. 17).

### Control Rods and Reserve Shutdown System

Toshiba and Ishikawajima-Harima Heavy Industries led development of the reactivity control instrumentation: the control rod drive mechanism and the reserve shutdown system (Ref. 14, p. 380).

There are 16 pairs of CRs: seven in the fuel region and nine in the replaceable reflector region. They are inserted into the holes of the CR guide columns. The control rod guide blocks have three holes: two for control rod insertion, and one for the emergency reserve shutdown system (RSS). Figure 1.43 shows the position of the CR insertion holes and RSS holes in the core and replaceable reflector region (Ref. 2, p. 21).

The center control rod can be removed such that the center column of the core could be used for an irradiation test. The control rod system can achieve subcriticality from any operation state and maintain subcriticality in the cold core even when a pair of control rods sticks into an operational position. In case of a scram during normal operations, the control rods in the replaceable reflector region are inserted at first, and the rest of the control rods are inserted after the core has cooled down. This prevents the control rod cladding from being exposed to temperatures above 900 °C. A pair of control rods is driven by one drive mechanism and released from the mechanism to be inserted by gravity in the event of a reactor scram.<sup>a</sup>

A pair of control rods is individually moved by a control rod drive mechanism located in stand-pipes connected to the hemispherical top head closure of the RPV (Ref. 2, p. 13).

<sup>a</sup> K. Kunitomi and S. Shiozawa, "Safety Design," *Nucl. Eng. Des.*, **233**: 45-58 (2004).



The control rod drive mechanism (Figure 1.59) consists of an AC motor, a decelerator, an electromagnetic clutch, a velocity-limiting brake, a shock absorbing mechanism, position indicators, manual control mechanisms, gears, and a control rod support cable. The drive motor is coupled to the cable drum through the decelerator, the electromagnetic clutch, and the gears. During normal operation, the position of the control rod is sustained by the torque of the motor. The maximum withdrawal velocity is limited to a value below 70 mm/s by the decelerator even when the motor is running at the maximum speed. When the electric current through the electromagnetic clutch is cut off by a scram signal, the clutch is separated to insert the control rod by gravity into the core. To protect the control rod support cable from overloading, the velocity-limiting brake maintains the insertion speed constant during a reactor scram by applying the braking torque in proportion to the speed. The shock absorbing mechanism reduces the speed of the control rods at the last stage of the insertion by using a gas damper to absorb any impact when the control rods are inserted for more than 80 % of their full stroke. The position indicator has three independent systems using synchro-mesh transmitters, as a built-in safety factor ([Ref. 7, pp. 91-92](#)).

The control rods were designed to provide a shutdown margin of more than 0.01  $\Delta k/k$ , even when one pair of rods with the maximum reactivity worth is completely withdrawn and cannot be inserted. The reserve shutdown system also is designed to provide more than 0.01  $\Delta k/k$ , even when the control rod system is unavailable. The maximum reactivity addition rate is limited to control rod withdrawal of less than 50 mm per step and a reactivity addition rate limit of  $2.4 \times 10^{-4}$   $\Delta k/k\cdot s$ . The maximum driving speed of the control rod is 60 cm/min ([Ref. 3, p. 24 and 31](#)).

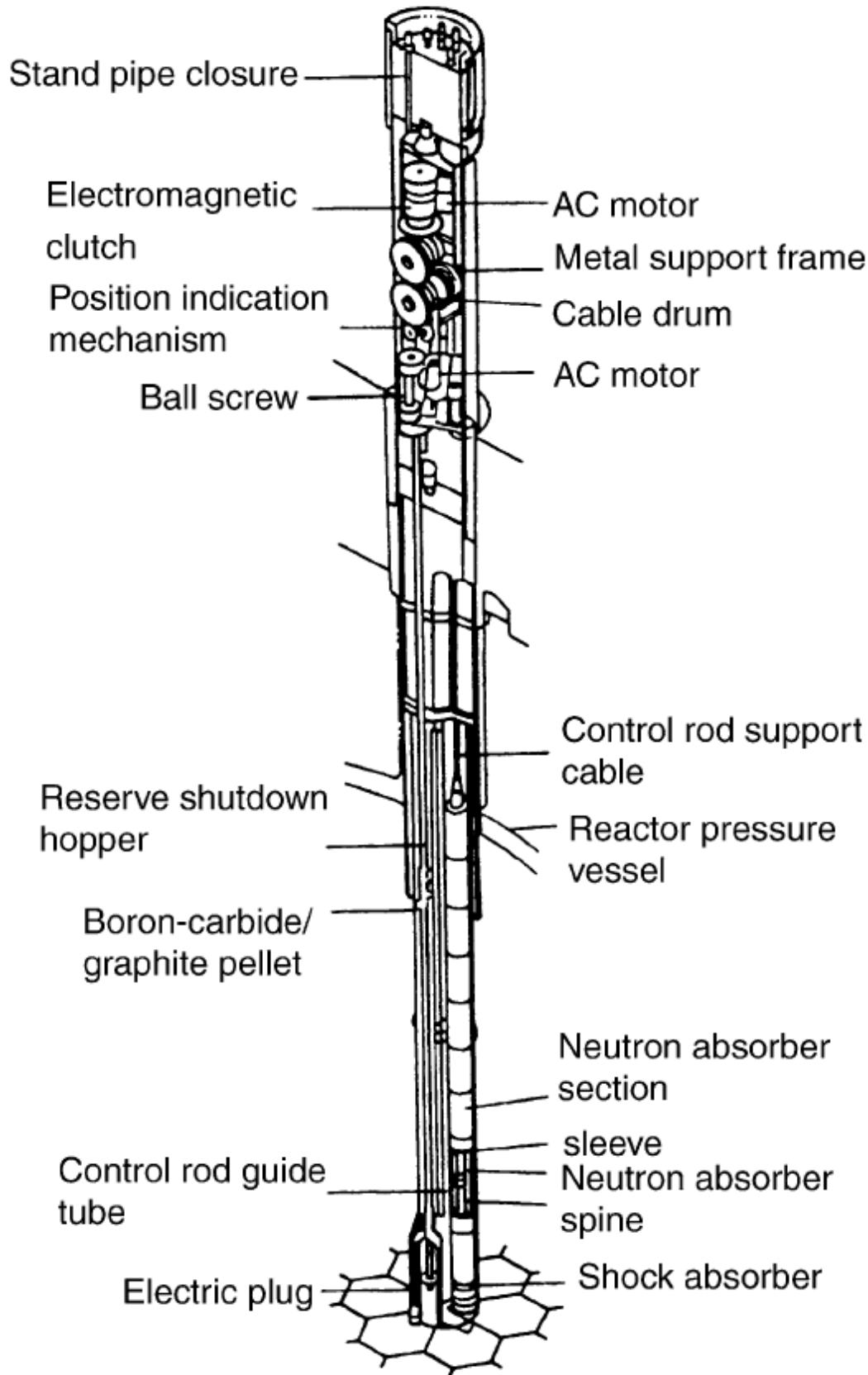


Figure 1.59. Control Rod System of the HTTR (Ref. 7, p. 91).

Each CR consists of 10 neutron absorber sections connected with metallic spines and support rings (Figure 1.60). Each section contains five sintered compacts of B<sub>4</sub>C and C as neutron absorbers in the annular space. The CRs were names center (C), ring 1 (R1), ring 2 (R2), and ring 3 (R3) from the center to the outside as shown in Figure 1.43. The CR positions at critical conditions were measured to evaluate the calculation accuracy. They were inserted from top to bottom. The bottoms of the inserted CRs were kept almost at the same level at critical conditions. The CR positions were defined with distance from the bottom of the fifth fuel layer. The R3 CRs were fully withdrawn at critical conditions ([Ref. 1 p. 317](#)).

To reduce thermal stresses in the sleeves, the neutron absorber sections and spine are separated from each other and each section is sustained by a support ring. The absorber section is assembled from parts with gaps between them, and all parts are mechanically connected without welds. Each absorber section contains a guide ring to restrain them from rolling too much. The lower end of the spine is attached to a tubular shock absorber. At the bottom of the core are graphite dishes, also called shock absorbers, which absorb the impact of a control rod. The annular space of each absorber section contains the sintered compacts with a density of 1.9 g/cm<sup>3</sup> ([Ref. 7, p. 90-91](#)).

The total height of the control rods is 3094 mm. The pellets are surrounded by a 3.5 mm thick sleeve of Alloy 800H. Inner and outer diameters of the CRs are 65 and 113 mm, respectively. The height of the pellet stack in a single absorber section is 29 cm ([Ref. 2, pp. 22 and 131](#)).

Table 1.15 lists the specifications of the CR and RSS systems.

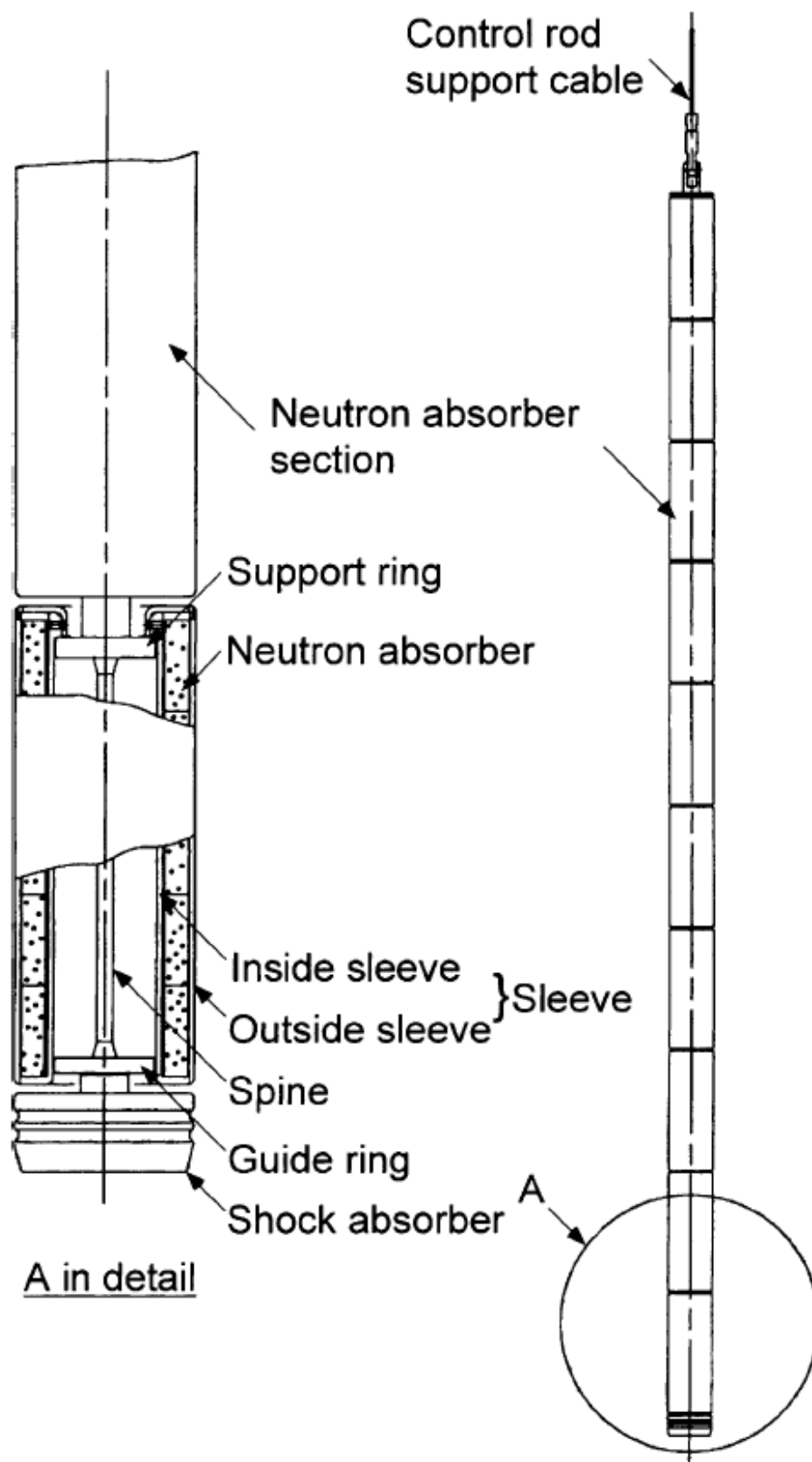


Figure 1.60. Control Rod of HTTR (Ref. 7, p. 92).

Table 1.15. Specifications of the CR System and the RSS (Ref. 4, p. 25 and Ref. 7, p. 91).

CR System	
Type	Double Cylinders with Lid and Vent
Number	16 Pairs (32 Rods) 15 Pairs when Irradiation Test is in Center Column
Total Length (m)	3.1
Outer Diameter (mm)	113
Inner Diameter (mm)	65
Sleeve	
Thickness (mm)	3.5 <sup>(a)</sup>
Material	Alloy 800H
Neutron Absorber	
Density (g/cm <sup>3</sup> )	1.9
Outer Diameter (mm)	105 <sup>(b)</sup>
Inner Diameter (mm)	75
Material	Sintered Compact of B <sub>4</sub> C/C
Natural Boron Concentration (wt.%)	30
Spine	
Diameter (mm)	10
Material	Alloy 800H
RSS	
Drive Mechanism	Drop B <sub>4</sub> C/C Pellets by Gravity
Number	16
Effective Length (m)	3.1
Hole	
Packing Fraction (vol. %)	55
B <sub>4</sub> C/C Graphite Pellets	
Density (g/cm <sup>3</sup> )	1.9
Diameter (mm)	13
Length (mm)	13
Material	Sintered Compact of B <sub>4</sub> C/C
Natural Boron Concentration (wt.%)	30

- (a) Mathematically, the clad thickness does not match up with the other dimensions provided for the control rod. It is unclear whether gaps exist between absorber and clad material or the cladding is thicker.
- (b) Reference 4 lists this value as 115, which cannot fit within the designated cladding dimension. All other values reported in Reference 4 match values in Reference 7.

The RSS absorbers are a system of pellets produced by sintering  $B_4C$  powder and graphite. These pellets can be released from the storage hoppers into holes in the control rod guide columns, to act as a backup system for the control rod system (Ref. 4, p. 31). In an accident when the control rods cannot be inserted, an electric plug is pulled out by a drive motor, and the neutron-absorbing pellets falls into the core by gravity. A schematic of the RSS is shown in Figure 1.61 and specifications are listed in Table 1.15 (Ref. 7, p. 99).

A cross-section depiction of the head of the reactor vessel with both CR and RSS configurations is shown in Figure 1.62.

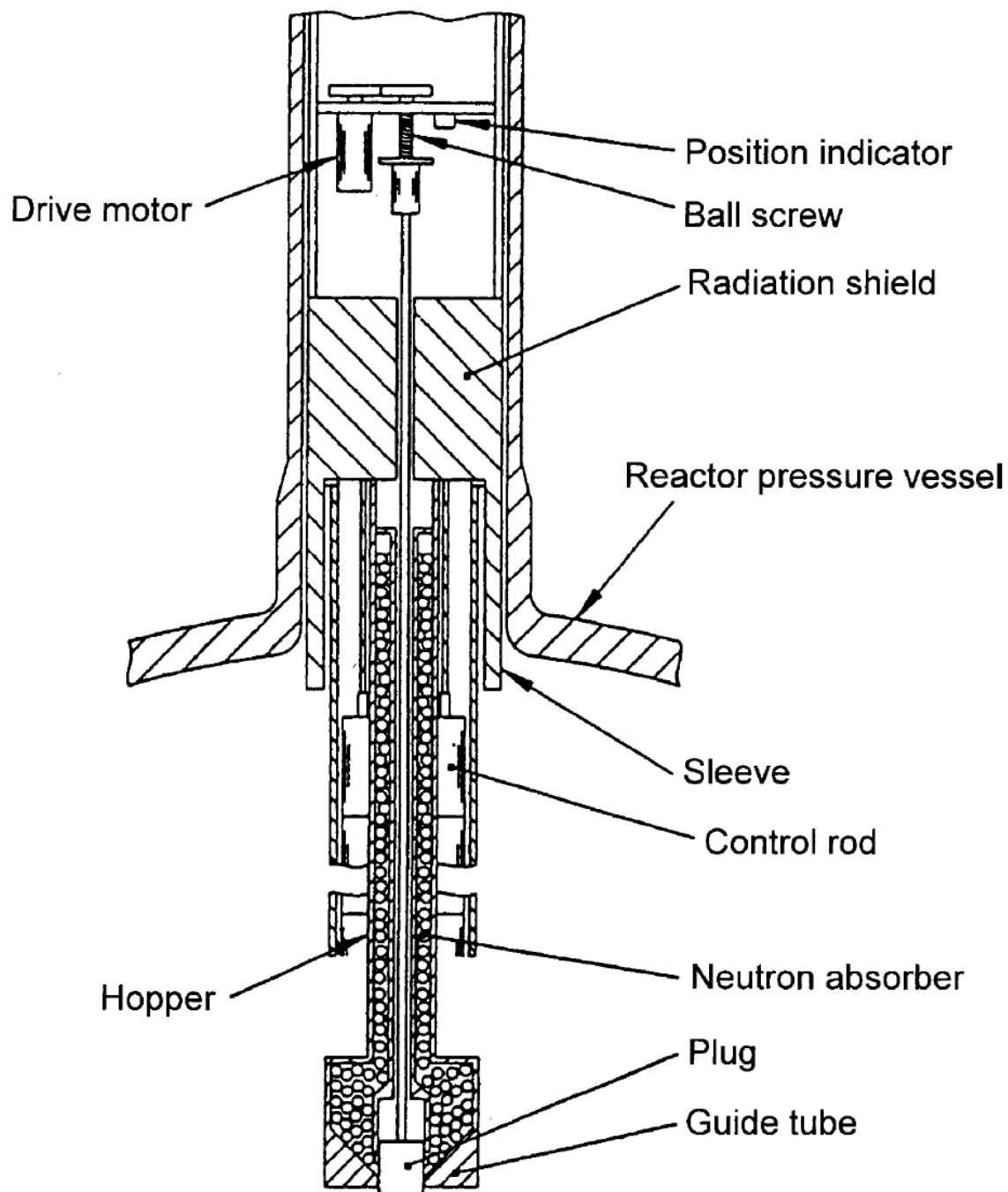
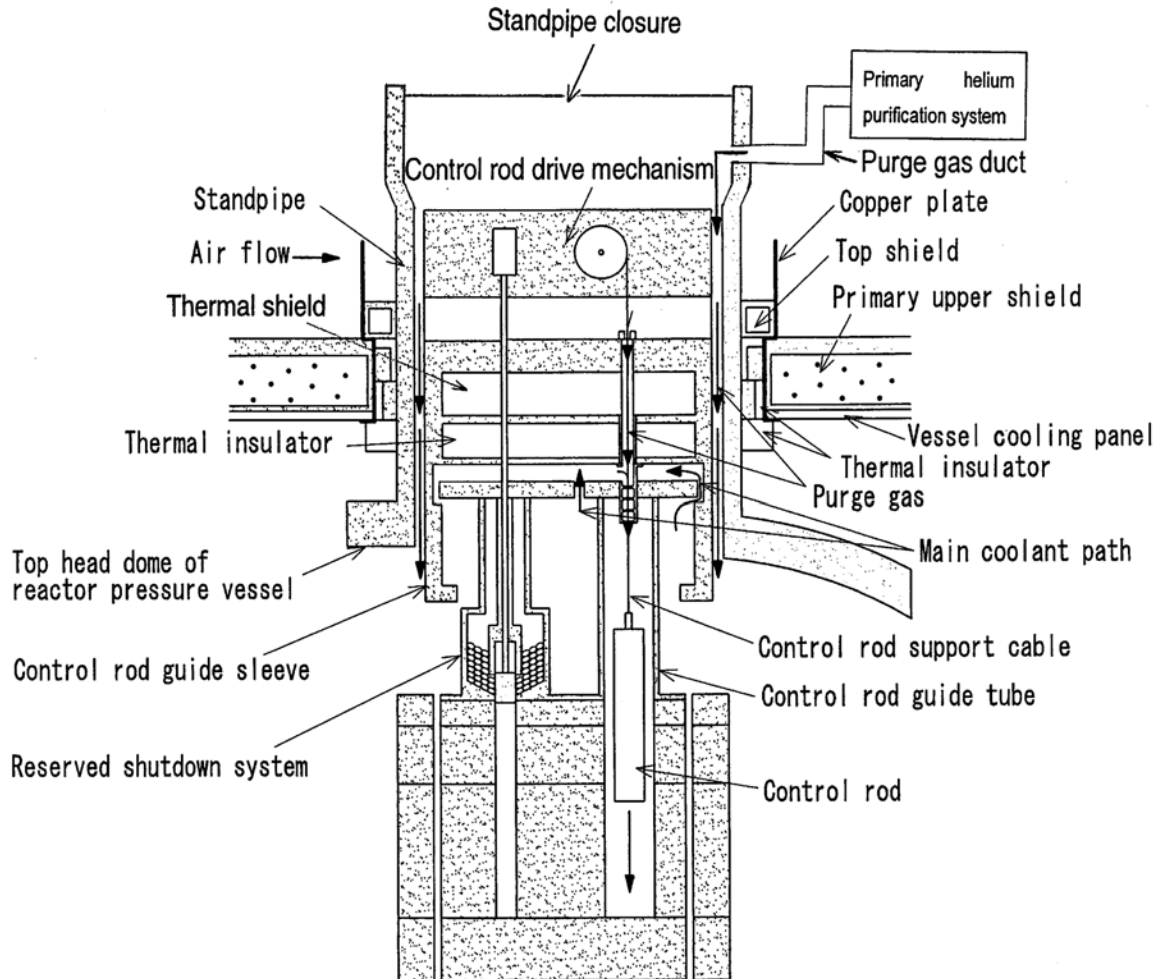


Figure 1.61. Reserved Shutdown System (Ref. 7, p. 99).



Figure 1.62. Vertical Cross-Section of Inside of Control Rod Standpipe.<sup>a</sup>

<sup>a</sup> T. Takeda and Y. Tachibana, "Indirect Air Cooling Techniques for Control Rod Drives in the High Temperature Engineering Test Reactor," *Nucl. Eng. Des.*, **223**: 25-40 (2003).

Control rod guide blocks have the same envelope dimensions as the fuel blocks and use the same dowel/socket connections as shown in Figures 1.63 and 1.64. The control rod and RSS insertion holes have a diameter of 123 mm. The bottom block of the CR guide column contains B<sub>4</sub>C/C pins for thermal neutron shield ([Ref. 5, p. 76](#)). Specifications for CR blocks are found in Table 1.16.

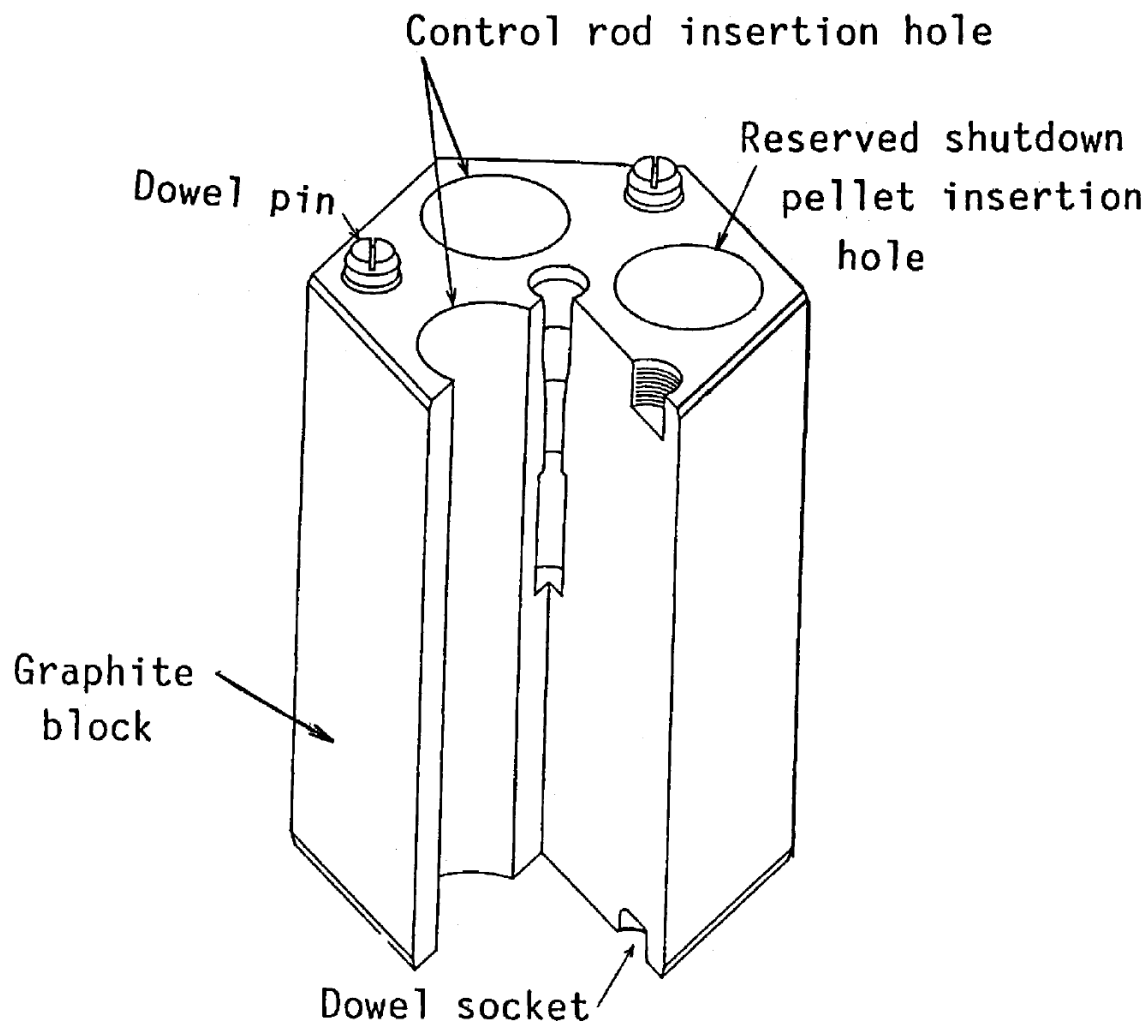
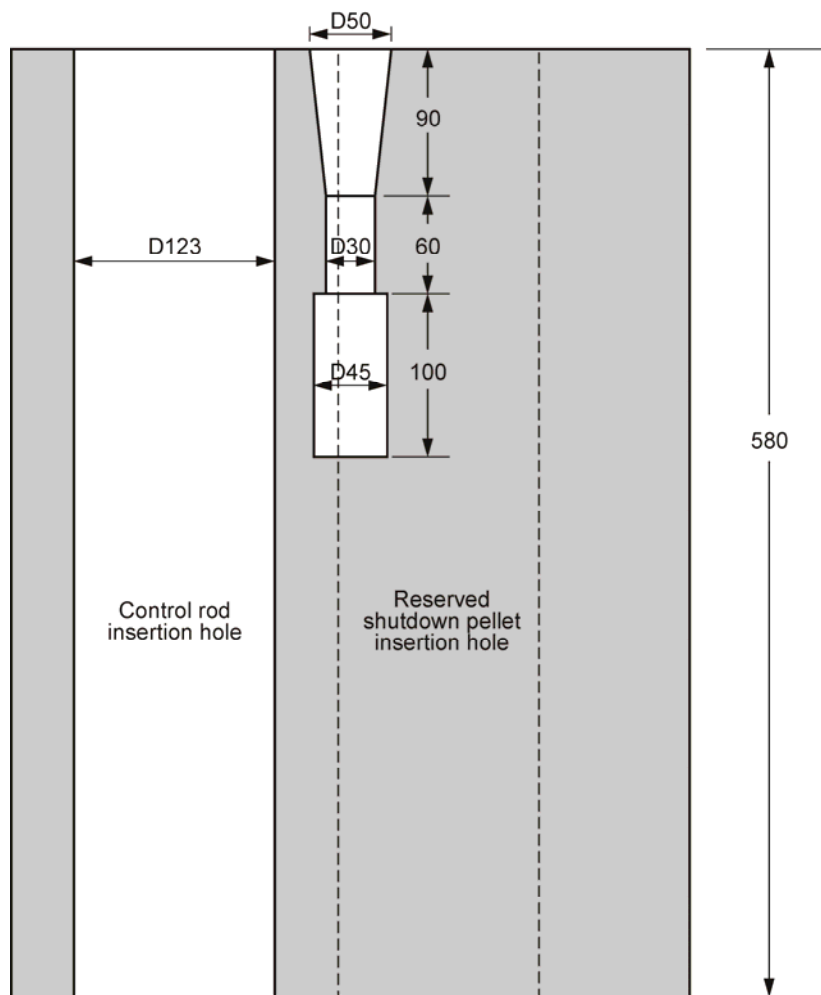
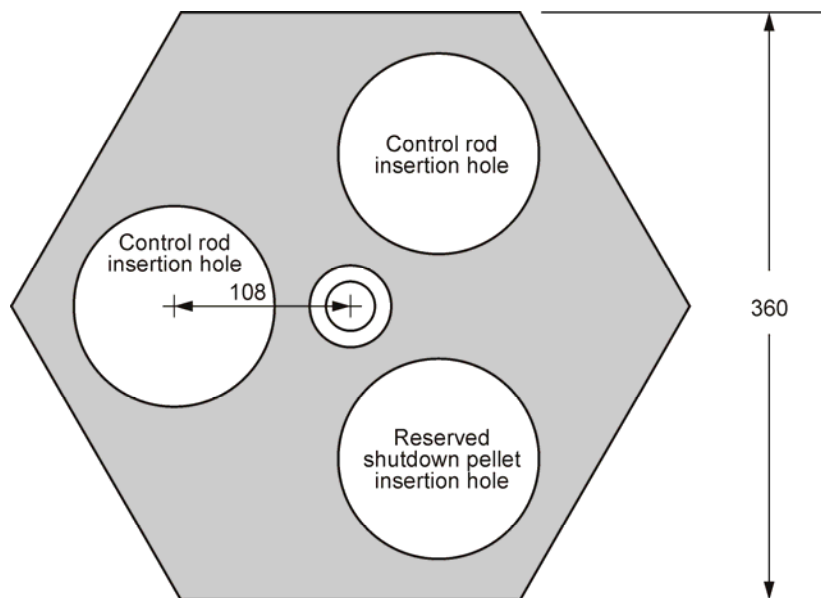


Figure 1.63. Control Rod Guide Block ([Ref. 5, p. 76](#)).



Dimensions in mm

08-GA50017-151-4

Figure 1.64. Control Rod Guide Block (Ref. 2, p. 19).

Dxx represents the diameter in xx (mm).

Table 1.16. Specifications of Reflector Blocks and CR Guide Blocks (Ref. 4, p. 25).<sup>(a)</sup>

Replaceable Reflector Blocks	
Material	IG-110 graphite
Width across Flats (cm)	36
Height (cm)	58
Density (g/cm <sup>3</sup> )	1.75
Impurity (ppm)	<1 (boron equivalent)
Control Rod Guide Blocks	
Material	IG-110 graphite
Width across Flats (cm)	36
Height (cm)	58
Density (g/cm <sup>3</sup> )	1.75
Impurity (ppm)	<1 (boron equivalent)
Hole	
Diameter (cm)	12.3
Number	3
Permanent Reflector	
Material	PGX graphite
Density (g/cm <sup>3</sup> )	1.73
Impurity (ppm)	<5 (boron equivalent)

- (a) K. Yamashita, R. Shindo, I. Murata, S. Maruyama, N. Fujimoto, and T. Takeda, "Nuclear Design of the High-Temperature Engineering Test Reactor (HTTR)," *Nucl. Sci. Eng.*, **122**: 212-228 (1996).

The position indicator of the control rod drive mechanism has an accuracy of  $\pm 5$  mm.<sup>a</sup>

The top of the control rod guide columns and irradiation columns are 100 mm lower than that of the fuel columns because the 9<sup>th</sup> layer blocks of these columns are 480 mm in height. Each horizontal gap width between two columns is 2 mm on average (Ref. 2, p. 13).

When the control rods are fully inserted, the lower ends of all control rods are on the same plane with the bottom face of the 7<sup>th</sup> layer of the fuel column. When the control rods are fully withdrawn, their upper limit is the upper face of the 1<sup>st</sup> replaceable reflector block over the fuel region with exception of the R2 CRs, which have upper limits at 725 mm below the top of the 1<sup>st</sup> block. Axial control rod positions are shown in Figure 1.65 (Ref. 2, p. 23 and 24).

The fully withdrawn positions of R2 were different from those of C, R1, and R3; they were limited because it was used for the scrams at full-power operations. The C and R1 CRs would then be inserted into the core only after core temperatures decreased, so as to maintain shutdown conditions (Ref. 1, p. 317).

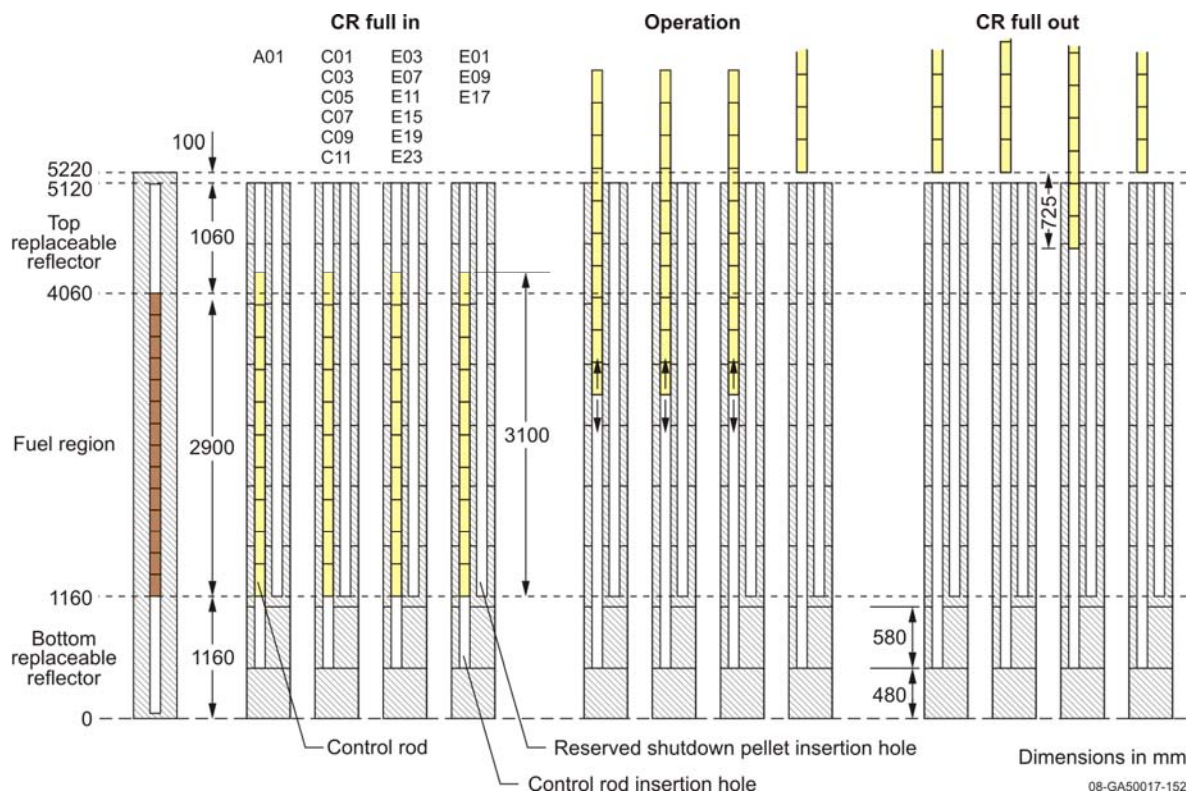


Figure 1.65. Axial Control Rod Positions.

<sup>a</sup> T. Takeda and Y. Tachibana, "Indirect Air Cooling Techniques for Control Rod Drives in the High Temperature Engineering Test Reactor," *Nucl. Eng. Des.*, **223**: 25-40 (2003).

### Instrumentation

Temporary neutron instrumentation was used for the start-up physics tests: three  $\text{BF}_3$  counters, three micro-fission chambers (FCs), and two gamma-ray compensated ionization chambers (CICs). They were located in three irradiation test columns, as shown in Figure 1.66. The inverse multiplication factor was evaluated from the increase in neutron detection in the  $\text{BF}_3$  counters. The FC was traversed in the irradiation test column for measurements of axial neutron flux distributions; they were 5 cm long and 0.6 cm in diameter and connected to the end of a long aluminum stick. The CIC was used for reactivity measurement using the inverse kinetic (IK) method. A temporary neutron source ( $\text{Am-Be}$ ,  $1.48 \times 10^4$  Bq) was inserted into one irradiation test column. The vertical position of equipment placed in each irradiation column is shown in Figure 1.67. An aluminum tube well was inserted into the holes of the irradiation test columns. The  $\text{BF}_3$  counter in channel 1 was installed  $\sim 2$  m apart from the neutron source to avoid effect of direct-flying neutrons from the source. The temporary neutron source was withdrawn from the core after establishment of critical conditions, to exclude the neutron source effect on reactivity measurements (Ref. 1, pp. 313 & 317).

The  $\text{BF}_3$  counters and FCs in the same channel were located at the same vertical position for redundancy during the initial 1/M monitoring (Ref. 2, p. 25).

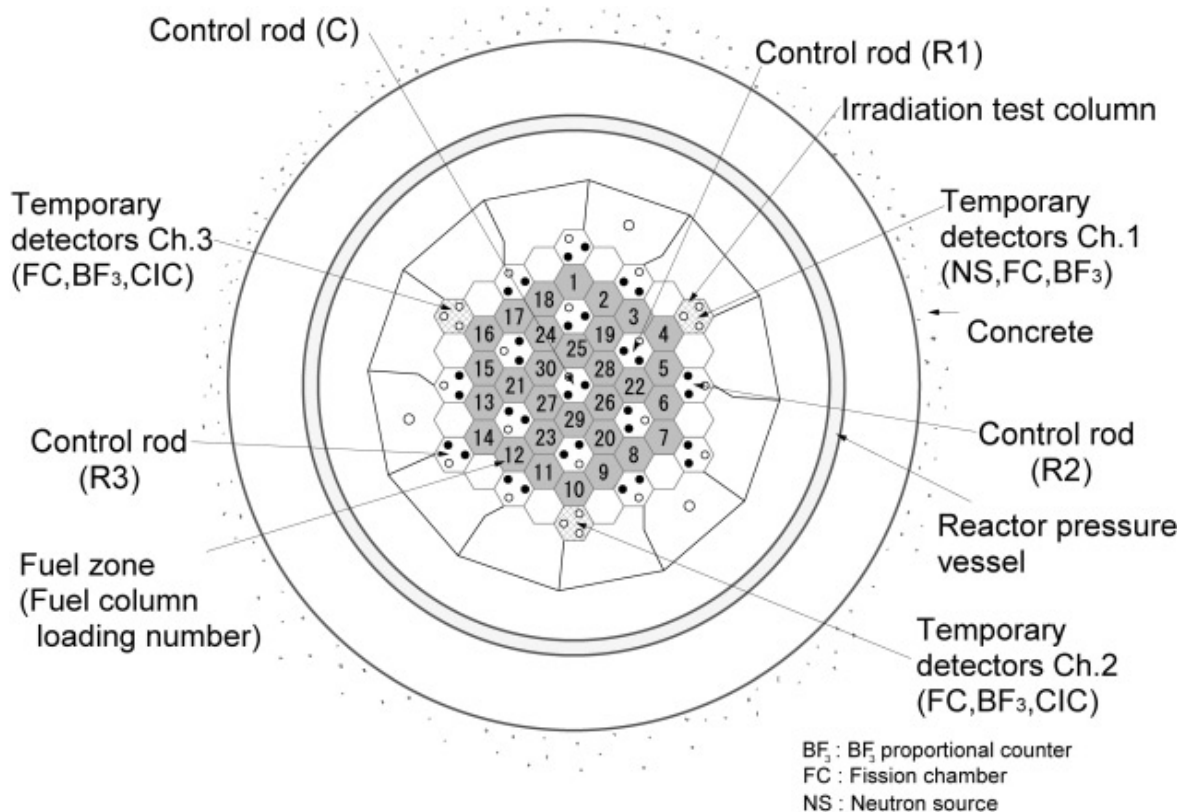


Figure 1.66. Fuel-Loading Scheme and Temporary Neutron Instrumentation (Ref. 1, p. 313).



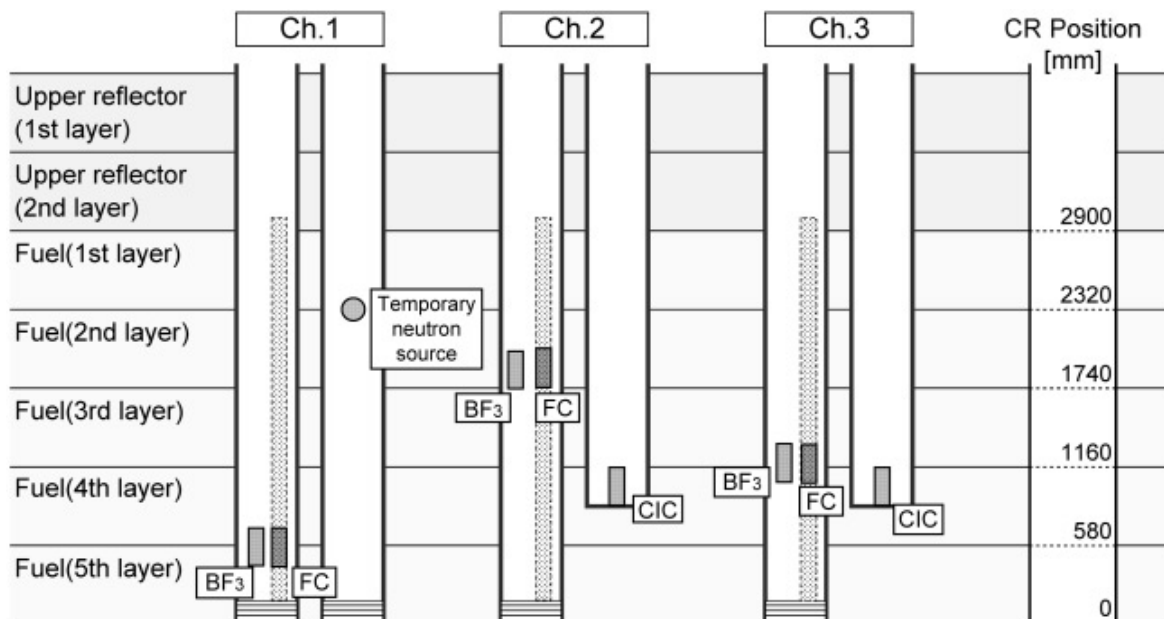


Figure 1.67. Vertical Position of the Temporary Neutron Detectors (Ref 1, p. 314).

### Graphite Blocks

The five fuel assembly blocks within a fuel column have two top reflector blocks and two bottom reflector blocks (Ref. 2, p. 12).

The top and bottom replaceable reflector blocks have the same basic configuration as the fuel blocks but do not contain fuel rods. These reflector blocks, above and below the active core, have the same arrangement of coolant channels as the fuel blocks within the same columns. The bottom replaceable reflector block below each fuel block column provides a transition of the many coolant channels to a single large channel which mates with the coolant channels within the hot plenum blocks. The alignment of the fuel, control rod guide, and replaceable reflector columns with the hot plenum blocks is maintained by dowel/socket connections in the bottom reflector blocks, which fit into the hot plenum blocks. The bottom layer of reflector blocks contains B<sub>4</sub>C/C pins for the thermal neutron shield (Ref. 5, p. 76).

The arrangement of coolant channels (with 23-mm inner diameter) in the top replaceable reflector blocks (Figure 1.68) corresponds to that of coolant channels (with 41-mm inner diameter) in the fuel assemblies within the same column. The upper bottom replaceable reflector block (8<sup>th</sup> layer) has the same arrangement of coolant channels as the top replaceable reflector block. The lower bottom reflector block (9<sup>th</sup> layer) has six large coolant channels (Ref. 2, p. 21).

All blocks except the bottom block of the control rod guide column and the irradiation column are 58 cm in height and 36 cm in width across flats. Boron pins are installed into the lower parts of blocks in the 9<sup>th</sup> layer for neutron shielding (Ref. 2, p. 23).

Consistent design of the dowel and socket system has a gap of  $\pm 0.3$  mm between dowel and socket. Diameters of dowel are 45 mm and 55 mm for the fuel and control rod blocks, respectively. The socket diameters are 45.6 and 55.6 mm, respectively.<sup>a</sup>

<sup>a</sup> N. Takikawa, M. Ishihara, T. Iyoku, S. Shiozawa, M. Tokumitsu, S. Koe, and M. Uno, "Assessment of the Load Capacity of the Dowel and Socket System in the HTTR Hexagonal Block," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 105-112 (February 1993).

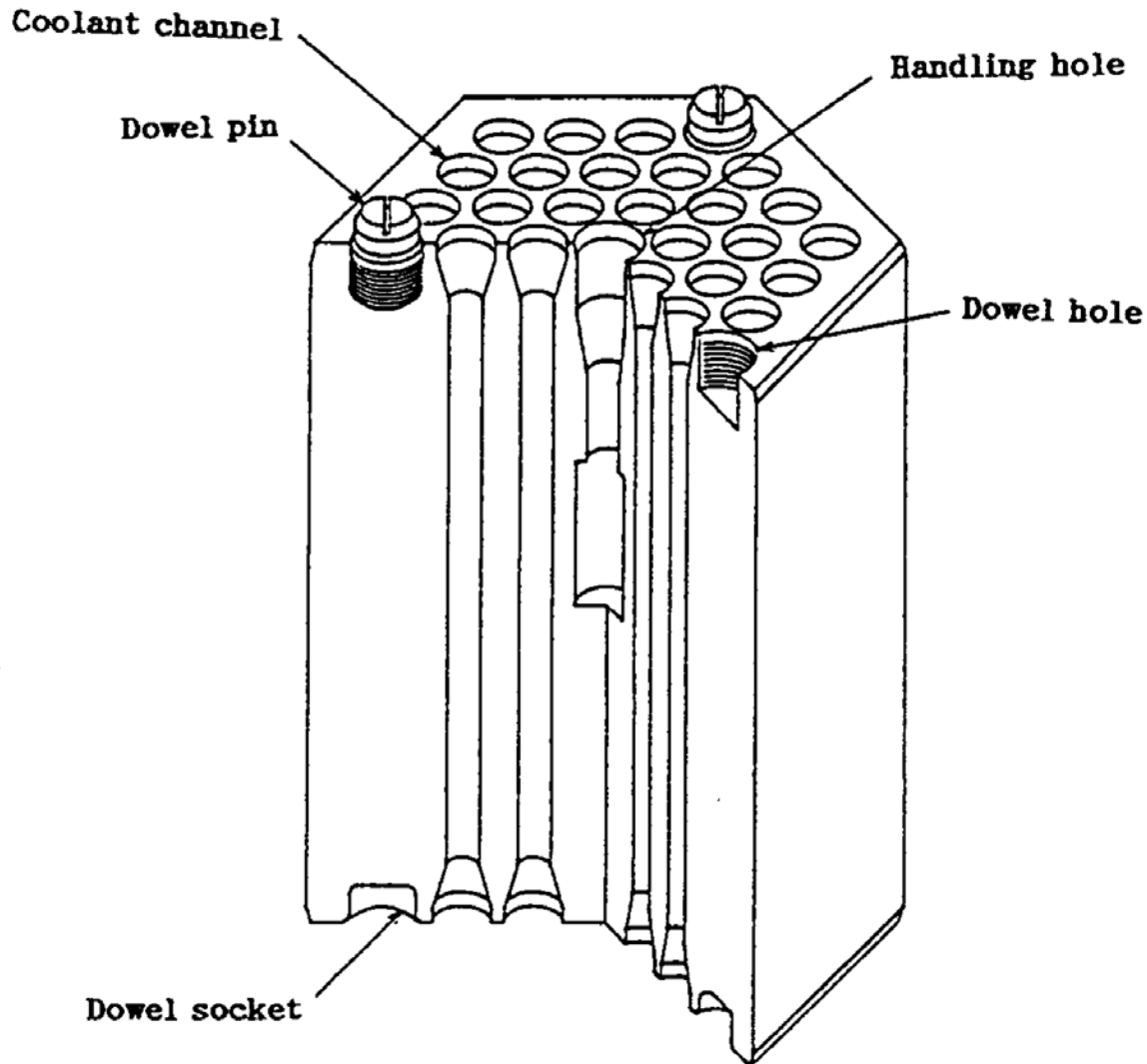


Figure 1.68. Axial Replaceable Reflector Block Used in Fuel Column Layers 1, 2, and 8 ([Ref 15, p. 55](#)).

### **Dummy Blocks**

Before fuel loading, the whole fuel region in the core was filled with graphite dummy blocks. There are two types of dummy blocks: graphite blocks with three large holes, and those with three small holes. The graphite blocks with three large holes were used as control rod guide columns for out-pile seismic tests of the core structure. The others with three small holes were newly produced as dummy fuel blocks ([Ref. 2, p. 14](#)).

The external form of the dummy fuel clock is the same as that of a fuel graphite block ([Ref. 2, p. 23](#)).

### **Replaceable Reflectors**

Twelve replaceable reflector columns are adjacent to the active core region and surrounded by permanent reflector blocks ([Ref. 2, p. 12](#)).

The side replaceable reflector blocks adjacent to the core have the same envelope dimensions as fuel blocks but are solid graphite and contain only a central handling hole for removal and insertion ([Ref. 5, p. 76](#)).

### **Permanent Reflectors**

A permanent graphite reflector region surrounds the core and holds it together (Figures 1.69 and 1.70). The 12 permanent reflector blocks are in the form of a large polygonal graphite block 4250 mm across flats. The permanent reflector blocks have holes for irradiation tests and neutron detectors, giving a void fraction of approximately 0.7 % (Ref. 2, p. 23).

The permanent reflector is an assembly of 12 circumferential segments in 8 axial layers (Ref. 6, p. 83).

The total height of the permanent reflectors is about 8 m. Each circumferential segment is keyed on the adjacent surface and bound to the core restraint mechanism. By-pass flow of the coolant is restricted between adjacent blocks by face contact and seal elements (see Figure 1.7).<sup>a</sup>

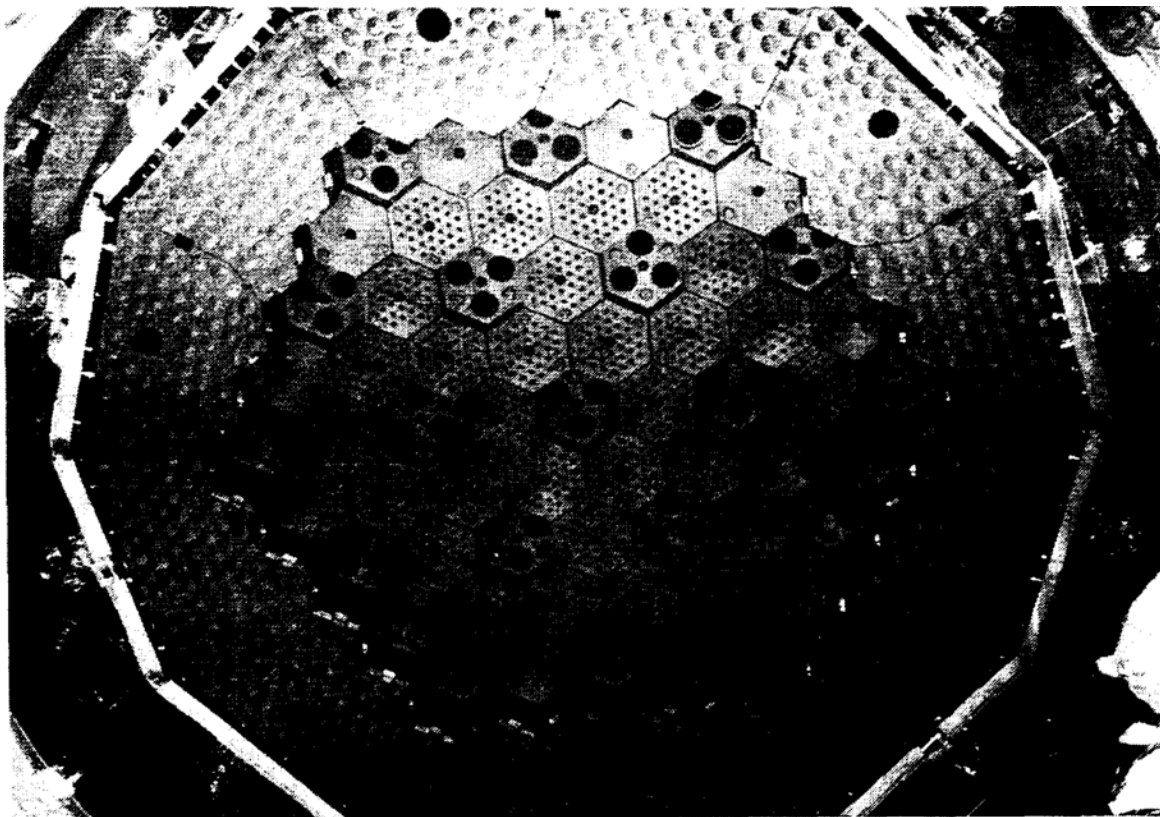


Figure 1.69. Installed Core Arrangement of Top Replaceable Reflector Region.<sup>b</sup>

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<sup>a</sup> S. Maruyama, A. Saikusa, S. Shiozawa, N. Tsuji, and T. Miki, "Results of Assembly Test of HTTR Reactor Internals," IAEA-TECDOC-901, International Atomic Energy Agency, Vienna, pp. 105-116 (August 1996).

<sup>b</sup> T. Tanaka, O. Baba, S. Shiozawa, M. Okubo, and K. Kunitomi, "Construction of the HTTR and Its Testing Program for Advanced HTGR Development," IAEA-TECDOC-899, International Atomic Energy Agency, Vienna, pp. 15-23 (August 1996).



Figure 1.70. Inside of the Reactor Pressure Vessel.<sup>a</sup>

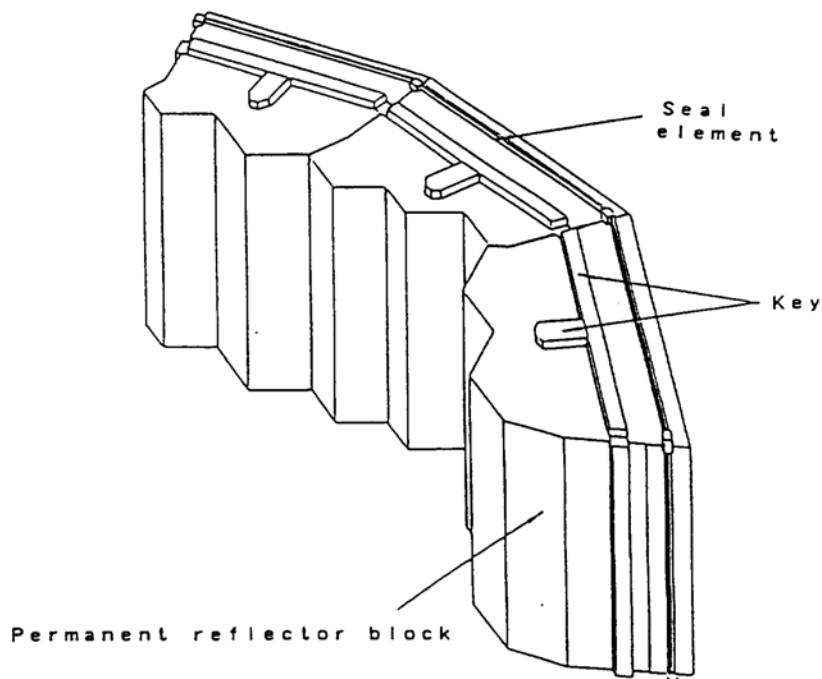


Figure 1.71. Schematic Diagram of the Permanent Reflector Blocks.<sup>a</sup>

<sup>a</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

## **Helium Coolant**

The primary coolant system is filled with helium at normal atmospheric pressure and was not in operation during fuel loading (Ref. 2, p. 14). Impurity concentrations during initial physics tests were negligible

At power operation, the pressure in the primary circuit is about 4 MPa, and the impurity concentration is that shown in Table 1.10.

Helium gas enters the core after reversing at the upper plenum in the reactor vessel. The gas is heated as it flows downward through the fuel elements in each core region and leaves the reactor before passing through an inner tube of the outlet gas duct after mixing in the hot plenum.<sup>b</sup>

### *1.1.2.4 Description of Criticality Measurements*

Before fuel loading, the whole region in the core was filled with graphite dummy blocks. Helium gas was filled up to 1 atm at room temperature. Dummy blocks were replaced with fuel blocks, column by column, in the fuel loading, starting from the core periphery to the center, as shown in Figure 1.43.

In the first phase of fuel loading, six fuel columns were loaded consecutively. In the second phase, three fuel columns were loaded at a time, until a total of 15 columns was achieved. In the third phase of fuel loading, columns were loaded one at a time, so as to identify the number of fuel columns needed for the first criticality. In every loading phase, all control rods were withdrawn from the core after each fuel loading, and inverse multiplication factors were measured. Initial criticality was achieved with 19 fuel columns loaded. Symmetric annular cores were formed at 21-, 24-, and 27-fuel-column loaded cores. The fully-loaded core consists of 30 fuel columns. Figure 1.72 shows some of the fuel-loading steps during the initial core loading (Ref. 13, p. 285).

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<sup>a</sup> S. Maruyama, A. Saikusa, S. Shiozawa, N. Tsuji, and T. Miki, "Results of Assembly Test of HTTR Reactor Internals," IAEA-TECDOC-901, International Atomic Energy Agency, Vienna, pp. 105-116 (August 1996).

<sup>b</sup> Y. Inagaki, R. Hino, K. Kunitomi, K. Takase, I. Ioka, and S. Maruyama, "R&D on Thermal Hydraulics of Core and Core-Bottom Structure," *Nucl. Eng. Des.*, **233**: 173-183.

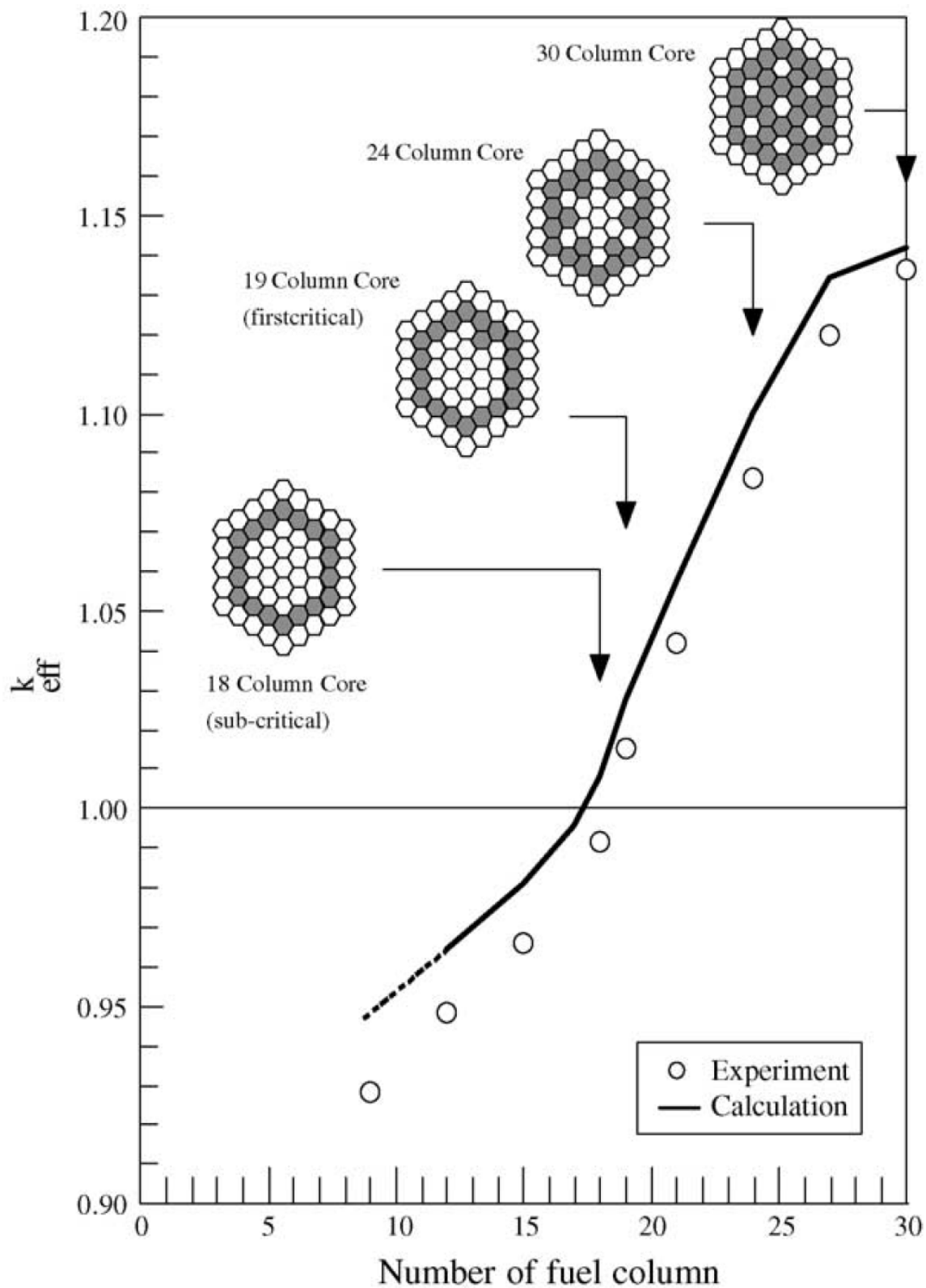


Figure 1.72. Change in  $k_{eff}$  Value on Critical Approach at Room Temperature using the Fuel Addition Method (Ref. 13, p. 285).



The inverse neutron multiplication factor was monitored in order to observe the approach to criticality; the core was regarded as critical when the neutron density was self-sustained after removing the temporary neutron source. After the initial criticality, reactivity increase was measured using the IK method. The excess reactivity of the core was then obtained by adding all increments of the reactivity from the first criticality to the fully-loaded core (Ref. 2, p. 14).

Inverse multiplication factors were evaluated at 0, 6, 9, 12, 15, 16, 17, and 18 fuel-column-loaded cores to predict the first criticality (Ref. 2, p. 39). The inverse multiplication data were recorded and are shown in Figure 1.73.

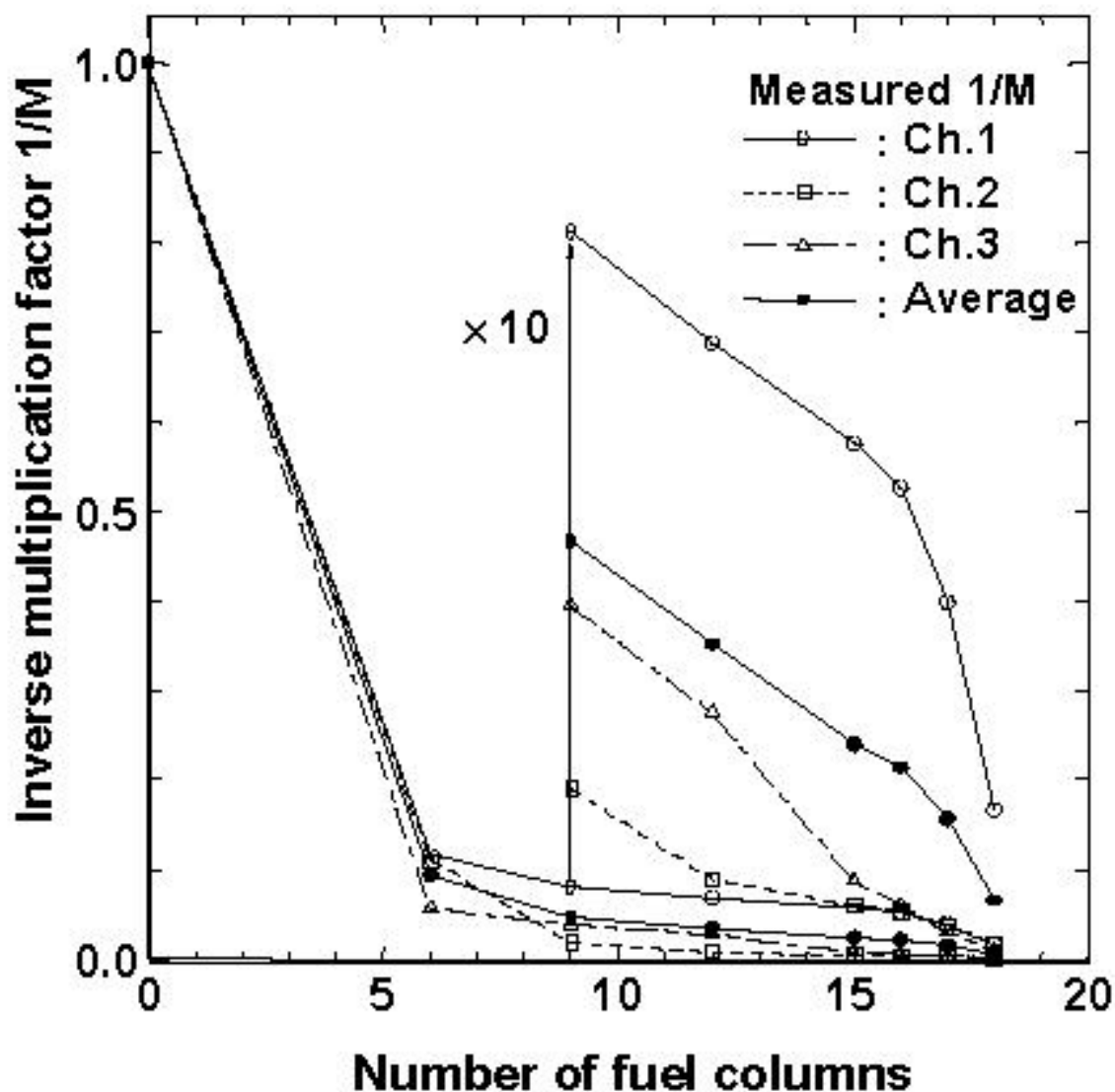


Figure 1.73. Inverse Multiplication Data for the Initial Approach to Critical of the HTTR.

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE

Measurements were carried out by using the signal processing system DSA-2 connected to the CICs, in real time. During the online reactivity experiments, the measured DC signals were digitized for use in the IK method. During the first approach to critical (after loading the 19<sup>th</sup> fuel assembly) the source criticality at very low power was achieved. Then the neutron source was removed and the central control rod moved to compensate for the change in reactivity. The first criticality of the reactor was attained on November 10, 1998. Figure 1.74 shows the result of the reactivity measurements during this approach to critical ([Ref. 2, p. 113](#)).

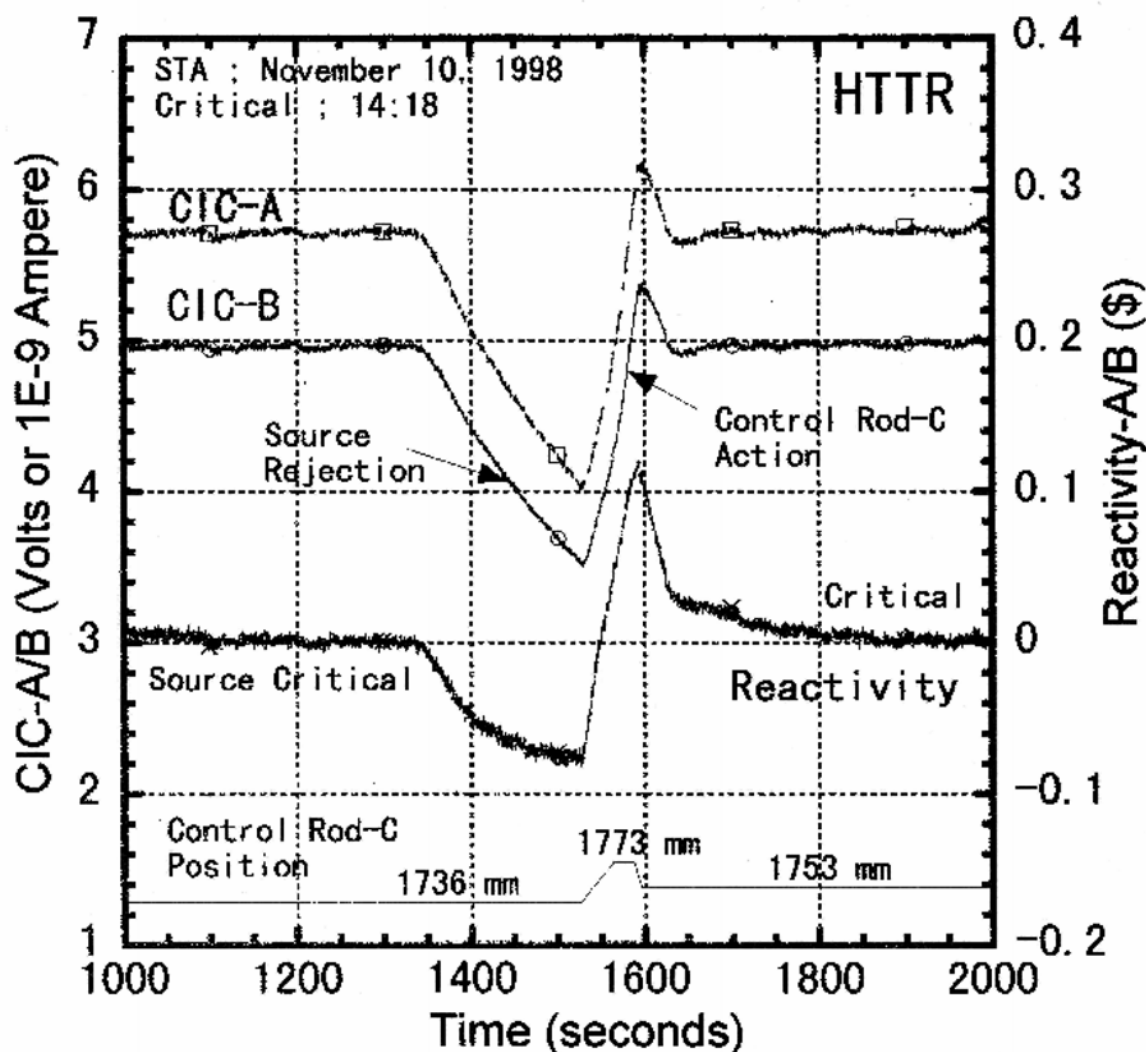


Figure 1.74. Initial Approach to Criticality ([Ref 2, p. 114](#)).

The CR positions are provided in Table 1.17 for the critical conditions of the annular and fully-loaded cores. The C-CR was inserted into the 19-column core to adjust initial criticality while the other CRs were fully withdrawn ([Ref. 1, p. 317](#)).

All control rod insertion levels are adjusted on the same level except for the three pairs of control rods in the most outer region in the side reflectors. These three pairs of CRs are usually fully withdrawn ([Ref. 2, p. 40](#)).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

There were two CR patterns used for the 24-fuel-column core. The first was the flat standard (FS) pattern, where the C, R1, and R2 CRs were inserted into the core at the same height, and the R3 CRs were fully withdrawn. The second pattern was the F23 pattern, where the R2 and R3 CRs in the side reflector were inserted into the core at the same height and the C and R1 CRs were fully withdrawn. The F23 pattern was used to simulate the control of future HTGRs, where the reflector CRs would mainly be used for reactivity control. The effective multiplication factors for the measured CR positions at critical conditions were evaluated for different annular cores and temperatures. The results in Table 1.17 are adjusted such that the multiplication factors could be evaluated at 27 °C, 300 K (Ref. 1, pp. 317-318).

Table 1.17. Measured Control Rod Positions for Critical Conditions (Ref. 1, p. 318).

Case	Fuel Columns	Temperature (°C)	$k_m^{(a)}$	Critical Rod Position (mm) <sup>(b)</sup>				Remark <sup>(c)</sup>
				C	R1	R2	R3	
1	19	23	1.00049	1739	4050	3325	4050	C
2	21	24	1.00037	2647	2645	2646	4049	FS
3	24	24	1.00037	2213	2215	2215	4049	FS
4	24	24	1.00037	4051	4050	1593	1592	F23
5	27	24	1.00037	1901	1899	1899	4050	FS
6	30	25	1.00025	1775	1775	1775	4049	FS

- (a) Measured  $k_m$  has been corrected with the measured temperature coefficient (TC) for the 30-fuel-column core [ $-1.23 \times 10^{-4}$  ( $\Delta k/k$ -°C) to 27 °C (300 K)]. It was assumed that use of the TC for the 30-fuel-column core was practically still useful for the other cores because of the small temperature difference of <4 °C.
- (b) Nominally fully withdrawn positions of C, R1, and R3 CRs are 4060 mm and that of the R2 CRs is 3335 mm.
- (c) C = criticality obtained using central control rod only.  
 FS = flat standard pattern where C, R1, and R2 CRs were inserted into the core at the same levels while R3 CRs were fully withdrawn.  
 F23 = only R2 and R3 CRs were used for control while C and R1 CRs were fully withdrawn.

A simpler statement for control rod positions of four of the critical cases is shown in Table 1.18, representing the average for rods C, R1, and R2 (with R3 completely withdrawn).

Table 1.18. Average Control Rod Positions for Critical Conditions (Ref. 2, p. 40).

Fuel Column	21 (Case 2)	24 (Case 3)	27 (Case 5)	30 (Case 6)
Rod Position (mm)	2646 ± 5	2215 ± 5	1899 ± 5	1775 ± 5

Effective corrections to critical rod position in the core include a sinking effect from the CR driving mechanism of about -14 mm and a temperature expansion effect of about +2 mm from 25 to 27 °C (Ref. 2, p. 46). Reported Japanese data typically had the sinking effect already accounted for in the reported rod heights.

### 1.1.2.5 Description of Subcriticality Measurements

Additional information for the fully-loaded core reports a measured subcritical effective multiplication factor of  $0.685 \pm 0.010$  when the control rods are completely inserted (Ref. 2, p. 111).

### 1.1.2.6 Description of Warm Criticality Measurements

Control-rod positions for criticality were also evaluated at 480 K during the evaluation of the core temperature coefficient. Control rod groups C, R1, and R2 were at the same level (1825 mm) and group R3 was fully withdrawn (Ref. 2, pp. 46, 48, and 331).

## 1.1.3 Material Data

### 1.1.3.1 Facility Description

#### **Site and Facility**

Additional information is not available.

#### **Containment Structure**

The containment vessel is made of carbon steel (Ref. 10, pp. 135-137).

#### **Reactor Pressure Vessel**

The RPV is made of 2-1/4Cr-1Mo steel, normalized and tempered. There are three kinds of 2-1/4Cr-1Mo steel: forgings of Japan Industrial Standard (JIS) specification SFVAF22B, equivalent to ASTM A-336 Gr F22, Cl 3; plates of JIS SCM4-2, equivalent to ASTM A-387 Gr 22, Cl2; and seamless pipes of JIS STPA24, equivalent to ASTM A-335 Gr P22. Material of components and weld lines are shown in Figure 1.75. The thickness of the RPV is determined so that general primary membrane stress is less than the allowable value for the design pressure of 4.7 MPa (Ref. 8, p. 106).

The fabrication procedure for the RPV is shown in Figure 1.76. There are two types of standpipes: 'a', a forging; and 'b', a forging and pipe welded together. The inner nineteen standpipes are for control rods and irradiation tests, which require dimensional accuracy in manufacturing, and are type 'a' standpipes. The heads on the outer 12 standpipes on the top head dome and the three standpipes on the shell are forgings that are welded to seamless pipes (Ref. 8, pp 107-108).

The top head dome is a very large forging, and the 19 nozzles for the inner standpipes are integrated into it (i.e., there are no weld lines between the dome and the nozzles). The other 13 nozzles, including a manhole nozzle, are welded to the top head dome. After welding the top head dome to the upper flange, stress relieving (SR) and final machining was performed. Then the standpipes were welded to the top head dome, followed by local stress relieving. The cylindrical shell of the RPV is made up of four plates formed and welded together. The shell was welded to the shell flange. The bottom head petal was welded to the bottom head dome, which consists of four formed plates, and the skirt. These were then welded together to build the RPV body for final stress relieving and machining. The welding of large components involved automatic narrow-gap (groove) MIG welding that was developed by Babcock-Hitachi, K.K. (Ref. 8, p. 108).

Additional RPV manufacturing details can be found elsewhere.<sup>a</sup>

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<sup>a</sup> Y. Tachibana, K. Kunitomi, S. Shiozawa, T. Asami, and S. Yamaguchi, "Material and Fabrication of the HTTR Reactor Pressure Vessel," IAEA-TECDOC-988, International Atomic Energy Agency, Vienna, pp. 305-318 (December 1997).

Irradiation embrittlement of the RPV is evaluated to be insignificant due to a lifetime fast fluence of less than  $1 \times 10^{17}$  n/cm<sup>2</sup> (E > 1 MeV). Temper embrittlement is a concern. To reduce embrittlement, the content of Si, Mn, P, Sn, and As must be reduced according to a J-factor and X-bar parameter:

$$J - factor = (Si + Mn)(P + Sn) \cdot 10^4 \leq 100$$

$$(Si, Mn, P, Sn : wt. \%)$$

and

$$X = \frac{10P + 5Sb + 4Sn + As}{100} \leq 10$$

$$(P, Sb, Sn, As : ppm)$$

Figure 1.77 shows the J-factor and X-bar of 19 heats of manufactures 2-1/4Cr-1Mo steels for the RPV. The J-factor and X-bar ranges from 25 to 55 and from 4.1 to 8.4, respectively, which satisfies the HTTR specification. There is almost a linear relationship between the J-factor and X-bar. The embrittlement factors for the weld metal of the RPV range from 60 to 84 and 6.0 to 9.0, respectively. The silicon and manganese content of weld metal is higher than that of the base metal (Ref. 8, pp. 108-109).

The thermal shield is equipped to protect the top head closure and the standpipes from high temperature exposure, and is made up of layers of metallic heat reflecting plates (Ref. 8, p. 106).

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

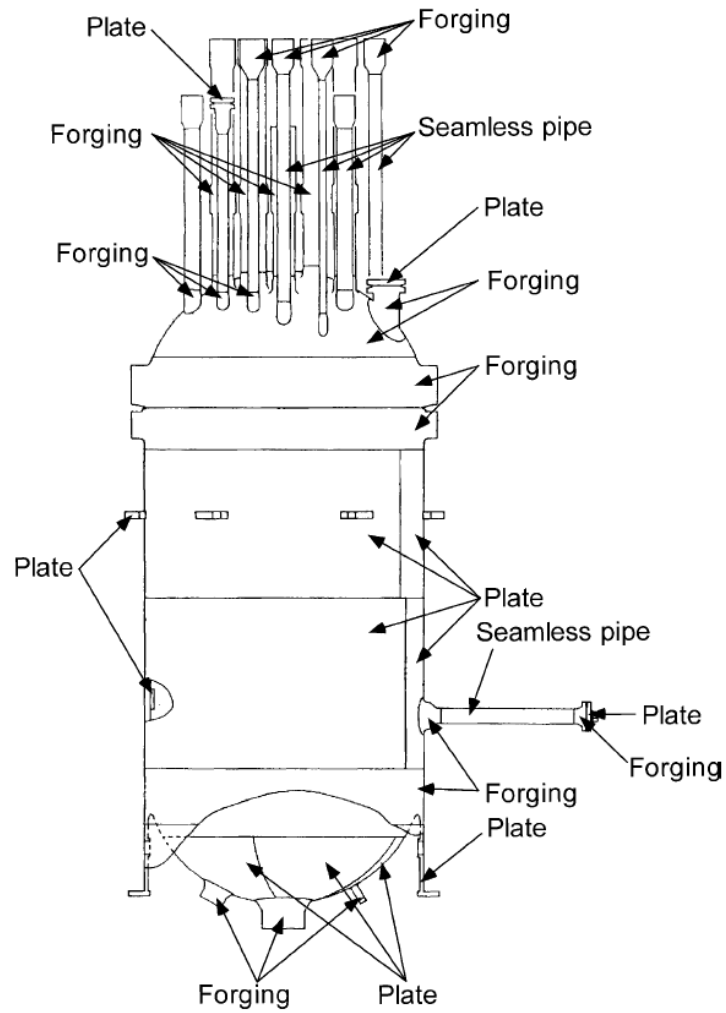


Figure 1.75. Material and Weld Lines of the RPV (Ref. 8, p. 106).



## Gas Cooled (Thermal) Reactor - GCR

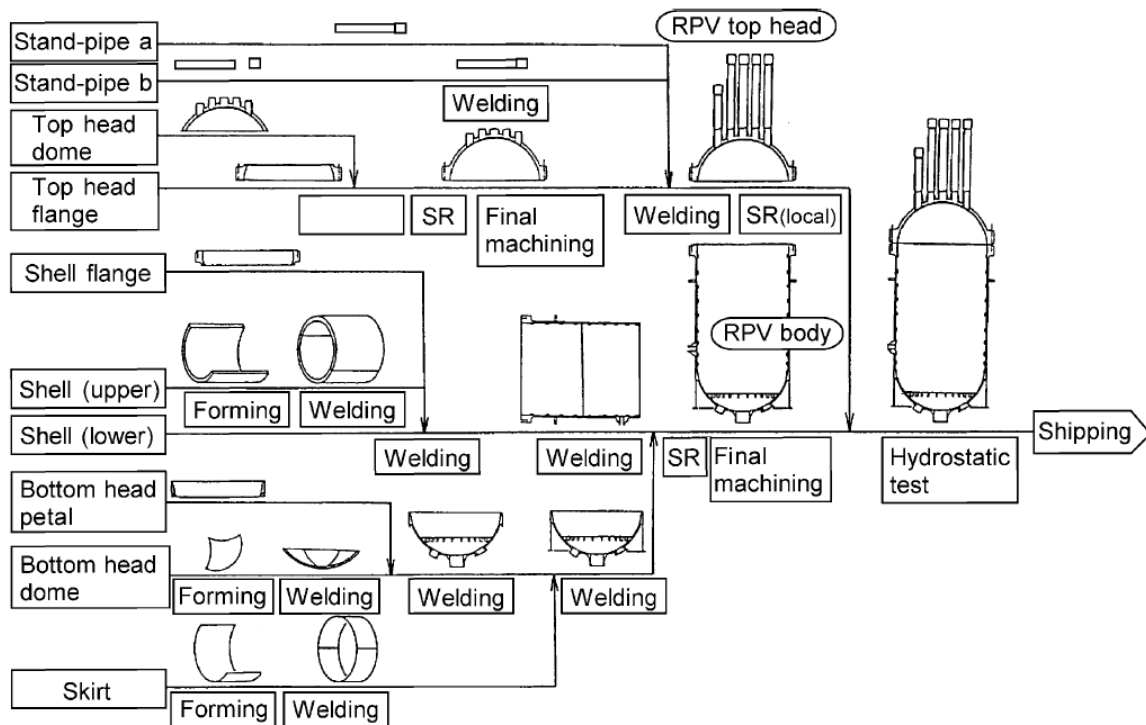
HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

Figure 1.76. Fabrication Procedure of the RPV (Ref. 8, p. 107).

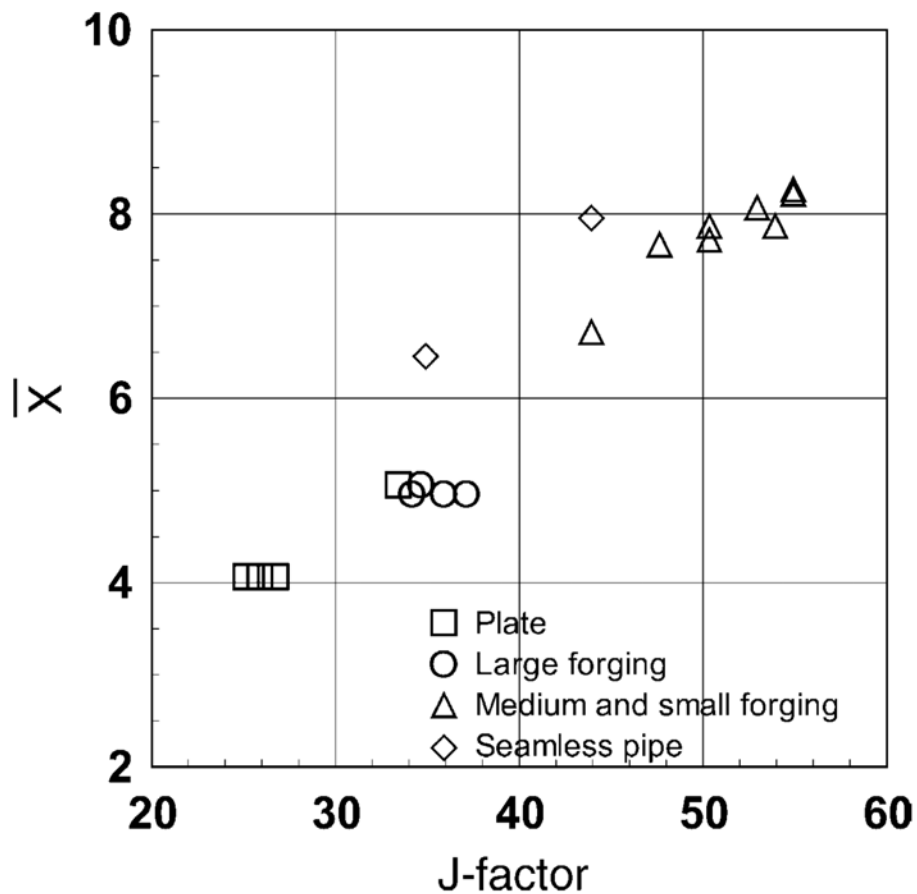


Figure 1.77. Distribution of X Parameter and J-factor on 2-1/4Cr-1Mo Steel for the RPV (Ref. 8, p. 109).

**Reactor Cooling System**

The material of the heat transfer tubes and the hot header of the IHX is Hastelloy XR, and the inner and outer shells are made of 2-1/4Cr-1Mo steel (Ref. 9, p. 117).

The inner and outer shells of the PPWC are made of 2-1/4Cr-1Mo steel. Austenitic stainless steel SUS321TB is used for the heat transfer tubes because of its strength and anti-corrosion characteristics at high temperatures. The material of the baffle plates is Hastelloy XR because they will be exposed to high temperature helium gas (Ref. 9, p. 115). The casing of the primary gas circulators is made of 2-1/4Cr-1Mo steel (Ref. 9, p. 118).

The outer and inner pipes of the primary concentric gas duct are made of 2-1/4Cr-1Mo steel, the liner material is Hastelloy XR, and the insulation material is a ceramic fiber composed of SiO<sub>2</sub> and Al<sub>2</sub>O<sub>3</sub> (Ref. 9, p.119).

For the secondary pressurized water cooler, 2-1/4Cr-1Mo steel of SCM4-2T is used for the inner and outer shells, SUS321TB is used for the heat transfer tubes, and Hastelloy XR is used for the baffle plates and liner. 2-1/4Cr-1Mo steel is used in the pipes and Hastelloy XR is used for the liner (Ref. 9, pp. 120-121)

**1.1.3.2 Reactor Internals****Graphite Support Structure**

Toyo Tanso supplied the core graphite components, fuel blocks, control rod guide blocks, replaceable reflector blocks, support posts, and post seats. They also manufactured the neutron absorber and boron-carbide/graphite (Ref. 14, p. 387).

The arrangement of graphite and carbon components in the HTTR are shown in Figure 1.78. Properties of the graphite and carbon materials are summarized in Table 1.19. Grade IG-110 graphite is isotropic fine-grained nuclear grade material. Grade PGX graphite is a structural grade with medium-to-fine grained mold. The carbon blocks at the bottom of the HTTR are fabricated from nuclear grade ASR-0RB carbon, so as to keep the metallic core support structures below 500 °C.<sup>a</sup>

High-grade graphite IG-110 with high strength, corrosion, and irradiation resistance was developed with established graphite structural design guidelines (Ref. 3, p. 19). IG-110 graphite materials are highly oxidation resistant.<sup>b</sup> A comparison of the irradiation resistance of graphite used in the HTTR is shown in Figure 1.79. The production process for the IG-110 graphite is shown in Figure 1.80. Some photos from the fabrication and inspection processes are shown in Figures 1.81 and 1.82.

The required impurity limit in the HTTR is a boron equivalent of less than 1 ppm. Seven additional elements with requirements for reduced quantities (due to large activation cross sections, long half-lives for radioactive products, and large concentrations) are shown in Table 1.20. Cobalt impurity has a concentration on the order of 0.01 ppm or less, and is proportional to the iron impurity.<sup>c</sup>

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<sup>a</sup> M. Ishihara, J. Sumita, T. Shibata, T. Iyoku, and T. Oku, "Principle Design and Data of Graphite Components," *Nucl. Eng. Des.*, **233**: 251-260 (2004).

<sup>b</sup> K. Kunitomi and S. Shiozawa, "Safety Design," *Nucl. Eng. Des.*, **233**: 45-58 (2004).

<sup>c</sup> N. Takikawa, T. Iyoku, S. Shiozawa, N. Ooka, M. Kambe, and A. Ide, "Non-Destructive Testing and Acceptance Test for HTTR Graphite Components," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 281-288 (February 1993).

Despite the large variance in the concentration of individual elements, the thermal-neutron absorption cross section of graphite does not exceed 4.5 – 4.8 mb.<sup>a</sup>

Thirteen elements such as Li, B, Na, Mg, Al, Ti, V, Cr, Fe, Mn, Ni, Co, and Zn were analyzed with ICP-MS (inductive coupled plasma-mass spectrometry) and three elements such as K, Ca, and Cu with MA-FLAAS (metal atomizer-flameless atomic absorption spectrophotometry). A quantitative analysis for the rare earths was also performed. The detection limits of the ICP-MS are shown in Figure 1.83. The determined impurities in IG-110 graphite are shown in Table 1.21. Existence of rare earth was trace in these analyses. Uranium and thorium impurity are possible but were not analyzed at the time. Because the graphite is halogen-gas purified, residual halides might be present, but are difficult to detect.<sup>b</sup>

The results from a later analysis of the impurity content are shown in Table 1.22.

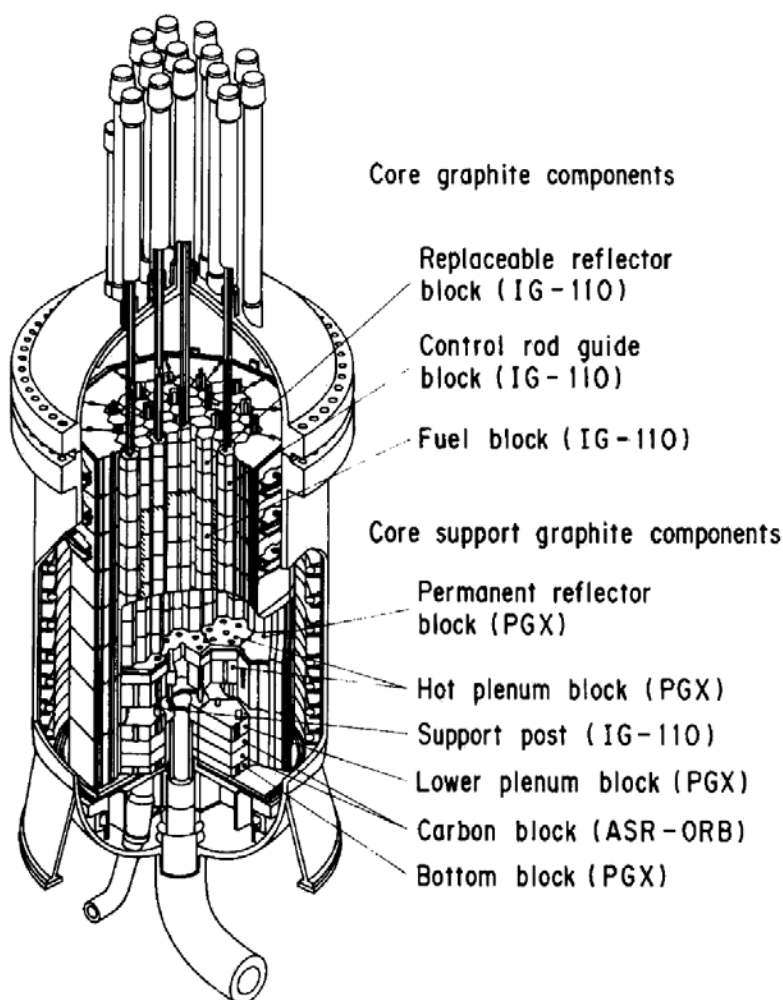


Figure 1.78. Arrangement of Graphite and Carbon Components in the HTTR.<sup>c</sup>

<sup>a</sup> Yu. S. Virgil'ev, "Impurities in and Serviceability of Reactor Graphite," *Atomic Energy*, **84**(1): 6-13 (1998).

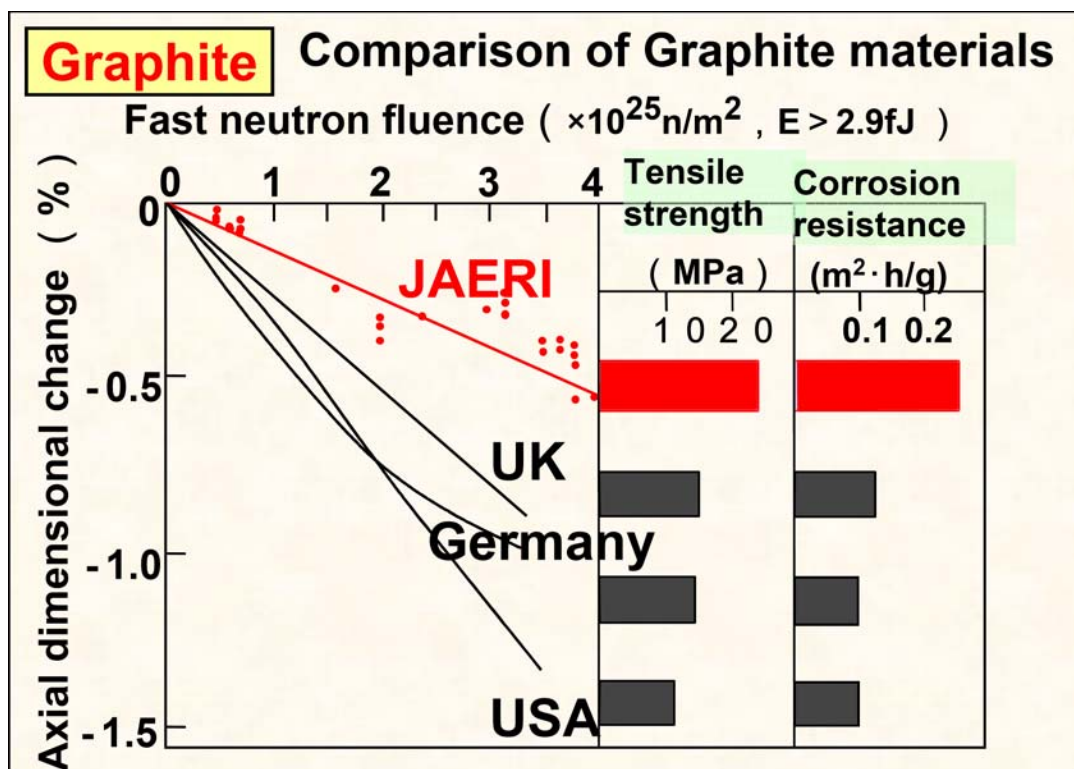
<sup>b</sup> T. Miyatani, H. Suzuki, and O. Yoshimoto, "Quantitative Analysis of Trace Amounts of Impurities Contaminating Pure Graphite with ICP-MS and Metal Atomizer FLAAS," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 304-308 (February 1993).

<sup>c</sup> M. Ishihara, J. Sumita, T. Shibata, T. Iyoku, and T. Oku, "Principle Design and Data of Graphite Components," *Nucl. Eng. Des.*, **233**: 251-260 (2004).

Table 1.19. Typical Thermo-Mechanical Properties of Graphite and Carbon Materials in the HTTR at Unirradiated, Unoxidized, and Room Temperature Conditions (Ref. 6, p. 84).

	IG-110 Graphite	PGX Graphite	ASR-0RB Carbon
Bulk Density ( $\text{mg/m}^3$ ) <sup>(a)</sup>	1.78	1.73	1.65
Mean Tensile Strength (MPa)	25.3	8.1	6.8
Mean Compressive Strength (MPa)	76.8	30.6	50.4
Young's Modulus (GPa)	7.9	6.5	8.7
Mean Thermal Expansion Coefficient from 293-673 K ( $10^{-6}$ K)	4.06	2.34	4.40
Thermal Conductivity (W/m-K at 673 K)	80	75	10
Ash (ppm)	100 Max	7000 Max	5000 Max
Grain Size ( $\mu\text{m}$ )	Mean = 20	800 Max	2000 Max

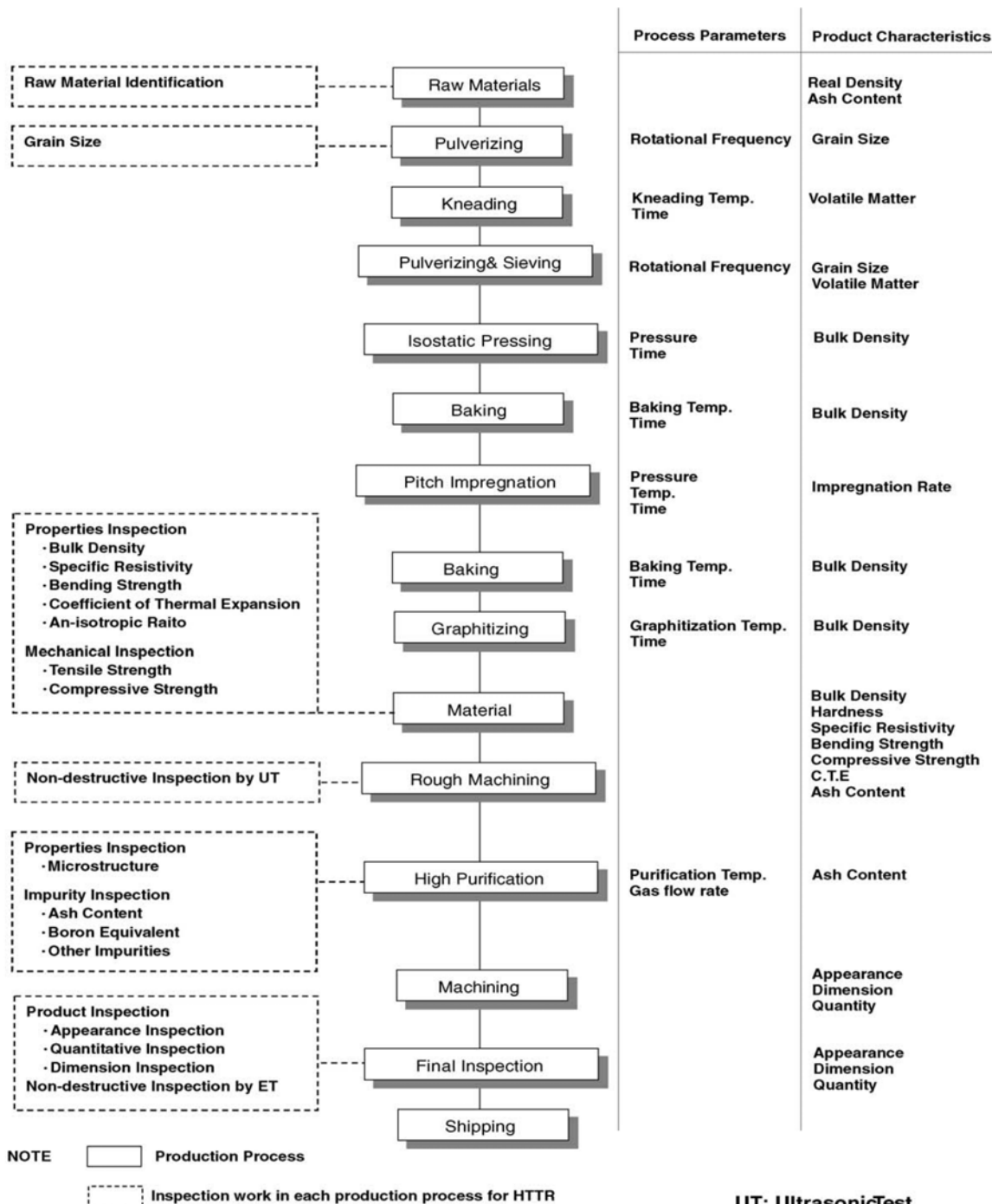
(a) The units for the bulk density were correctly copied from the reference yet do not match up with other reported density units. The author most likely intended to write  $\text{mg/mm}^3$ , which is equal to  $\text{g/cm}^3$ , causing the reported densities to be more realistic.

Figure 1.79. Comparison of Graphite Materials.<sup>a</sup>

<sup>a</sup> S. Shiozawa, "Status of Japan's HTR Program," *Key Note Presentation at the 2007 Int. Congress on Advances in Nuclear Power Plants (ICAPP 2007)*, Nice, France (May 13-18, 2007).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE



(A)

Figure 1.80. Production and Inspection Processes of IG-110 Graphite (Ref. 14, p. 388).



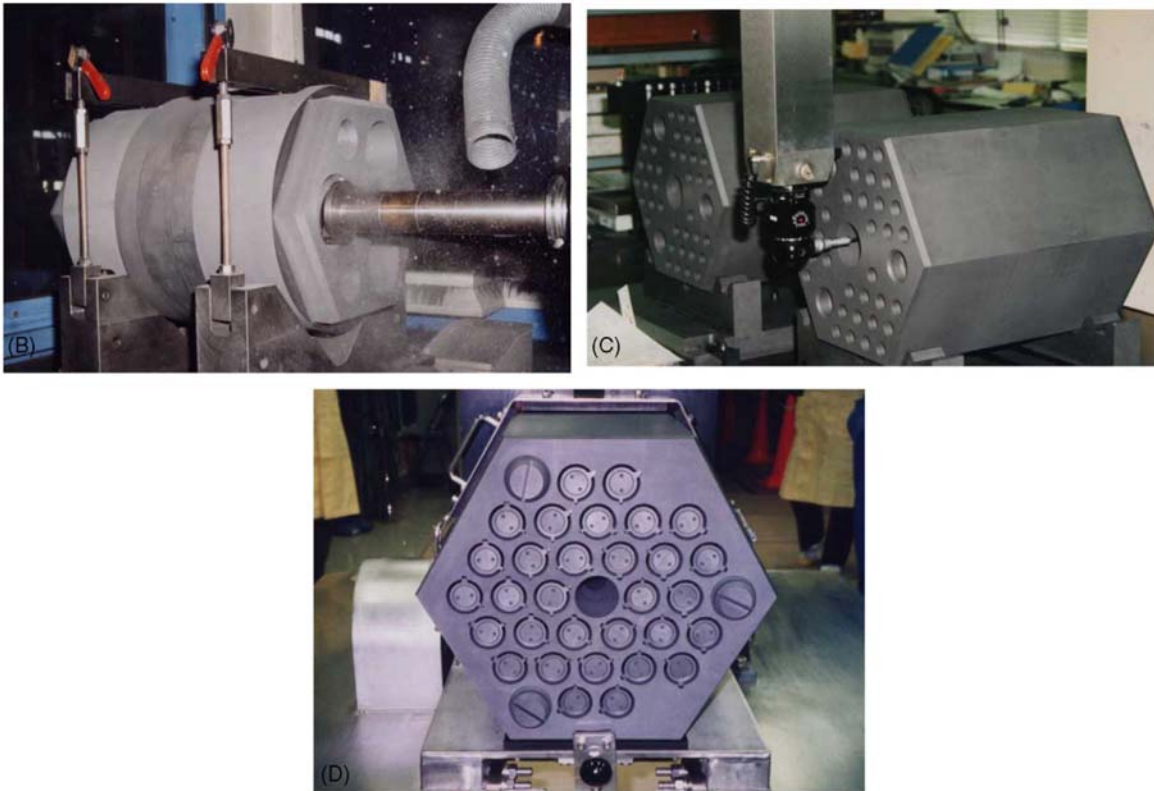


Figure 1.81. Production and Inspection Processes of IG-110 Graphite (Ref. 14, p. 388).



Figure 1.82. HTTR Graphite Block Undergoing Machining.<sup>a</sup>

<sup>a</sup> R. Bratton and T. Burchell, "NGNP Graphite Testing and Qualification Specimen Selection Strategy," INL/EXT-05-00269 (May 2005).



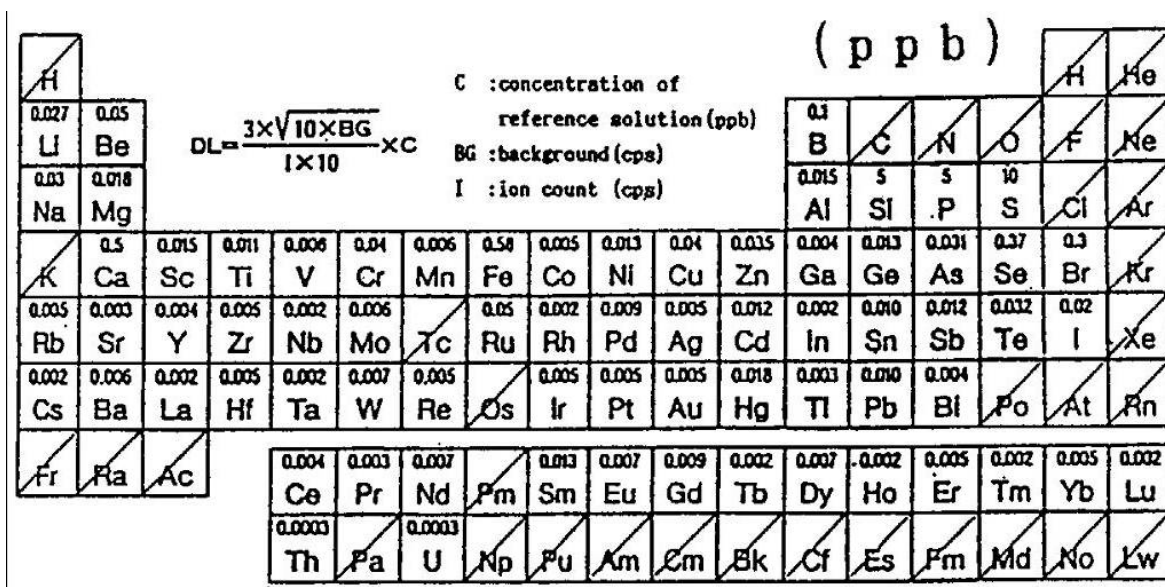
## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATETable 1.20. Typical Graphite Impurities. <sup>(a)</sup>

	Density (g/cm <sup>3</sup> )	Ash (ppm)	Impurity (ppm)					
			Ca	Al	Fe	Ti	V	Si
IG-110	1.75	<80	2	ND <sup>(b)</sup>	2	0.2	ND <sup>(b)</sup>	<0.1
IG-11	1.75	1450	490	30	84	53	180	78
PGX	1.74	1000	116	25	304	38	5	58
ASR-ORB	1.69	3000	--	--	--	--	--	--

(a) S. Nomura, H. Imai, K. Fujii, and M. Shindo, "Relation between Gasification Rates and Gas Desorption Behavior with Metallic Impurities of Carbon and Graphite Materials for the HTTR," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 159-168 (February 1993).

(b) ND = Not Detectable.

Figure 1.83. Detection Limits of ICP-MS. <sup>a</sup>

<sup>a</sup> T. Miyatani, H. Suzuki, and O. Yoshimoto, "Quantitative Analysis of Trace Amounts of Impurities Contaminating Pure Graphite with ICP-MS and Metal Atomizer FLAAS," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 304-308 (February 1993).

Table 1.21. Determined Value of Impurities in the Highly Purified Graphite (IG-110). <sup>(a)</sup>

Element	Determined Value (ppm)	Lower Limit of Determination (ppm)
Li	0.002	0.0004
B	<0.02	0.02
Na	0.05	0.02
Mg	<0.006	0.006
Al	0.06	0.02
Ca	0.08	0.07
K	0.03	0.03
Ti	<0.006	0.006
V	0.018	0.002
Cr	0.006	0.006
Fe	0.06	0.01
Mn	<0.004	0.004
Co	0.014	0.003
Ni	0.006	0.006
Cu	<0.05	0.05
Zn	0.06	0.02

- (a) T. Miyatani, H. Suzuki, and O. Yoshimoto, "Quantitative Analysis of Trace Amounts of Impurities Contaminating Pure Graphite with ICP-MS and Metal Atomizer FLAAS," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 304-308 (February 1993).

Table 1.22. Mass Fraction of Impurities and Ash in Masonry Graphite of the HTTR. <sup>(a)</sup>

Graphite	B	Al	Ca	Ni	V	Fe	Cd	Ash
IG-110	<0.1	0.6	2	0.2	0.2	2	--	82
PGX	--	25	116	38	5	304	--	1000

- (a) Yu. S. Virgil'ev, "Impurities in and Serviceability of Reactor Graphite," *Atomic Energy*, **84**(1): 6-13 (1998).

### **Metallic Support Structure**

High temperature structural materials were chosen for the high temperature components of the HTTR. The materials used include a nickel-based, corrosion and heat resistant superalloy, Hastelloy XR; normalized and tempered (NP) 2-1/4Cr-1Mo steel; two types of austenitic steels, SUS321TB and SUS316; and 1Cr-0.5Mo-V steel, which is an alloy steel bolting material for high temperature service. The components and service conditions for these materials is listed in Table 1.23 (Ref. 12, pp. 261-262).

Nuclear grade Hastelloy XR was developed by tightening the chemical composition specifications of Hastelloy X. Improvements include an optimization of the manganese and silicon contents, lowering of aluminum and titanium contents, lowering of the cobalt content, and optimization of the boron content (addition of within 40-60 ppm). A comparison of Hastelloys X and XR is shown in Table 1.24 (Ref. 12, pp. 262-263). The average composition of the base and weld metals of Hastelloy XR is shown in Table 1.25. A separate study found the average composition of Hastelloy XR to be that reported in Table 1.26.

Although the chemical composition of Hastelloy XR is similar to that of Hastelloy X, the corrosion resistance of Hastelloy XR is much better in high-temperature helium surroundings (Figure 1.84).<sup>a</sup>

Hastelloy XR was developed by JAEA and Misubishi Material Co. (Ref. 14, p.382).

The component materials of the restraint band are SNB16 steel for the tensile element and SUS316 for the compression element. Although the thermal expansion coefficients of the steels are different from those of graphite and the core structural materials, the coefficient of the band itself is adjusted to be equal to that of graphite (Ref. 6, p. 85).

2-1/4Cr-1Mo steel has a variety of applications to pressure-retaining components in service at elevated temperatures, in particular to boilers and liquid metal fast breeder nuclear reactors (LMFBRs) including the Japanese prototype LMFBR “Monju” (Ref. 12, p. 269).

SUS321TB and SUS316 steels are both well-experienced heat-resistant materials with superior mechanical strengths at temperatures up to about 700 °C. In the HTTR, SUS321TB is applied to heat transfer tubes of the PPWC, secondary pressurized water cooler, and the AHX. The tube maximum temperature is about 400 °C during normal operation. SUS316 is applied to reactor internal structures such as a core restraint mechanism and control rod guide tubes. The maximum temperature of these structures is also 400 °C under normal operating conditions (Ref. 12, p. 270).

1Cr-0.5Mo-V steel has widespread application for bolts in chemical plant vessels, and is used as a bolting material for nuclear applications below 425 °C. In the HTTR, 1Cr-0.5Mo-V steel (JIS SNB16) is applied not just in bolts for the RPV and gas circulator castings, but also for parts in the core restraint mechanism (Ref. 12, p. 270).

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<sup>a</sup> T. Takeda, J. Iwatsuki, and Y. Inagaki, “Permeability of Hydrogen and Deuterium of Hastelloy XR,” *J. Nucl. Mat.*, **326**: 47-58 (2004).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

Table 1.23. Materials and Service Conditions of HTTR High Temperature Components (Ref. 12, p 262).

Material	Product Form	Components	Service Conditions Design Temperature (°C)	Design Pressure (MPa)	Maximum Allowable Temperature (°C)
9/4Cr-1Mo Steel	Plate, Forging, Pipe	Reactor Pressure Vessel	440	4.8	550
		Shells of Intermediate Heat Exchanger, Primary Pressurized Water Cooler, etc.	430		
		Outside Pipe of Concentric Double Pipe	430		
Hastelloy XR	Tube, Plate, Forging	Intermediate Heat Exchanger	955	0.29	1000
		Heat Transfer Tubes			
		Intermediate Heat Exchanger Hot Header	940		
SUS321	Tube	Primary Pressurized Water Cooler Heat Transfer Tubes	380	4.8	650
SUS315	Bar	Core Restraint Mechanism	450	--	650
1Cr-0.5Mo-V Steel	Forging	Core Restraint Mechanism	450	--	450

Table 1.24. Chemical Compositions (wt.%) of Hastelloys XR and X (Ref. 12, p 263).

Material	Range	C	Mn	Si	P	S	Cr	Co	Mo	W	Fe	Ni	B	Al	Ti	Ca
XR	Max	0.15	1.00	0.50	0.040	0.030	23.00	2.50	10.00	1.00	20.00	Rest	0.010	0.05	0.03	0.50
	Min	0.05	0.75	0.25	--	--	20.50	--	8.00	0.20	17.00	Rest	--	--	--	--
X	Max	0.15	1.00	1.00	0.040	0.030	23.00	2.50	10.00	1.00	20.00	Rest	0.010	0.50	0.15	0.50
	Min	0.05	--	--	--	--	20.50	0.50	8.00	0.20	17.00	Rest	--	--	--	--

Table 1.25. Chemical Composition (wt.%) of Hastelloys XR Base and Weld Material. <sup>(a)</sup>

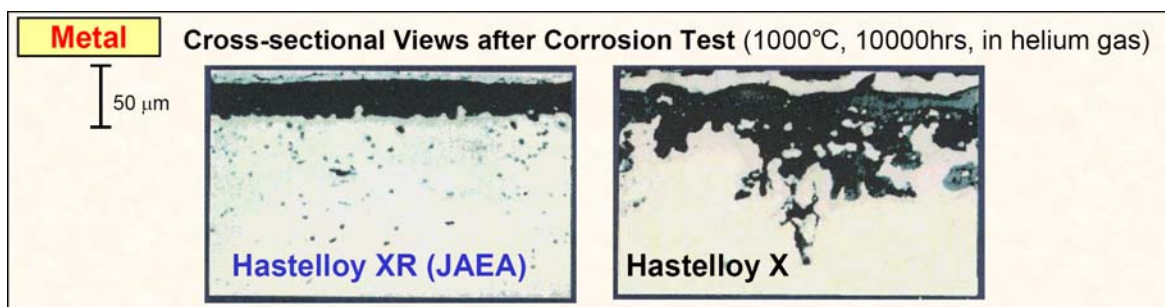
Material	C	Mn	Si	Cr	Co	Mo	W	Fe	Ni	B	Al	Ti
Base	0.060	0.89	0.33	21.6	0.03	9.06	0.49	17.99	Rest	0.0037	<0.01	0.01
Weld	0.051	0.80	0.39	21.3	0.001	9.20	0.53	18.2	Rest	0.0033	0.05	0.002

(a) Y. Kurata, T. Tanabe, I. Mutoh, H. Tsuji, K. Hiraga, M. Shindo, and T. Suzuki, "Creep Properties of Base Metal and Welded Joint of Hastelloy XR Produced for High-Temperature Engineering Test Reactor in Simulated Primary Coolant Helium," *J. Nucl. Sci. Tech.*, **36**(12): 1160-1166 (December 1999).

Table 1.26. Chemical Composition (wt.%) of Hastelloys XR.<sup>(a)</sup>

C	Mn	Si	P	S	Cr	Co	Mo	W	Fe	Ni	B	Al	Ti	Cu	N
0.08	0.94	0.36	<0.001	0.001	21.18	0.01	9.07	0.49	18.02	Rest	0.0021	0.02	<0.01	<0.01	0.009

- (a) Y. Kaji, H. Tsuji, H. Nishi, Y. Muto, H-J. Penkalla, and F. Schubert, "Multiaxial Creep Behavior of Nickel-Base Heat-Resistant Alloys Hastelloy XR and Ni-Cr-W Superalloy at Elevated Temperature," *J. Nucl. Sci. Tech.*, **39**(8): 923-928 (August 2002).

Figure 1.84. Demonstration of Improved Corrosion Resistance of Hastelloy XR.<sup>a</sup>

### Reactor Core Shielding

The top shielding block is hexagonal and composed of sintered B<sub>4</sub>C/C, a casing of SUS316, and dowel pins. The side shielding block is also composed of B<sub>4</sub>C/C and a SUS316 casing. The support plate is composed of SUS316 (Ref. 6, p. 86).

### General Instrumentation and Control Systems

Hastelloy XR was implemented in the instrumentation and control systems of the HTTR due to the high temperature environment. Heat and corrosion resistant superalloy Hastelloy XR can be used at temperatures as high as 950 °C at normal operation and 1000 °C in accidents. High temperature structural design guidelines for high temperature metallic components were established (Ref. 3, p. 19).

#### 1.1.3.3 Reactor Core Components

### Reactor Core

The core consists of nine block layers with 30 columns in the active core surrounded by removable and permanent reflectors, the composition of which is described below.

### Prismatic Fuel

Some fuel material properties are contained within Tables 1.13 and 1.14. Uranium enrichment details are provided in Table 1.11.

Figure 1.85 shows a flow diagram of the HTTR fuel production process. The UO<sub>2</sub> kernels were fabricated in a gel-precipitation process, where a uranyl nitrate solution containing methocel and an additive was dropped into concentrated ammonia solution. Gel-particles formed in the ammonia solution

<sup>a</sup> S. Shiozawa, "Status of Japan's HTR Program," *Key Note Presentation at the 2007 Int. Congress on Advances in Nuclear Power Plants (ICAPP 2007)*, Nice, France (May 13-18, 2007).

were then dried, calcinated, and sintered. The coating layers were deposited on the kernels in a CVD process using a fluidized coater. The buffer and high density PyC coating layers were derived from  $C_2H_2$  and  $C_3H_6$ , respectively, and the SiC layer from  $CH_3SiCl_3$ . In the fabrication process of the fuel compacts, the coated particles were overcoated in advance with phenol-resinated graphite powder, and these particles were formed into green compacts by warm-press. The green compacts were carbonized by a heat treatment at 800 °C, followed by degassing at 1800 °C.

In the process of the fuel compact fabrication, the overcoat of the graphite matrix powder was affixed on the coated particles aimed to protect the coating layer from mechanical damage by pressing in metal dies and by graphite-matrix shrinkage during heat treatments of the green compacts. Dual-layered overcoat of about 200  $\mu m$  thickness was applied, providing 30 vol. % packing fraction of the coated particles in the fuel compacts. The first layer of overcoat was 80  $\mu m$  thick, followed by curing at about 180 °C. Then the second layer was added to achieve the desired packing fraction.

The resinated graphite powder is prepared by mixing electrographite powder, natural graphite powder, and phenol resin as a binder in the ratio 16:64:20, followed by grinding the mixture to powder.<sup>a</sup>

The final heat-treatment of the green fuel compacts is at 1073 K in flowing  $N_2$  to carbonize the binder. Then the compacts are degassed in vacuum at 2073 K.<sup>b</sup>

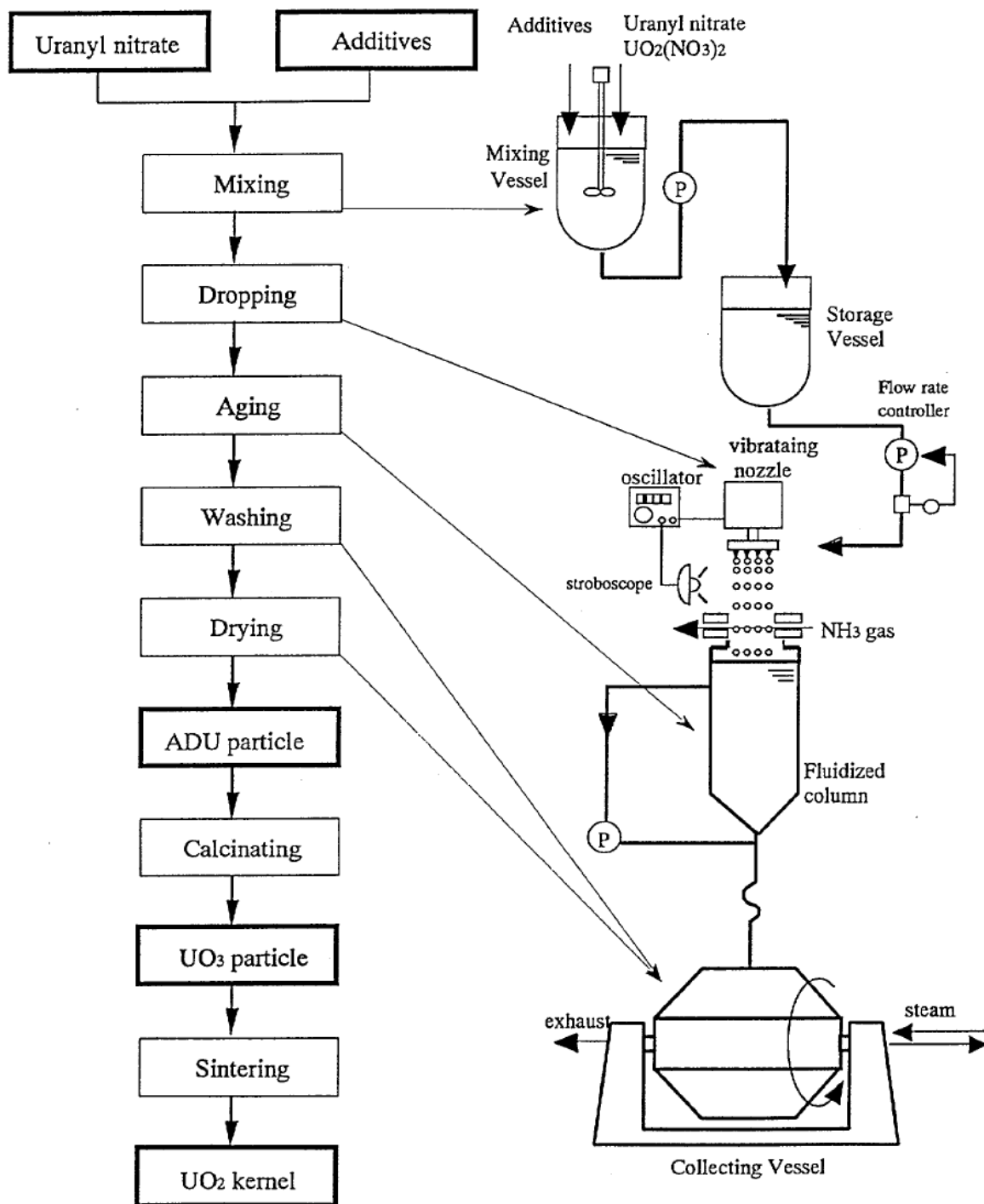
The fabrication schedule for the first HTTR fuel loading is shown in Figure 1.86.

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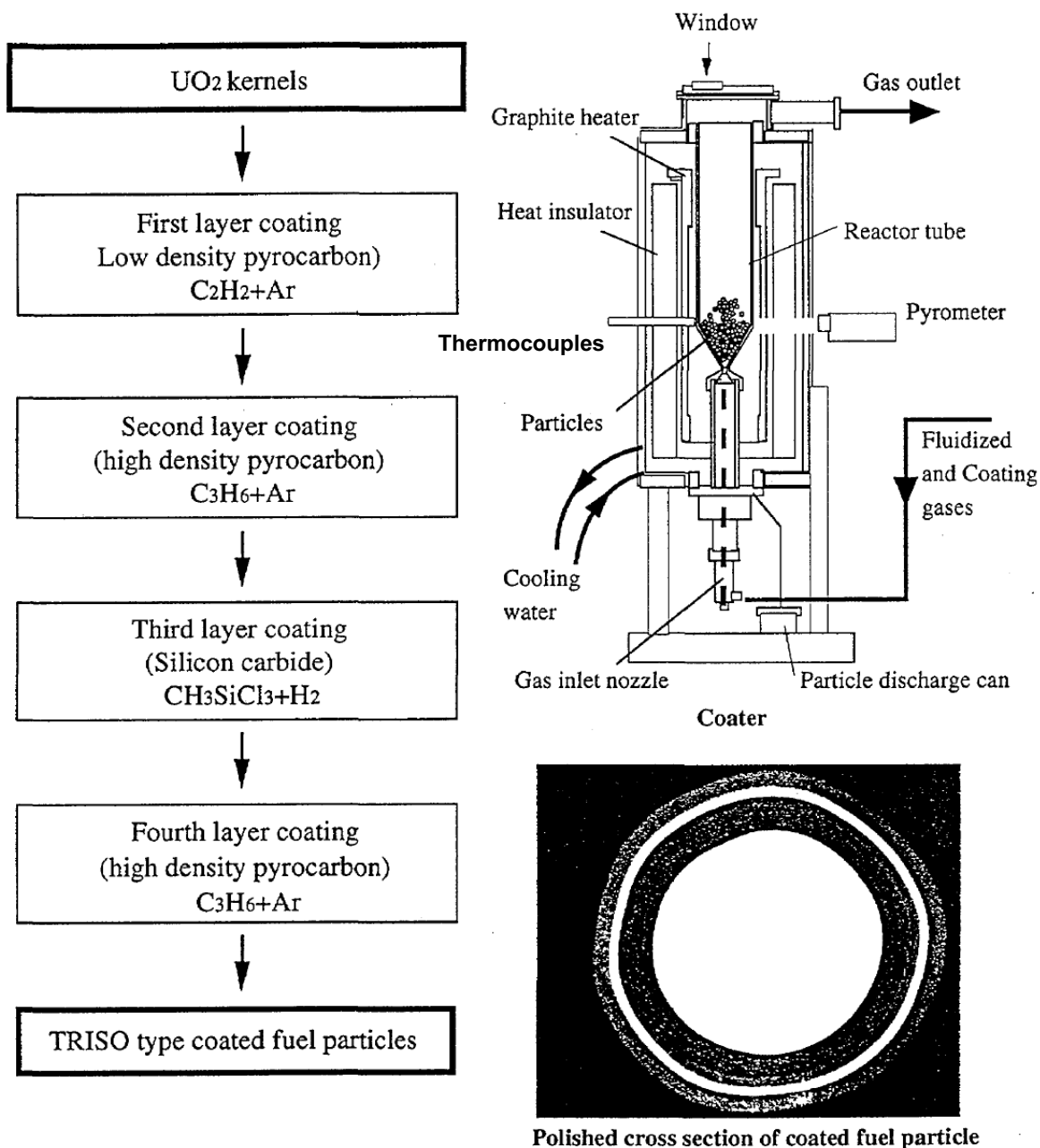
<sup>a</sup> K. Minato, H. Kikuchi, T. Tobita, K. Fukuda, M. Kaneko, N. Suzuki, S. Yoshimuta, and H. Tomimoto, "Improvements in Quality of As-Manufactured Fuels for High-Temperature Gas-Cooled Reactors," *J. Nucl. Sci. Tech.*, **34**(3): 325-333 (March 1997).

<sup>b</sup> K. Minato, H. Kikuchi, T. Tobita, K. Fukuda, M. Kaneko, N. Suzuki, S. Yoshimuta, and H. Tomimoto, "Improvements in Quality of As-Manufactured Fuels for High-Temperature Gas-Cooled Reactors," *J. Nucl. Sci. Tech.*, **34**(3): 325-333 (March 1997).

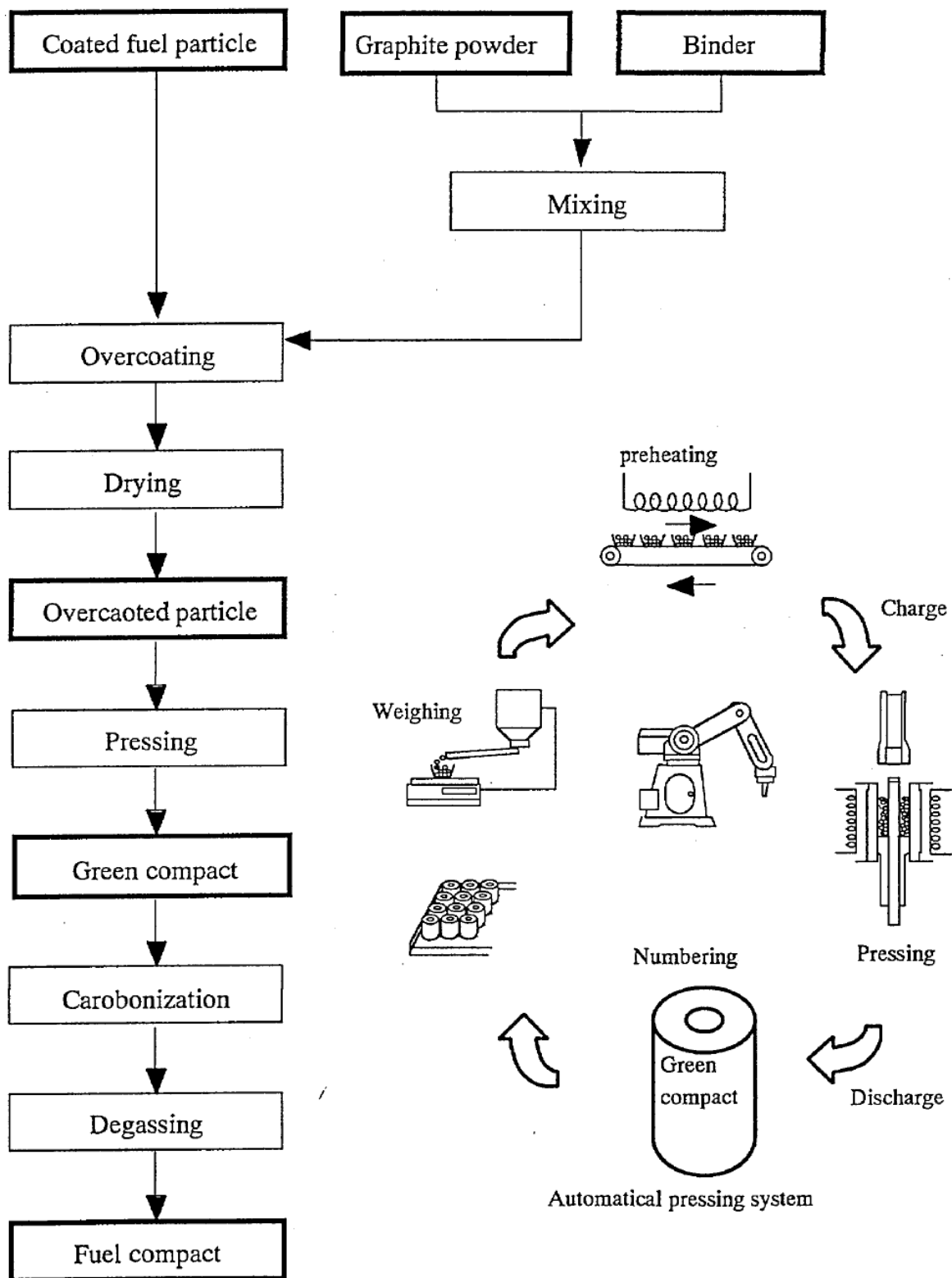


Figure 1.85. Flow Diagram of HTTR Fuel Fabrication Process.<sup>a</sup>

<sup>a</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

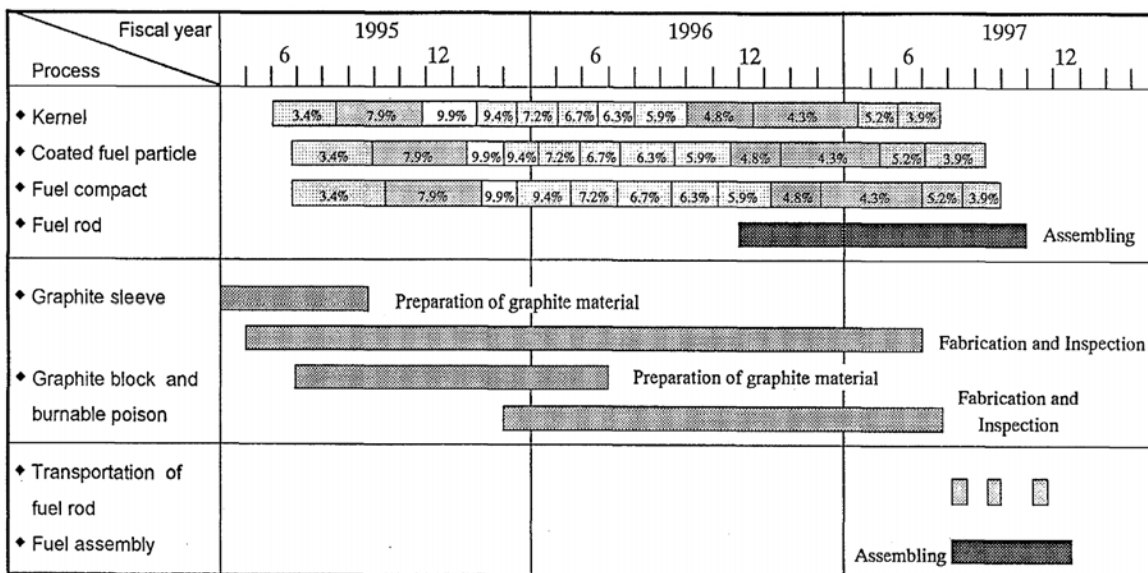
Figure 1.85 (cont.). Flow Diagram of HTTR Fuel Fabrication Process.<sup>a</sup>

<sup>a</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

Figure 1.85 (cont.). Flow Diagram of HTTR Fuel Fabrication Process.<sup>a</sup>

<sup>a</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATEFigure 1.86. Fabrication Schedule of the First-Loading Fuel of the HTTR.<sup>a</sup>

Various mandatory fuel inspection standards were employed. The sphericity of the fuel kernel and coated fuel particles should be less than 1.2 within a 95 % confidence limit to prevent locally high stress during irradiation. The O/U ratio in the fuel kernel should be almost 2.0 to assure irradiation performance. The coating layer density and thickness must confirm integrity of coated fuel particles during irradiation. The OPTAF in high-density PyC layers should be less than 1.04 to prevent excessive deformation by fast neutron irradiation. The failure fractions (exposed uranium and SiC-failure) must be sufficiently low so as to provide fission product retention.

The enrichment of  $^{235}\text{U}$  and uranium content in the fuel compact were measured by  $\gamma$ -ray spectro analysis. Diameter and sphericity of fuel kernels and the coated fuel particles were measured by optical particle size analysis. Density of the fuel kernel was measured by mercury substitution method. The O/U ratio was measured by oxidation and weighing. Impurities were measured by emission spectro analysis. Coating layer thickness and density were measured by X-ray radiograph and solvent substitution or sink-float methods, respectively. The OPTAF of the PyC layers was measured by polarization photometry. Appearance and ceramographed cross sections of the coated fuel particles and the fuel compacts were visually inspected. Strength of the coated fuel particles was measured by point crushing.

Density, impurities, grain size, water content in graphite powder, ash, melting point, and impurities in binder are inspected. Exposed uranium fraction and SiC-failure fraction are measured by deconsolidation/acid leaching and burn/acid leaching, respectively. Packing fraction of coated fuel particles and matrix density in a fuel compact is measured by weighing and calculation. Dimensions of the fuel compacts were measured by micrometer. Marking on the fuel compact was inspected by visual observation. Compressive strength of the fuel compact was also measured.

Uranium content in the fuel rods is confirmed by calculation of measured fuel compacts. Total length of the fuel rod was measured. The number of fuel compacts in a fuel rod is checked by assembling record. The Stack length of the fuel compacts is checked by calculation. Surface contamination of the fuel rod is measured by smear method. Appearance of the fuel rod and the fuel assembly is visually observed. The

<sup>a</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

$^{235}\text{U}$  enrichment is checked by fuel compacts and graphite sleeve markings. Components in the fuel rod and the fuel assembly are checked by assembling record. The uranium content in the fuel assembly was calculated. Weight of the fuel assembly was also measured.<sup>a</sup>

The total number of particles with a through-coating failure in the active core has been estimated to be 2500, which translates into a fraction of about  $2.5 \times 10^{-6}$ . The contamination of the fuel element matrix graphite has been estimated to be approximately 2.5 g of uranium. According to the total loading of 900 kg of U in the core, the respective contamination fraction is about  $2.5 \times 10^{-6}$ . Free uranium is considered to exist as heavy metal contamination of the fuel graphite, as particles with a through-coating failure, and those with a defective SiC layer. The irradiation performance of fuel particles in the HTTR is shown in Figure 1.86; the expected fuel quality for the first-core fuel in the HTTR is shown in Table 1.26.<sup>b</sup>

The free uranium fractions of the fuel compacts were measured by the deconsolidation followed by the acid leaching on four fuel compacts for each fuel compact fabrication lot (containing up to about 700 fuel compacts). The HTTR first-loading fuel compacts were fabricated from 126 fuel compact lots.<sup>c</sup>

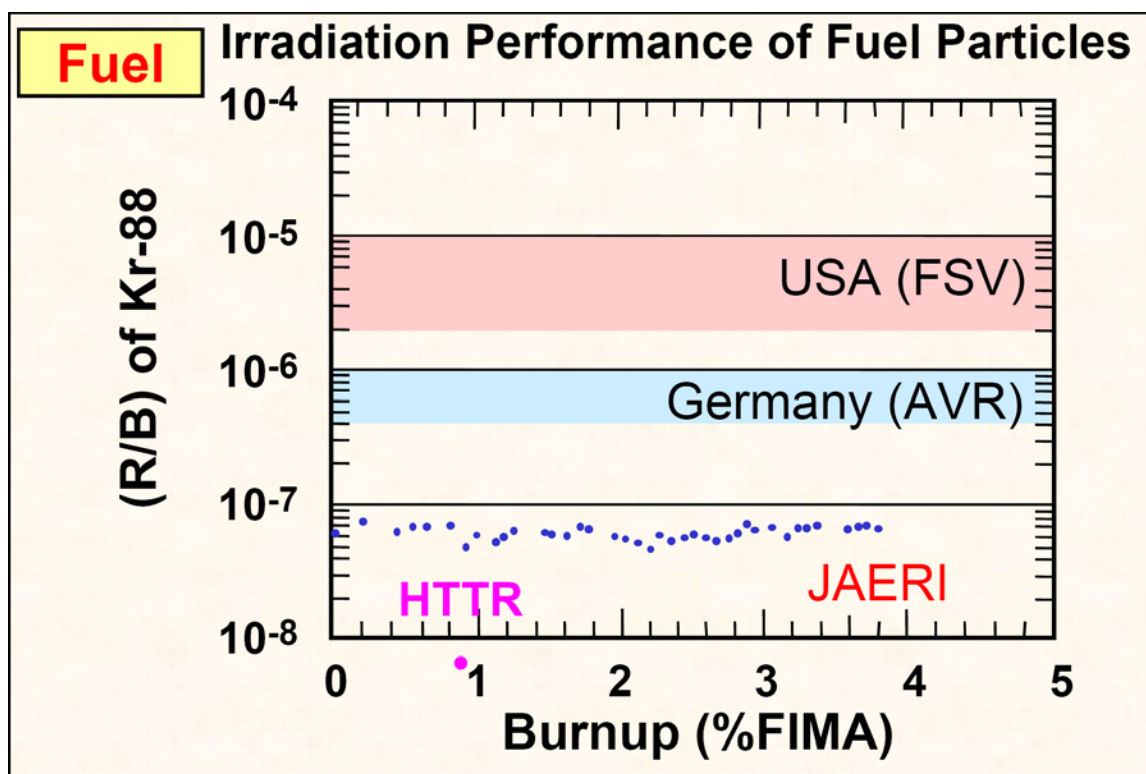


Figure 1.86. Irradiation Performance of HTTR Fuel Particles.<sup>d</sup>  
R/B represents the rate of release over the rate of fuel burnup.

<sup>a</sup> K. Sawa, S. Suzuki, and S. Shiozawa, "Safety Criteria and Quality Control of HTTR Fuel," *Nucl. Eng. Des.*, **208**: 305-313 (2001).

<sup>b</sup> K. Verfondern, J. Sumita, S. Ueta, and K. Sawa, "Modeling of Fuel Performance and Metallic Fission Product Release Behavior During HTTR Normal Operating Conditions," *Nucl. Eng. Des.*, **210**: 225-238 (2001).

<sup>c</sup> K. Sawa, T. Tobita, H. Mogi, S. Shiozawa, S. Yoshimuta, S. Suzuki, and K. Deushi, "Fabrication of the First-Loading Fuel of the High Temperature Engineering Test Reactor," *J. Nucl. Sci. Tech.*, **36**(8): 683-690 (August 1999).

<sup>d</sup> S. Shiozawa, "Status of Japan's HTR Program," *Key Note Presentation at the 2007 Int. Congress on Advances in Nuclear Power Plants (ICAPP 2007)*, Nice, France (May 13-18, 2007).

Table 1.26. Fuel Quality of the HTTR First Core Fuel Production. <sup>(a)</sup>

Contribution	Defect Fraction		
	Expected	Regulatory	Safety Design
Heavy Metal Contamination	$2.5 \times 10^{-6}$	$1.5 \times 10^{-4}$	$2.0 \times 10^{-3}$
Initial Through-Coating Defects	$2.5 \times 10^{-6}$	--	--
Initial SiC Defects	$8.0 \times 10^{-5}$	$1.5 \times 10^{-3}$	--
Irradiation-Induced Failures	0	$2.0 \times 10^{-3}$	$8.0 \times 10^{-3}$
Total at EOL	$8.5 \times 10^{-5}$	$3.7 \times 10^{-3}$	$1.0 \times 10^{-2}$

(a) K. Verfondern, J. Sumita, S. Ueta, and K. Sawa, "Modeling of Fuel Performance and Metallic Fission Product Release Behavior During HTTR Normal Operating Conditions," *Nucl. Eng. Des.*, **210**: 225-238 (2001).

Additional specified impurity parameters for some of the graphite components in the active core of the HTTR are listed in Table 1.27.

Table 1.27. Parameters for Graphites (Ref. 2, p. 105).

Component	Density (g/cm <sup>3</sup> )	Impurity (ppm B <sub>nat</sub> )
Matrix	1.690	0.82
Sleeve	1.770	0.37
Fuel / Control Block	1.770	0.40
Replaceable Reflector	1.760	0.37
Permanent Reflector	1.732	1.91

### **Burnable Poisons**

The compositions of the BP pellets are B<sub>4</sub>C and C. There are two types of BP pellets: H-I with 2.22 at.% natural boron concentration and H-II with 2.74 at.%, with material properties shown in Table 1.28 (Ref. 2, p. 21).



Table 1.28. Material Properties of BP and Graphite Disks (Ref. 2, p. 17).

Burnable Poison Pellet		
Material	B <sub>4</sub> C-C Composite	
Type	H-I	H-II
Density (g/cm <sup>3</sup> )	1.79	1.82
Natural Boron Concentration (at.%)	2.22	2.74
Diameter (mm)	13.9	13.9
<sup>10</sup> B Abundance Ratio (wt.%)	18.7 <sup>(a)</sup>	18.7 <sup>(a)</sup>
Graphite Disk		
Material	Graphite	
Diameter (mm)	14.0	
Density (g/cm <sup>3</sup> )	1.77	
Impurity (ppm, natural B equivalent)	0.37	

(a) These may be typographical errors or listed in atomic percent since natural boron has an abundance between 19.1 and 20.3 wt.% (see Section 2.1.3.4).

### **Control Rods and Reserve Shutdown System**

Some control rod material properties are contained within Table 1.15.

Typical impurities in sintered B<sub>4</sub>C pellets are shown in Table 1.29. Information is provided as a comparison of manufacturing results.

Ferritic superalloy Alloy 800H is selected for material of metallic parts of the control rod. Alloy 800H is selected for the metal parts of the control rods mainly because iron-based alloys are superior to nickel-rich alloys in both post-irradiation tensile and creep properties (Ref. 7, pp. 91 and 93).

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CRIT-SUB-REAC-COEF-KIN-RRATETable 1.29. Typical Impurities in Sintered B<sub>4</sub>C/C Pellets. <sup>(a)</sup>

	B <sub>4</sub> C/C (30 %)	B <sub>4</sub> C/C (40 %)	B <sub>4</sub> C/C (50 %)	B <sub>4</sub> C
Pellet Density (g/cm <sup>3</sup> )	2.04 ± 0.03	2.06 ± 0.03	2.25 ± 0.03	2.31
Boron Content (wt.%)	30 ± 3	40 ± 3	50 ± 3	78
Impurity (ppm)	Na	~100	~100	~10
	Al	~1000	~1000	~1000
	Si	~1000	~1000	~1000
	Ca	~1000	~1000	~100
	Ti	~100	~1000	~1000
	Mn	~10	~10	~10

(a) K. Fujii, S. Nomura, H. Imai, and M. Shindo, "Corrosion Behavior of Sintered Pellet of Graphite and Boron Carbide in Helium Containing Water Vapor," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 169-176 (February 2003).

Some control rod guide block material properties are contained within Table 1.16.

### **Instrumentation**

No material information is available regarding the BF<sub>3</sub> counters, micro-fission chambers, or the gamma-ray compensated ionization chambers.

### **Graphite Blocks**

Graphite composition is discussed in the section associated with Tables 1.19 through 1.22.

### **Dummy Blocks**

Dummy fuel blocks contain higher impurity than the fuel graphite blocks (Ref. 2, p. 23).

While the material of blocks and dowels in the HTTR is IG-110, typically IG-11 was used for research studies of various components. IG-11 is the same as IG-110 except without purity and economy.<sup>a</sup>

Some of the graphite used for the dummy blocks was originally manufactured for out-of-pile seismic tests, and there was an uncertain amount of impurity in them. The impurity in the boron equivalent was 3.1 ppm (Ref. 1, p. 314).<sup>b</sup>

### **Replaceable Reflectors**

Some replaceable reflector block material properties are contained within Table 1.16. Additional information is provided in the section associated with Tables 1.19 through 1.22.

<sup>a</sup> N. Takikawa, M. Ishihara, T. Iyoku, S. Shiozawa, M. Tokumitsu, S. Koe, and M. Uno, "Assessment of the Load Capacity of the Dowel and Socket System in the HTTR Hexagonal Block," IAEA-TECDOC-690, International Atomic Energy Agency, Vienna, pp. 105-112 (February 1993).

<sup>b</sup> Fujimoto, N., Nakano, M., Takeuchi, M., Fujisaki, S., and Yamashita, K., "Start-Up Core Physics Tests of High Temperature Engineering Test Reactor (HTTR), (II): First Criticality by an Annular Form Fuel Loading and Its Criticality Prediction Method," *J. Atomic Energy Society Japan*, **42**(5), 458-464 (2000).

## **Permanent Reflectors**

Some permanent reflector material properties are contained within Table 1.16. Additional information is provided in the section associated with Tables 1.19 through 1.22.

The permanent reflectors are fabricated from grade PGX graphite (Ref. 6, p. 83).

Around the permanent reflector blocks are side shielding blocks consisting of B<sub>4</sub>C/C (Ref. 2, p. 23).

## **Helium Coolant**

See Table 1.10 for the upper impurity limit concentration in the primary helium coolant.

The primary helium purification system of the HTTR is designed to have hydrogen concentration limited to less than 3 ppm.<sup>a</sup>

The impurity limits provided are for full power operation. During the cold critical tests, the helium coolant impurity concentrations would be negligible.

### **1.1.4 Temperature Data**

The core is at room temperature (Ref. 2, p. 14). Table 1.17 provided the temperature of the experiment for each of the six critical cases, ranging between 23 and 25°C.

### **1.1.5 Additional Information Relevant to Critical and Subcritical Measurements**

Additional information is not available.

## **1.2 Description of Buckling and Extrapolation Length Measurements**

Buckling and extrapolation length measurements were not made.

## **1.3 Description of Spectral Characteristics Measurements**

Spectral characteristics measurements were not made.

## **1.4 Description of Reactivity Effects Measurements**

### **1.4.1 Overview of Measurement**

See Section 1.1.1.

Experimental information gathered from the references is provided in this section, but not currently evaluated.

### **1.4.2 Geometry of the Experiment Configuration and Measurement Procedure**

See Section 1.1.2.

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<sup>a</sup> T. Takeda, J. Iwatsuki, and Y. Inagaki, "Permeability of Hydrogen and Deuterium of Hastelloy XR," *J. Nucl. Mat.*, **326**: 47-58 (2004).

#### 1.4.2.1 Excess Reactivity

After the initial criticality, reactivity increase was measured using the inverse kinetic (IK) method. The excess reactivity of the core was then obtained by adding all increments of the reactivity from the first criticality to the fully-loaded core (Ref. 2, p. 14).

A fuel addition method was applied for the measurement of the excess reactivity. The CR position at criticality became deeper after fuel loading, allowing for measurement of the change in reactivity. The reactivity of the inserted part of each CR was measured via the IK method after every addition of three fuel columns. The measurement was performed for the 21-, 24-, 27-, and 30-fuel-column cores. It was expected that the excess reactivity could be obtained by summation of the reactivity of each CR; however, the measurement was strongly affected by the negative shadowing effect between neighboring CRs. The measured excess reactivity was then revised with the following equations to correct for the shadowing effects:

$$\rho_{exm} = \sum_{step}^n \Delta\rho_{exm} ,$$

$$\Delta\rho_{exm} = R \cdot \Delta\rho_{ikm} ,$$

and

$$\Delta\rho_{ikm} = \sum_{CR}^m \Delta\rho_{ikRod} ,$$

where n is the number of fuel addition steps, m is the number of CRs,  $\rho_{exm}$  is the revised excess reactivity (%  $\Delta k/k$ ),  $\Delta\rho_{exm}$  is the revised increase in excess reactivity (%  $\Delta k/k$ ),  $\Delta\rho_{ikm}$  is the increase in excess reactivity measured with IK method after fuel addition (%  $\Delta k/k$ ), and  $\Delta\rho_{ikRod}$  is the measured reactivity of inserted part of CR after fuel addition (%  $\Delta k/k$ ). The revising factor, R, was estimated with diffusion calculation using the nuclear design code system of the HTTR with the following relation:

$$R = \frac{\Delta\rho_{excal}}{\Delta\rho_{ikcal}} ,$$

where  $\Delta\rho_{ikcal}$  is the increase in excess calculated according the IK measurement procedure (% $\Delta k/k$ ), and  $\Delta\rho_{excal}$  is the increase in excess reactivity calculated by withdrawing all CRs together (% $\Delta k/k$ ). The theoretical excess reactivity was calculated with the following relation:

$$\rho_{exMVP} = \frac{k-1}{k} ,$$

where k was obtained with the Japanese Monte Carlo MVP code (150 batches  $\times$  20000 histories/batch). The revised and calculated excess reactivity is shown in Table 1.30; they agree within the experimental errors for each core, and the difference between the measured and calculated mean values was reduced from 3.5 to 1 %  $\Delta k/k$  by introducing the revising factor (Ref. 1, pp. 218-219).

The excess reactivity for core configurations containing 18 and 19 fuel columns are reported as -0.9 and 1.5 %  $\Delta k/k$ , respectively (Ref. 13, p. 286).

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Fuel Columns	Measurement		Revising Factor R <sup>(b)</sup>	Revised Excess Reactivity		Calculation	Difference ( $\rho_{\text{exMVP}} - \rho_{\text{exm}}$ ) (%Δk/k)
	$\rho_{\text{ikm}}$ (%Δk/k)	$\Delta\rho_{\text{ikm}}$ (%Δk/k)		$\Delta\rho_{\text{exm}}$ (%Δk/k)	$\rho_{\text{exm}}$ (%Δk/k)	$\rho_{\text{exMVP}}$ (%Δk/k)	
21	2.3±0.23	2.3±0.23	1.69±0.12	4.0±1.1	4.0±1.1	5.0±0.15	1.0
24	5.0±0.50	2.7±0.27	1.40±0.11	3.7±1.0	7.7±2.1	8.6±0.15	0.9
27	7.4±0.74	2.4±0.24	1.26±0.15	3.0±0.9	10.7±3.0	11.3±0.15	0.6
30	8.5±0.85	1.1±0.11	1.19±0.04	1.3±0.3	12.0±3.3	12.0±0.15	0.0

(a) Based upon a reported  $\beta_{\text{eff}}$  of 0.0065 (Ref. 2, p. 42).(b) The error of R is evaluated from errors of calculated CR worth  $\Delta\rho_{\text{ikcal}}$  and  $\Delta\rho_{\text{excal}}$ . The errors of CR worth were determined in analyses of the critical assembly VHTRC experiments.

## 1.4.2.2 Shutdown Margin

The scram reactivity, or shutdown margin, was evaluated for two cases: 1) all reflector CRs inserted at critical condition, and 2) all CRs in reflector and core inserted at critical condition. The core was fully loaded with fresh fuel.

The scram reactivity for Case 1 was evaluated as follows:

$$\rho_R = \frac{k_{\text{Crit}} - k_{\text{RCR-in}}}{k_{\text{Crit}} \cdot k_{\text{RCR-in}}},$$

where  $\rho_R$  is the scram reactivity of the reflector CRs (Δk/k),  $k_{\text{Crit}}$  is the effective multiplication factor at critical CR position, and  $k_{\text{RCR-in}}$  is the effective multiplication factor at CR position after scram.

The temperature of the core and reflector is 300 K. Critical rod positions for Case 1 are provided in Table 1.31 (Ref. 2, pp. 26 and 42).

Table 1.31. Control Rod Position (mm) for Case 1 (Ref. 2, p. 42).

CR Group	Critical Position	Position after Scram
C	1775	1775
R1	1775	1775
R2	1775	-55 <sup>(a)</sup>
R3	Full Out	-55 <sup>(a)</sup>

(a) Control rods are inserted slightly into the top of the bottom reflector.

The scram reactivity of all control rods (Case 2) was evaluated as follows:

$$\rho_A = \frac{k_{\text{Crit}} - k_{\text{ACR-in}}}{k_{\text{Crit}} \cdot k_{\text{ACR-in}}},$$

where  $\rho_A$  is the scram reactivity of all CRs (Δk/k),  $k_{\text{Crit}}$  is the effective multiplication factor at critical CR position, and  $k_{\text{ACR-in}}$  is the effective multiplication factor at CR position after scram.

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The temperature for core and reflector is 300 K. Critical rod positions for Case 2 are provided in Table 1.32 (Ref. 2, pp. 26 and 43).

Table 1.32. Control Rod Position (mm) for Case 2 (Ref. 2, p. 43).

CR Group	Critical Position	Position after Scram
C	1775	-55 <sup>(a)</sup>
R1	1775	-55 <sup>(a)</sup>
R2	1775	-55 <sup>(a)</sup>
R3	Full Out	-55 <sup>(a)</sup>

(a) Control rods are inserted slightly into the top of the bottom reflector.

The measured scram reactivity for Cases 1 and 2 were 12.1 and 46.3 % $\Delta k/k$ , respectively (Ref. 2, p. 44). These values were then later reported (in the same document) as 12.0 and 46.0 % $\Delta k/k$ , respectively (Ref. 2, p. 331).

Elsewhere, the shutdown margins, which were determined via the IK and rod drop methods, were reported as shown in Table 1.33. Generally the rod-drop method is used for shutdown margin measurement. However, as it takes about 10 seconds for the CR insertion of the HTTR, the rod-drop method was not directly applicable. The measured shutdown margin becomes smaller than the real value when it takes a few seconds for CR insertion. Therefore the IK method was tried for measurement of the shutdown margin and the reactivity was evaluated continuously by computer from the neutron density signals of the CIC detectors using the IK method.<sup>a</sup>

Table 1.33. Shutdown Margins of the HTTR (Ref. 12, p. 287).

	Experiment (% $\Delta k/k$ )		Calculation (% $\Delta k/k$ )
	IK Method	Rod Drop Method	
One-Step Scram			
All Control Rods	-46.7	-43.7	-42.9
Two-Step Scram			
1 <sup>st</sup> : Control Rods in Reflector Region	-12.1	--	-8.6
2 <sup>nd</sup> : Control Rods in Fuel Region	-34.2	--	-34.3
All Control Rods (Total)	-46.3	-49.0	-42.9

#### 1.4.2.3 Control Rod Worth

The reactivity worth of each control rod was continuously measured by the inverse kinetic method from full insertion to full withdrawal. The differential control rod worth for the center control rod is shown in Figure 1.87. The full insertion and withdrawal levels correspond to 0 and 4060 mm, respectively, in

<sup>a</sup> "Present Status of HTGR Research and Development," Japan Atomic Energy Research Institute, Tokaimura, March 2004.

axial distance from the bottom of the fuel region. The reactivity addition rate is below the limit of  $2.4 \times 10^{-4} \Delta k/k\text{-s}$  (Ref. 13, p. 287).

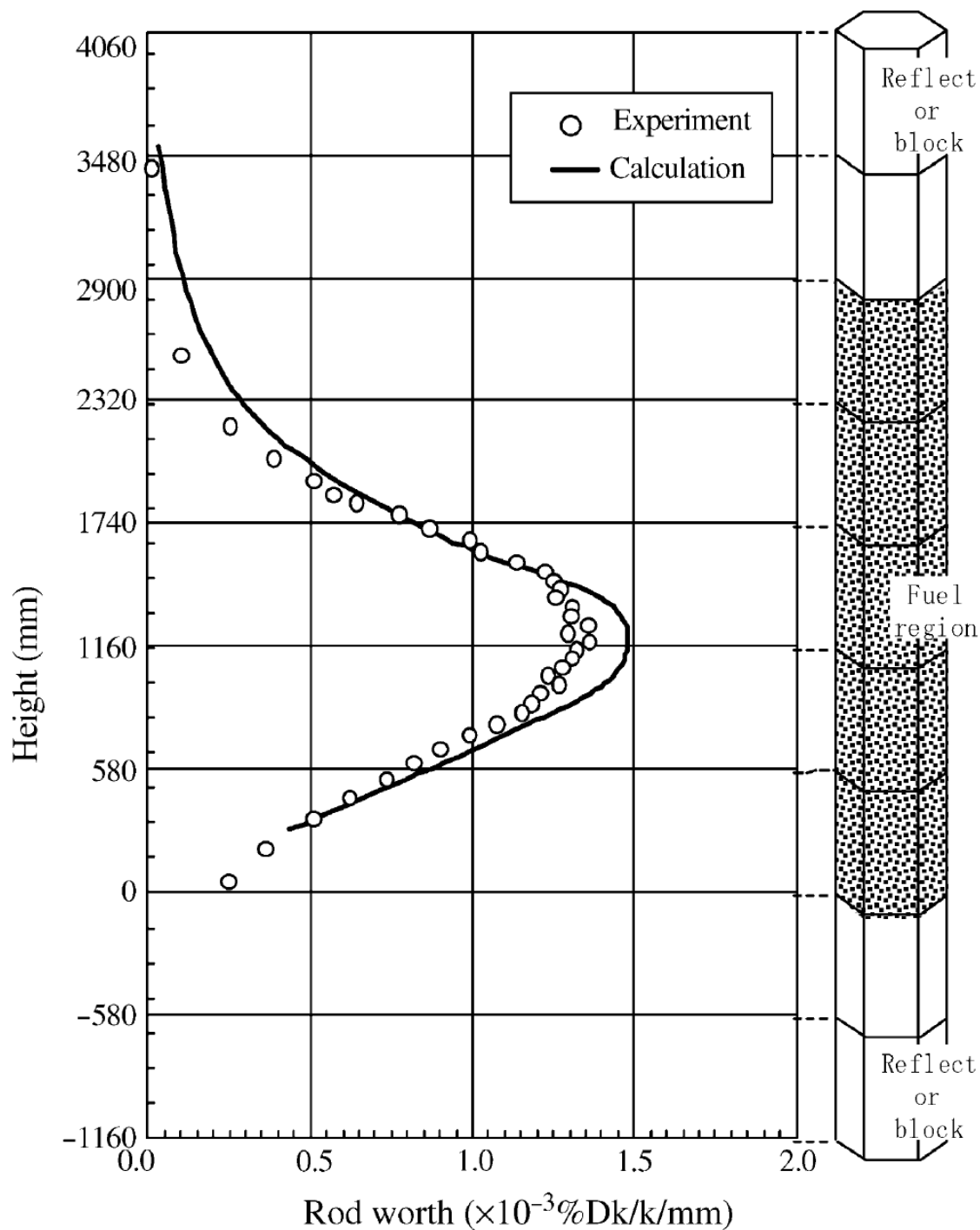


Figure 1.87. Differential Control Rod Worth for Zero Power Operation (Ref. 13, p. 286).



### 1.4.3 Material Data

See Section 1.1.3.

### 1.4.4 Temperature Data

See Section 1.1.4. Experiments were essentially performed at room temperature.

### 1.4.5 Additional Information Relevant to Reactivity Effects Measurements

Additional information is not available.

## 1.5 Description of Reactivity Coefficient Measurements

### 1.5.1 Overview of Measurement

See Section 1.1.1.

Experimental information gathered from the references is provided in this section, but not currently evaluated.

### 1.5.2 Geometry of the Experiment Configuration and Measurement Procedure

See Section 1.1.2.

#### 1.5.2.1 Isothermal Temperature Coefficient

Reference 2 contains calculations regarding evaluation of the isothermal temperature coefficients in the HTTR but does not actually represent experimental data. Reference 13 contains information pertinent to the actual determination of the isothermal temperature coefficient in the HTTR.

The reactivity coefficient of the HTTR is dominated by the temperature coefficient of the fuel and reflector. The temperature coefficient was determined by the reactivity difference incurred by changes in the core temperature. The reactor power was maintained constant and the coolant temperature was changed by adjusting the heat removal rate from the secondary coolant system. The reactivity was then estimated from the control rod worth curve. Figure 1.88 shows the results for calculating the isothermal temperature coefficient. The absolute value of the coefficient decreases as the average core temperature increases; however, all values remain negative. Previous calculation results show a discrepancy between calculated and actual coefficient values for temperatures below 400 °C ([Ref. 13, pp. 287-288](#))

The isothermal temperature coefficients of the fully-loaded core can be evaluated from the effective multiplication factors, using the following relationship:

$$\rho_n = \frac{k_{n+1} - k_n}{k_{n+1} \cdot k_n} \cdot \frac{1}{(T_{n+1} - T_n)},$$

where  $\rho_n$  is the temperature coefficient between  $T_n$  and  $T_{n+1}$  ( $\Delta k/k$ )/K,  $T_n$  is the core temperature of the  $n^{\text{th}}$  measurement (K),  $T_{n+1}$  is the core temperature of the  $(n+1)^{\text{th}}$  measurement (K),  $k_n$  is the effective multiplication factor at  $T_n$ , and  $k_{n+1}$  is the effective multiplication factor  $T_{n+1}$ . The critical control rod positions were changed with temperature elevation during reactor operation but typically not during the calculation of the reactivity difference.

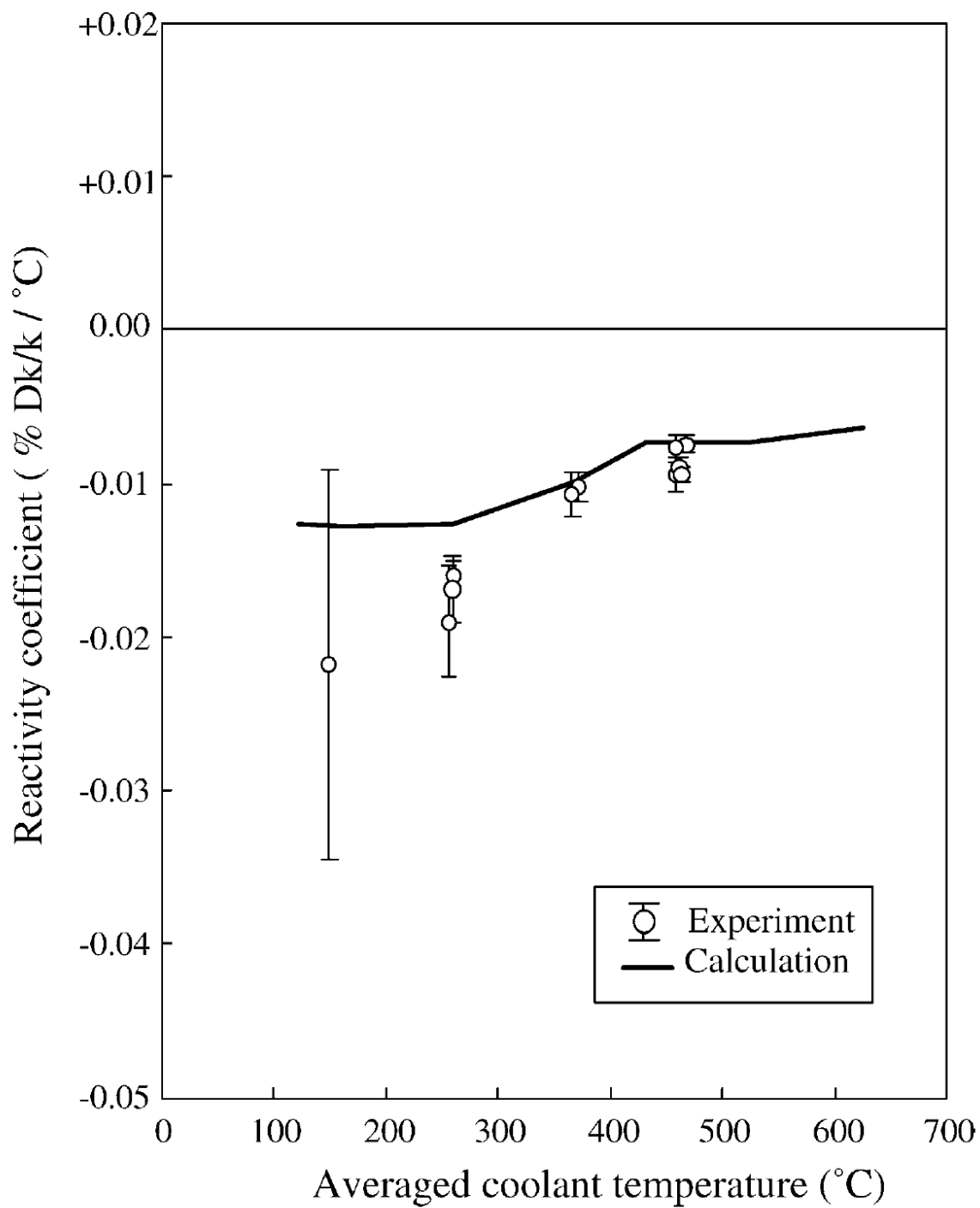


Figure 1.88. Temperature Coefficients vs. Average Coolant Temperature ([Ref. 13, p. 288](#)).

### **1.5.3 Material Data**

See Section 1.1.3.

### **1.5.4 Temperature Data**

See Figure 1.88.

### **1.5.5 Additional Information Relevant to Reactivity Effects Measurements**

Additional information is not available.

## **1.6 Description of Kinetics Measurements**

### **1.6.1 Overview of Measurement**

See Section 1.1.1.

Experimental information gathered from the references is provided in this section, but not currently evaluated.

### **1.6.2 Geometry of the Experiment Configuration and Measurement Procedure**

See Section 1.1.2.

“Inherent to this type of reactor, the neutron generation time is long and was calculated to be:  $1.173 \pm 0.001$  ms. The prompt neutron decay constant is quite close to the decay of the fastest delayed neutrons, therefore no intermediate plateau can be recognized in the measured spectral function” (Ref. 2, p. 115).

### **1.6.3 Material Data**

See Section 1.1.3.

### **1.6.4 Temperature Data**

See Section 1.1.4.

### **1.6.5 Additional Information Relevant to Reactivity Effects Measurements**

Additional information is not available.

## **1.7 Description of Reaction-Rate Distribution Measurements**

### **1.7.1 Overview of Measurement**

See Section 1.1.1.

Experimental information gathered from the references is provided in this section, but not currently evaluated.

## 1.7.2 Geometry of the Experiment Configuration and Measurement Procedure

See Section 1.1.2.

### 1.7.2.1 Axial Reaction Rate Distribution

The reaction rate of  $^{235}\text{U}$  was regarded as a neutron flux distribution because the neutron flux distribution could not be directly measured in the HTTR core. The three micro FCs placed in the irradiation test columns were used to measure the axial reaction rate. The FCs are 5 cm long with a diameter of 0.6 cm. An FC was connected to the end of a long aluminum stick such that it could be axially traversed by moving the stick in the aluminum tube well inserted into the holes of the irradiation test columns. The temporary neutron source was withdrawn from the core during measurements. The reaction rates were then measured at several points around the peak of the reaction rate distribution to search for the peak. The reactor power was changed by the movement of the FC because reactivity was added by movement of the attached stick and cable, which caused a change in the reaction rates of the FCs. The reaction rate of the traversing FC was then normalized with the reaction rate of the FC fixed in another irradiation test column. The axial reaction rate distributions were measured for the 24- and 30-fuel-column cores. Both FS and F23 patterns were formed in the 24-fuel-column core. The experimental error of the neutron flux was  $\sim 0.2\%$ , and was evaluated from the number of FC pulse counts. Measured distributions are shown (compared with some of Japan's calculated results) in Figure 1.89 ([Ref. 1, pp. 317-318](#)).

Figure 1.90 shows the measured and calculated axial neutron flux distribution for a reactor power of 27 W. The control rod position at the measurement was about 1780 mm. From the comparison of experimental to calculation results, it was confirmed that the neutron flux in the reflector region could be accurately calculated for the provision of fundamental data in irradiation tests ([Ref. 13, pp. 288-289](#)).

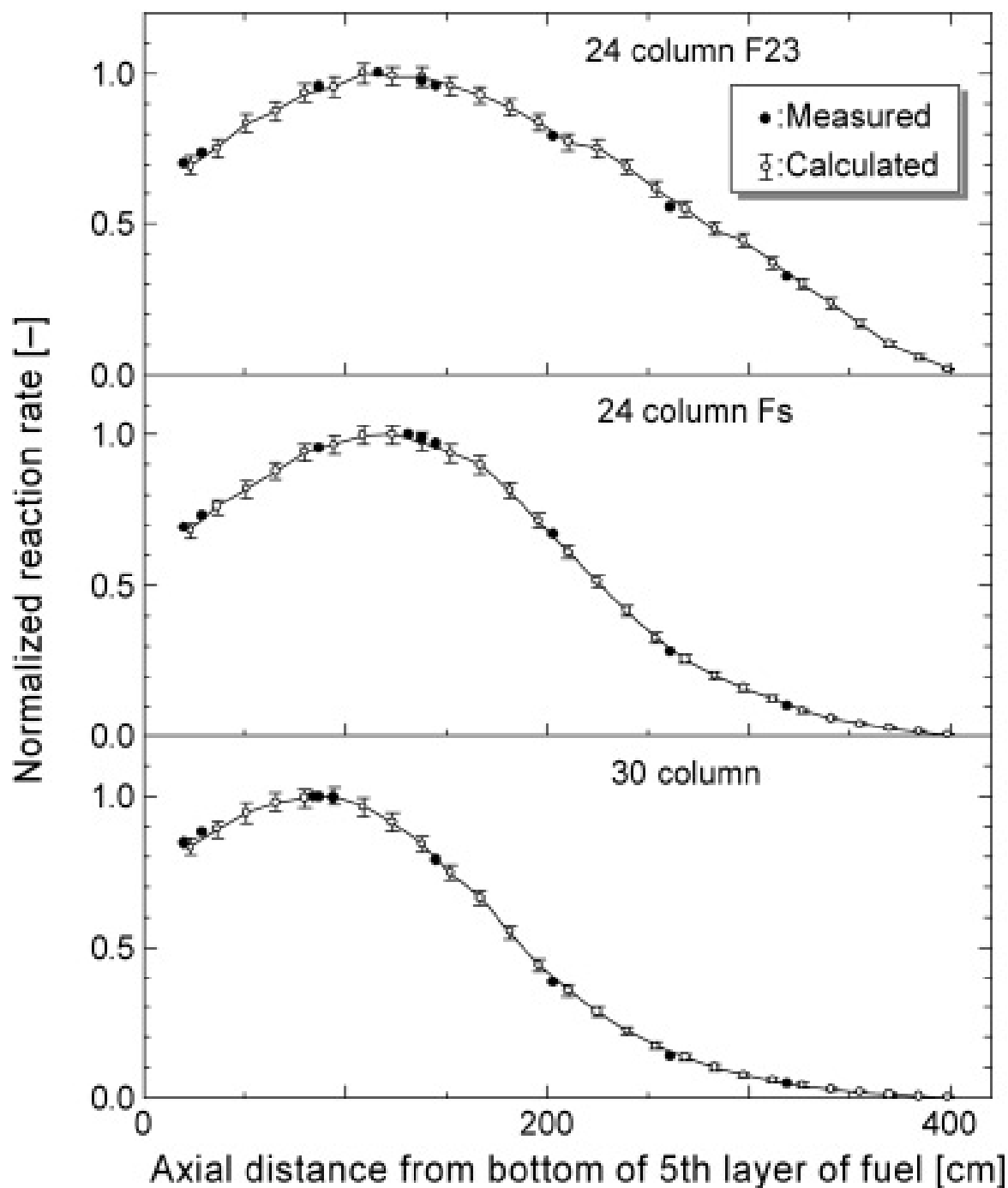


Figure 1.89. Axial Reaction Rate Distribution in Irradiation Column for 24- and 30-Fuel-Column Cores (Ref. 1, p. 318).

F23 and FS are defined in the footnote of Table 1.17.

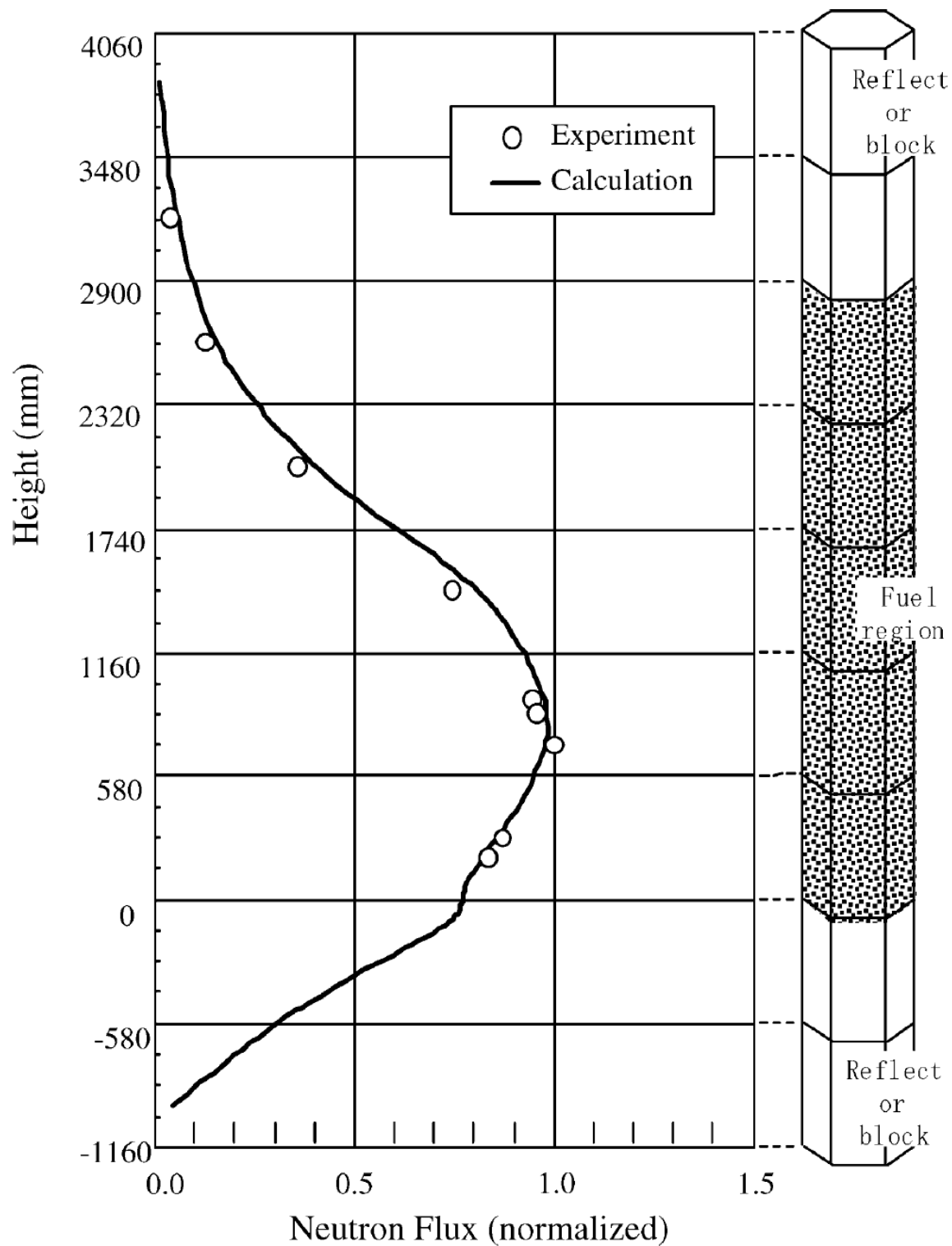


Figure 1.90. Axial Neutron Flux Distribution for Zero Power Operation (Ref. 13, p. 289).

### **1.7.3 Material Data**

See Section 1.1.3.

### **1.7.4 Temperature Data**

See Section 1.1.4. Experiments were essentially performed at room temperature.

### **1.7.5 Additional Information Relevant to Reactivity Effects Measurements**

Additional information is not available.

## **1.8 Description of Power Distribution Measurements**

Power distribution measurements were not made.

## **1.9 Description of Isotopic Measurements**

Isotopic measurements were not made.

## **1.10 Description of Other Miscellaneous Types of Measurements**

Other miscellaneous types of measurements were not made.



## 2.0 EVALUATION OF EXPERIMENTAL DATA

The overall uncertainty in the calculated value of  $k_{\text{eff}}$ , which is a function of multiple input parameters, is given by<sup>a</sup>

$$u_c^2(k_{\text{eff}}) = \sum_{i=1}^N (\Delta k_i)^2 + 2 \sum_{i=1}^{N-1} \sum_{j=i+1}^N (\Delta k_i)(\Delta k_j)r_{i,j}. \quad (2.1)$$

In Equation 2.1,  $\Delta k_i$  is the change in  $k_{\text{eff}}$  when parameter  $i$  is changed by the increment  $\sigma_i$ , the standard deviation in the parameter, and  $r_{i,j}$  is the correlation coefficient for parameters  $i$  and  $j$ . This expression for the uncertainty in  $k_{\text{eff}}$  can be evaluated accurately when the standard deviations in the parameters on which  $k_{\text{eff}}$  is dependent are available.

Where standard deviations are available, they are used for calculating the effects these uncertainties might have on  $k_{\text{eff}}$ , in terms of  $\Delta k_i$ . Where observed ranges are given, but not standard deviations, the limiting values of the observed ranges are usually applied, and plausible distribution functions are assumed for finding  $\Delta k_i$ . Where only tolerances are given, their limiting values are used, along with plausible distribution functions. Where no guidance is given on the variability of a parameter, engineering judgment is used to select a range of variation that will produce the largest reasonable uncertainty in  $k_{\text{eff}}$ . The bounding values in this range are then applied in the uncertainty analysis. If the overall uncertainty in  $k_{\text{eff}}$  predicted by this approach is small enough that the experiment can be judged an acceptable benchmark, one can be confident that the real experiment is actually even better. All uncertainties are adjusted to values of one standard deviation ( $1\sigma$ ). No information is available on correlations among parameters, so all parameters and their uncertainties are assumed to be uncorrelated.

Usually, no information is publicly available about the distribution function of the deviation of a parameter from its nominal value. In most cases, it is reasonably assumed that the most relevant quantity is uniformly distributed. For example, if the change in  $k_{\text{eff}}$  from its nominal value is dependent on the change in the volume of a spatial region, then it is assumed that the deviation of the volume of that region from its nominal value is uniformly distributed.

When a variable  $x$  is judged to be distributed uniformly between the bounding values  $\mu-a$  and  $\mu+a$ , the standard deviation in  $x$  can be shown to be  $a/\sqrt{3}$ .<sup>a</sup> Then the uncertainty in a linear function  $f(x)$  is taken as  $|f(\mu+a/\sqrt{3}) - f(\mu)|$ .

When a variable is judged to be distributed uniformly between the bounding values  $\mu$  and  $\mu+a$ , then the mean is actually  $\mu+a/2$ , and the standard deviation in  $x$  from the mean is  $a/(2\sqrt{3})$ . Then the uncertainty in a linear function  $f(x)$  is taken as  $|f(\mu + (a/2) + a/(2\sqrt{3})) - f(\mu+a/2)| = |f(\mu + a((1/2) + 1/(2\sqrt{3}))) - f(\mu+a/2)| = |(1/2 + 1/(2\sqrt{3}) - 1/2)(f(\mu+a) - f(\mu))| = |(f(\mu+a) - f(\mu))/(2\sqrt{3})|$ .

When a variable  $x$  is most likely to have a value  $c$  in the interval  $a < x < b$ , then the distribution function may be taken as triangular, and the variance in the triangular distribution is given by<sup>b</sup>

$$\sigma^2 = \frac{a^2 + b^2 + c^2 - ab - ac - bc}{18}, \quad (2.2)$$

and the uncertainty in a linear function  $f(x)$  in the interval is

<sup>a</sup> *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)03/I-VIII, OECD-NEA, "ICSBEF Guide to the Expression of Uncertainties," Revision 1, p. 29, September 30, 2004.

<sup>b</sup> M. Evans, Nicholas Hastings, and Brian Peacock, *Statistical Distributions*, 3rd Edition, John Wiley and Sons, New York, 2000, pp. 187-188.

$$|f(c \pm \sigma) - f(c)|, \quad (2.3)$$

where the choice of the plus or minus sign depends on the specific circumstances.

These observations are used repeatedly in the following analysis.

The following sections discuss the calculation of the effects of uncertainties in the parameters listed in tables in each section. The values of the tabulated parameters are computed in the benchmark critical configuration and in the configuration with each parameter assigned its maximum variation (or its standard deviation when available), one parameter at a time. The bases for the choices of the parameter values are discussed.

In all cases where tolerances or observed variations apply to large numbers of objects (pebbles and fuel particles or portions thereof), both deterministic (or systematic) uncertainties (applying to all the objects equally) and random uncertainties (different from one object to the next) will occur. For the fuel particles and their subregions especially, the random uncertainties are extremely small (the tolerance limit for the random uncertainty divided by the square root of the number of fuel particles in the core. In all cases, division by such large numbers would make the random component of the uncertainty negligible. Positional dependence of objects within the assembly also influences the effective proportional effect on the resultant uncertainty and bias calculations. However, since no information is available about how the uncertainties are divided between the systematic and random components, it is assumed throughout that the uncertainties are 25 % systematic and 75 % random.

This assumption provides a basic prediction of the effect on  $k_{\text{eff}}$  until additional information regarding systematic uncertainties can be better evaluated. The 25 % systematic uncertainty is bound by the fact that most systematic uncertainties would be below 50 % of the total uncertainty and above the historic approach of ignoring the unknown systematic components (i.e., treat it with a 0 % probability). In actuality, careful experimenters may have an unknown systematic uncertainty that is approximately 10-15 % of their total reported uncertainty. Evaluated uncertainties are listed as calculated, such that the reader may adjust results according to some desired systematic to random uncertainty ratio. The summary in Section 2.1.7 does list the systematic and random components of the uncertainty as separate entities, however.

It should be noted that assuming a uniform distribution of a parameter between its limits leads to overprediction of the effect on  $k_{\text{eff}}$ .

## 2.1 Evaluation of Critical and / or Subcritical Configuration Data

In the preliminary Japanese computational evaluation, the predicted number of fuel columns to achieve criticality was  $16 \pm 1$ , much less than the experimental result of 19. It was regarded that the neglect of nitrogen in the porous graphite led to this discrepancy. Two reevaluations were carried out after the initial criticality experiment (Ref. 1, pp. 313-314). The first reevaluation examined the air content in the graphite, simplification of graphite geometry, and impurity concentration in the dummy blocks. The predicted number of fuel columns for initial criticality was changed to  $18 \pm 1$ .<sup>a</sup> The second reevaluation wasn't performed until after the full power tests at 30 MW and 850 °C. Heterogeneity effects and air

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<sup>a</sup> Fujimoto, N., Nakano, M., Takeuchi, M., Fujisaki, S., and Yamashita, K., "Start-Up Core Physics Tests of High Temperature Engineering Test Reactor (HTTR), (II): First Criticality by an Annular Form Fuel Loading and Its Criticality Prediction Method," *J. Atomic Energy Society Japan*, **42**(5), 458-464 (2000).

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composition in the graphite was analyzed; the reevaluation also predicted  $18 \pm 1$  fuel columns for initial criticality.<sup>a</sup>

Monte Carlo n-Particle (MCNP) version 5.1.40 was utilized to predict the biases and uncertainties associated with the experimental results for HTTR critical configurations in this evaluation. MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled n-particle, Monte Carlo transport code.<sup>b</sup> The Evaluated Neutron Data File library, ENDF/B-VI.8<sup>c</sup>, was utilized in analysis of the benchmark model biases and uncertainties.

Elemental data such as molecular weights and isotopic abundances were taken from the 16<sup>th</sup> edition of the Chart of the Nuclides.<sup>d</sup> These values are summarized in Appendix C.

Only the primary components of the HTTR active core and surrounding reflector region were included in the analysis of uncertainties in this evaluation. Currently only the 30-fuel-column start-up core critical test was evaluated. The uncertainty analysis was performed using a model temperature of 20.5 °C.

For all impurity assessments, only the equivalent natural-boron content is utilized in the evaluation. In the compositions used in the evaluation models, the natural-boron content is adjusted to include only the primary absorber <sup>10</sup>B, according to the isotopic abundance of 19.9 at. %.

All MCNP calculations have statistical uncertainties between 0.00013 and 0.00014, resulting in  $\Delta k$  statistical uncertainties of approximately 0.00020, assuming no correlation between the individual MCNP results.

## 2.1.1 Experimental Measurements

### 2.1.1.1 Temperature

Various temperature coefficients of reactivity are reported, ranging from  $-1.23 \times 10^{-4} \Delta k/k-^{\circ}\text{C}$  (Table 1.17) to  $-14.2 \text{ pcm}/^{\circ}\text{C}$  (Ref. 2, p. 113 and 132). An average of  $-13.25 \text{ pcm}/^{\circ}\text{C}$  with a deviation of  $\pm 0.95$  ( $1\sigma$ ) was selected to represent the effective change in reactivity with temperature adjustment. The experiments were performed near room temperature, and an uncertainty of  $\pm 5^{\circ}\text{C}$  ( $1\sigma$ ) was selected to represent the uncertainty in the temperature of the experiment. Results are shown in Table 2.1.

The uncertainty in the temperature is considered all systematic with no random component.

Table 2.1. Effect of Uncertainty in Temperature.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-5°	0.00066	$\pm$	0.00005	1	0.00066	$\pm$	0.00005
+5°	-0.00066	$\pm$	0.00005	1	-0.00066	$\pm$	0.00005

<sup>a</sup> Yamashita, K., Fujimoto, N., Takeuchi, M., Fujisaki, S., Nakano, M., Umeda, M., Takeda, T., Mogi, H., and Tanaka, T., "Startup Core Physics Tests of High Temperature Engineering Test Reactor (HTTR), (I): Test Plan, Fuel Loading and Nuclear Characteristics Tests," *J. Atomic Energy Society Japan*, **42**(1), 30-42 (2000).

<sup>b</sup> X-5 Monte Carlo Team, "MCNP – a General Monte Carlo n-Particle Transport Code, version 5," LA-UR-03-1987, Los Alamos National Laboratory (2003).

<sup>c</sup> H. D. Lemmel, P. K. McLaughlin, and V. G. Pronyaev, "ENDF/B-VI Release 8 (Last Release of ENDF/B-VI) the U.S. Evaluated Nuclear Data Library for Neutron Reaction Data," IAEA-NDS-100 Rev. 11, International Atomic Energy Agency, Vienna (November 2001).

<sup>d</sup> Nuclides and Isotopes: Chart of the Nuclides, 16<sup>th</sup> edition, (2002).

### 2.1.1.2 Control Rod Position

Control rod position were varied  $\pm 5$  mm ( $1\sigma$ ), as shown in Table 1.18, to determine the effective change in  $k_{\text{eff}}$ . The sinking effect of -14 mm had already been applied to the rod position shown in Tables 1.17 and 1.18. If the unconfirmed temperature expansion effect of  $\pm 2$  mm (reported at the end of Section 1.1.2.4) is applied with the  $\pm 5$  mm in a root mean square approach, the overall uncertainty in the height remains approximately  $\pm 5$  mm. No additional bias or bias uncertainty was evaluated for the sinking effect of the control rods. Results are shown in Table 2.2.

The total number of control rods used in the core is 32. For determining the random component of the uncertainty in Section 2.1.7, the results in Table 2.2 are divided by  $\sqrt{32}$ .

Table 2.2. Effect of Uncertainty in Control Rod Position.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-5 mm ( $1\sigma$ )	-0.00053	$\pm$	0.00020	1	-0.00053	$\pm$	0.00020
+5 mm ( $1\sigma$ )	0.00065	$\pm$	0.00020	1	0.00065	$\pm$	0.00020

### 2.1.1.3 Measured Value of $k_{\text{eff}}$

There is no additional information regarding the accuracy of the  $k_{\text{eff}}$  measurements for the critical core conditions reported in Table 1.17.

## 2.1.2 Geometrical Properties

### 2.1.2.1 Coated Fuel Particles

#### Kernel Diameter

The kernel diameter was varied  $\pm 55$   $\mu\text{m}$  from the nominal value of 600  $\mu\text{m}$  (Tables 1.12 through 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . It is a bounding limit. All other thicknesses in the TRISO particle were maintained the same. The packing fraction (i.e., the ratio of the TRISO particle volume to the volume of a unit cell of fuel containing the TRISO particle) of the TRISO particles in the fuel was not conserved but remained within  $2\sigma$  of the nominal value of 30 vol. % ( $\pm 3$  % in Table 1.14). Results are shown in Table 2.3.

The uncertainty in the kernel diameter causes a large uncertainty in  $k_{\text{eff}}$ . It may likely that the uncertainty reported for the diameter  $3\sigma$  value. Figure 4 in the IAEA TECDOC-1210<sup>a</sup> displays standard deviations of fuel kernel diameters between 15 and 20  $\mu\text{m}$ , which corresponds with treatment of the  $\pm 55$   $\mu\text{m}$  as a  $3\sigma$  uncertainty with a  $1\sigma$  value of  $\pm 18.33$   $\mu\text{m}$ . Even reduced from a  $3\sigma$  to  $1\sigma$  uncertainty, the uncertainty in kernel diameter, which would provide uncertainty in the total uranium mass in the core, is still quite large before accounting for the statistical reduction of random particles.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.3 would be divided by  $\sqrt{861,140,000}$ .

<sup>a</sup> S. Kato, S. Yoshimuta, T. Hasumi, K. Sato, K. Sawa, S. Suzuki, H. Mogi, S. Shiozawa, and T. Tanaka, "Fabrication of HTTR First Loading Fuel," IAEA-TECDOC-1210, International Atomic Energy Agency, Vienna, pp. 187-199 (April 2001).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

Because of the overspecification (i.e., provision of four parameters of which only two can be independent) of the TRISO particles in Table 1.14 and the correlation of uranium kernel diameter, density, TRISO packing fraction and mass, the effect of the uncertainty in the kernel diameter is not included in the total uncertainty (See Section 2.1.6).

Table 2.3. Effect of Uncertainty in Kernel Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-55 $\mu\text{m}$	-0.04785	$\pm$	0.00020	$\sqrt{3}$	-0.02762	$\pm$	0.00011
+55 $\mu\text{m}$	0.03174	$\pm$	0.00019	$\sqrt{3}$	0.01833	$\pm$	0.00011

**Buffer Diameter**

The buffer thickness was varied  $\pm 12 \mu\text{m}$  ( $3\sigma$ ) from the nominal value of  $60 \mu\text{m}$  (Tables 1.12 and 1.14, and Figure 1.48) to determine the effective change in  $k_{\text{eff}}$ . All other thicknesses in the TRISO particle were maintained the same. The packing fraction of the TRISO particles in the fuel was not conserved, so as to conserve fuel mass, but remained within  $1\sigma$  of the nominal value of 30 vol. % ( $\pm 3\%$  in Table 1.14). Results are shown in Table 2.4.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.4 would be divided by  $\sqrt{861,140,000}$ .

Table 2.4. Effect of Uncertainty in Buffer Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-12 $\mu\text{m}$ ( $3\sigma$ )	0.00040	$\pm$	0.00019	3	0.00013	$\pm$	0.0.0006
+12 $\mu\text{m}$ ( $3\sigma$ )	-0.00002	$\pm$	0.00020	3	-0.00001	$\pm$	0.00007

**IPyC Diameter**

The IPyC thickness was varied  $\pm 6 \mu\text{m}$  ( $3\sigma$ ) from the nominal value of  $30 \mu\text{m}$  (Tables 1.12 and 1.14, and Figure 1.48) to determine the effective uncertainty in  $k_{\text{eff}}$ . All other thicknesses in the TRISO particle were maintained the same. The packing fraction of the TRISO particles in the fuel was not conserved but remained within  $1\sigma$  of the nominal value of 30 vol. % ( $\pm 3\%$  in Table 1.14). Results are shown in Table 2.5.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.5 would be divided by  $\sqrt{861,140,000}$ .

Table 2.5. Effect of Uncertainty in IPyC Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-6 $\mu\text{m}$ ( $3\sigma$ )	-0.00005	$\pm$	0.00019	3	-0.00002	$\pm$	0.00006
+6 $\mu\text{m}$ ( $3\sigma$ )	-0.00004	$\pm$	0.00019	3	-0.00001	$\pm$	0.00006

**SiC Diameter**

The SiC thickness was varied  $\pm 12 \mu\text{m}$  ( $3\sigma$ ) from the nominal value of  $30 \mu\text{m}$  (Tables 1.12 and 1.14, and Figure 1.48) to determine the effective uncertainty in  $k_{\text{eff}}$ . Although there is a negative component to the deviation in the thickness (Figure 1.48), TRISO particles with SiC thicknesses less than  $25 \mu\text{m}$  are not acceptable for use as a final product. The thickness of  $25 \mu\text{m}$  for the thickness appears to be a manufacturing limit while the thickness of  $30 \mu\text{m}$  is more likely to be the actual thickness of the SiC layer. All other thicknesses in the TRISO particle were maintained the same. The packing fraction of the TRISO particles in the fuel was not conserved but remained within  $1\sigma$  of the nominal value of 30 vol. % ( $\pm 3\%$  in Table 1.14). Results are shown in Table 2.6.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.6 would be divided by  $\sqrt{861,140,000}$ .

Table 2.6. Effect of Uncertainty in SiC Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-12 $\mu\text{m}$ ( $3\sigma$ )	0.00153	$\pm$	0.00020	3	0.00051	$\pm$	0.00007
+12 $\mu\text{m}$ ( $3\sigma$ )	-0.00112	$\pm$	0.00019	3	-0.00037	$\pm$	0.00006

**OPyC Diameter**

The OPyC thickness was varied  $\pm 6 \mu\text{m}$  ( $3\sigma$ ) from the nominal value of  $45 \mu\text{m}$  (Tables 1.12 and 1.14, and Figure 1.48) to determine the effective uncertainty in  $k_{\text{eff}}$ . All other thicknesses in the TRISO particle were maintained the same. The packing fraction of the TRISO particles in the fuel was not conserved but remained within  $1\sigma$  of the nominal value of 30 vol. % ( $\pm 3\%$  in Table 1.14). Results are shown in Table 2.7.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.7 would be divided by  $\sqrt{861,140,000}$ .

Table 2.7. Effect of Uncertainty in OPyC Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-6 $\mu\text{m}$ ( $3\sigma$ )	-0.00002	$\pm$	0.00020	3	-0.00001	$\pm$	0.00006
+6 $\mu\text{m}$ ( $3\sigma$ )	-0.00001	$\pm$	0.00020	3	0.00000	$\pm$	0.00007

**Overcoat Diameter**

Because insufficient data is available for the final composition and density of the graphite overcoat, this layer is being treated with equal properties to that of the surrounding compact graphite matrix (Table 1.14). Therefore modification of the overcoat diameter would not generate an effective uncertainty in  $k_{\text{eff}}$  beyond statistical uncertainty.

***2.1.2.2 Prismatic Fuel Compact*****Dimensions**

The inner and outer diameters (ID and OD) of a fuel compact were each individually varied  $\pm 0.1$  mm ( $1\sigma$ ) and the height (H) was varied  $\pm 0.5$  mm ( $1\sigma$ ) to determine the effective uncertainty in  $k_{\text{eff}}$ . For the effective change in height, the fuel stack height was adjusted by  $\pm 7.0$  mm and the effect from a height change in a single fuel compact was then determined by dividing by 14 for the number of compacts in a fuel rod. The nominal values for the inner diameter, outer diameter, and height of the fuel compacts are 10.0, 26.0, and 39.0 mm, respectively (Table 1.14). Results are shown in Table 2.8. In essence, changing the dimensions of the fuel compact without changing the number of TRISO particles would adjust the packing fraction.

The total number of fuel compacts used in the core is approximately 66,780. For determining the random component of the uncertainty, the results in Table 2.8 would be divided by  $\sqrt{66,780}$ .

Table 2.8. Effect of Uncertainty in Compact Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.1 mm ( $1\sigma$ ) ID	0.00025	$\pm$	0.00020	1	0.00025	$\pm$	0.00020
+0.1 mm ( $1\sigma$ ) ID	0.00028	$\pm$	0.00019	1	0.00028	$\pm$	0.00019
-0.1 mm ( $1\sigma$ ) OD	-0.00020	$\pm$	0.00020	1	-0.00020	$\pm$	0.00020
+0.1 mm ( $1\sigma$ ) OD	0.00058	$\pm$	0.00020	1	0.00058	$\pm$	0.00020
-7.0 mm ( $14 \times 1\sigma$ ) H	-0.00002	$\pm$	0.00020	14	0.00000	$\pm$	0.00001
+7.0 mm ( $14 \times 1\sigma$ ) H	0.00041	$\pm$	0.00020	14	0.00003	$\pm$	0.00001



**Packing Fraction**

The packing fraction was varied  $\pm 3$  vol. % from the nominal value of 30 % (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . It is a bounding limit. Results are shown in Table 2.9. This adjustment was performed by varying the number of TRISO particles packed into a single compact. Fuel mass is not conserved when adjusting the packing fraction, providing correlation between uncertainties in the fuel mass and packing fraction.

The total number of fuel compacts used in the core is approximately 66,780. For determining the random component of the uncertainty, the results in Table 2.8 would be divided by  $\sqrt{66,780}$ .

Because of the overspecification of the TRISO particles in Table 1.14 and the correlation of uranium kernel diameter, density, TRISO packing fraction and mass, the effect of the uncertainty in the packing fraction is not included in the total uncertainty (See Section 2.1.6).

Table 2.9. Effect of Uncertainty in Compact Packing Fraction.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-3 vol. % (+TRISO content)	0.01301	$\pm$	0.00019	$\sqrt{3}$	0.00751	$\pm$	0.00011
+3 vol. % (-TRISO content)	-0.01580	$\pm$	0.00020	$\sqrt{3}$	-0.00912	$\pm$	0.00011

**2.1.2.3 Graphite Sleeves**

The uncertainty in the sleeve thickness was unreported. An inner diameter uncertainty of  $\pm 0.5$  mm (bounding limit), which is limited by the 0.25 mm gap width between the sleeve and fuel compacts (Table 1.13), and an outer diameter uncertainty of  $\pm 2$  mm (bounding limit) were assumed; their effects on the uncertainty of  $k_{\text{eff}}$  were determined. Figure 1.51 shows an inner diameter of 26.25 mm for the graphite sleeves, which would only provide a gap space of 0.125 mm. The assumed uncertainty encompasses the discrepancy in this value. Results are shown in Table 2.10.

The fuel sleeve height was varied  $\pm 0.5$  mm ( $1\sigma$ ) from the nominal value of 577 mm (Table 1.14 and Figure 1.51) and the effect on the uncertainty of  $k_{\text{eff}}$  was determined. Results are shown in Table 2.10.

The total number of graphite sleeves used in the core is approximately 4,770. For determining the random component of the uncertainty, the results in Table 2.8 would be divided by  $\sqrt{4,770}$ .

Table 2.10. Effect of Uncertainty in Graphite Sleeve Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.5 mm ID	0.00150	$\pm$	0.00019	$\sqrt{3}$	0.00087	$\pm$	0.00011
+0.5 mm ID	-0.00157	$\pm$	0.00019	$\sqrt{3}$	-0.00091	$\pm$	0.00011
-2 mm OD	-0.00833	$\pm$	0.00019	$\sqrt{3}$	-0.00481	$\pm$	0.00011
+2 mm OD	0.00857	$\pm$	0.00019	$\sqrt{3}$	0.00495	$\pm$	0.00011
-0.5 mm ( $1\sigma$ ) H	0.00040	$\pm$	0.00019	1	0.00040	$\pm$	0.00019
-0.5 mm ( $1\sigma$ ) H	0.00015	$\pm$	0.00020	1	0.00015	$\pm$	0.00020

#### 2.1.2.4 Burnable Poisons

The uncertainty in the diameters of the BPs and BP insertion holes was unreported. A BP diameter uncertainty of  $\pm 1$  mm (bounding limit) from the nominal diameter (D) of 14 mm and a BP insertion hole diameter uncertainty of  $\pm 1$  mm (bounding limit) from the nominal diameter of 15 mm were assumed and their effects on the uncertainty of  $k_{\text{eff}}$  were determined. Results are shown in Table 2.11.

The uncertainty in the height stack of the BPs was unreported. A stack height (H) uncertainty of  $\pm 1$  mm ( $1\sigma$ ) from the nominal height of 200 mm was assumed (approximately  $\pm 0.1$  mm per BP pellet) and the effective uncertainty in  $k_{\text{eff}}$  was determined. The height of the BP pin hole was adjusted to accommodate the change in BP stack height. Results are shown in Table 2.11.

The uncertainty in the dimensions of the graphite disks was unreported. A diameter uncertainty of  $\pm 1$  mm (bounding limit) from the nominal diameter of 14 mm and a height uncertainty of  $\pm 1$  mm ( $1\sigma$ ) from the nominal stack height of 100 mm were assumed and their effects on the uncertainty of  $k_{\text{eff}}$  were determined. The height of the BP pin hole was adjusted to accommodate the change in graphite-disk stack height. Results are shown in Table 2.11.

The total number of burnable poison pellets used in the core is approximately 171,720. For determining the random component of the uncertainty, the results in Table 2.11 would be divided by  $\sqrt{171,720}$ . The total number of BP stacks is 19,080, of pin holes is 28,620, and of graphite disk stacks is 19,080.

Table 2.11. Effect of Uncertainty in BP Pin Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm BP D	0.01362	$\pm$	0.00020	$\sqrt{3}$	0.00786	$\pm$	0.00011
+1 mm BP D	-0.01271	$\pm$	0.00020	$\sqrt{3}$	-0.00734	$\pm$	0.00011
-1 mm ( $1\sigma$ ) BP Stack H	0.00076	$\pm$	0.00020	1	0.00076	$\pm$	0.00020
+1 mm ( $1\sigma$ ) BP Stack H	-0.00064	$\pm$	0.00019	1	-0.00064	$\pm$	0.00019
-1 mm Hole D	0.00108	$\pm$	0.00020	$\sqrt{3}$	0.00062	$\pm$	0.00011
+1 mm Hole D	-0.00101	$\pm$	0.00020	$\sqrt{3}$	-0.00058	$\pm$	0.00011
-1 mm Disk D	-0.00014	$\pm$	0.00020	$\sqrt{3}$	-0.00008	$\pm$	0.00011
+1 mm Disk D	0.00003	$\pm$	0.00020	$\sqrt{3}$	0.00002	$\pm$	0.00011
-1 mm ( $1\sigma$ ) Disk H	0.00002	$\pm$	0.00020	1	0.00002	$\pm$	0.00020
+1 mm ( $1\sigma$ ) Disk H	-0.00009	$\pm$	0.00020	1	-0.00009	$\pm$	0.00020

### 2.1.2.5 Control Rods

#### **Absorber Dimensions**

The uncertainty in the absorber dimensions was unreported. Assumed inner and outer diameter (ID and OD) uncertainties are  $\pm 5$  mm (bounding limit) each from the nominal values of 75 and 105 mm (Table 1.15), respectively, and their effects on the uncertainty of  $k_{\text{eff}}$  were determined. The uncertainty in the pellet height was assumed to be  $\pm 1$  cm (bounding limit) from the nominal stack height (H) of 29 cm. The alternate value of 115 mm provided for the outer diameter of the absorber material is believed to be a typographical error (see footnote of Table 1.15). Results are shown in Table 2.12.

The total number of control rod absorber pellets used in the core is approximately 1,300. The assumption is based on that only 13 control positions are approximately 50% inserted into the core. For determining the random component of the uncertainty, the results in Table 2.12 would be divided by  $\sqrt{1,300}$ .

Table 2.12. Effect of Uncertainty in CR Absorber Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-5 mm Absorber ID	-0.00020	$\pm$	0.00020	$\sqrt{3}$	-0.00012	$\pm$	0.00011
+5 mm Absorber ID	0.00036	$\pm$	0.00019	$\sqrt{3}$	0.00021	$\pm$	0.00011
-5 mm Absorber OD	0.00175	$\pm$	0.00020	$\sqrt{3}$	0.00101	$\pm$	0.00011
+5 mm Absorber OD	-0.00171	$\pm$	0.00020	$\sqrt{3}$	-0.00099	$\pm$	0.00011
-1 mm Absorber H	0.00086	$\pm$	0.00020	$\sqrt{3}$	0.00050	$\pm$	0.00011
+1 mm Absorber H	-0.00084	$\pm$	0.00020	$\sqrt{3}$	-0.00048	$\pm$	0.00011

#### **Clad Dimensions**

The uncertainty in the clad dimensions was unreported, and detailed dimensions for anything similar to the diagram shown in Figure 1.60 were unavailable. Therefore a solid clad material encasing the control rod absorber was defined without detail for the end caps. Inner and outer thicknesses were varied  $\pm 5$  mm (bounding limit) from the nominal values of 65 and 113 mm (Table 1.15), respectively, to determine their effects on the uncertainty of  $k_{\text{eff}}$ . The spine diameter was varied  $\pm 10$  mm (bounding limit) from the nominal value of 10 mm (Table 1.15) and its effect on the uncertainty of  $k_{\text{eff}}$  was determined. The clad height of a single control rod section was varied  $\pm 1$  cm (bounding limit). The reported clad thickness of 3.5 mm does not appear to conform to the difference between reported diameters of the clad and absorber materials. It is unclear whether a gap exists or there is rounding of values in the original table. The uncertainties evaluated encompass the overall uncertainty in this discrepancy. The control rod diameters are chosen as the more appropriate dimensions for modeling, such that inner and outer clad thicknesses are approximately 5 and 4 mm, respectively.. Results are shown in Table 2.13.

The total number of control rod sections used in the core is approximately 130. The assumption is based on that only 13 control positions are approximately 50% inserted into the core. For determining the random component of the uncertainty, the results in Table 2.13 would be divided by  $\sqrt{130}$ .

Table 2.13. Effect of Uncertainty in CR Clad Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-5 mm Clad ID	-0.00002	$\pm$	0.00020	$\sqrt{3}$	-0.00001	$\pm$	0.00011
+5 mm Clad ID	-0.00019	$\pm$	0.00020	$\sqrt{3}$	-0.00011	$\pm$	0.00011
-5 mm Clad OD	-0.00110	$\pm$	0.00020	$\sqrt{3}$	-0.00064	$\pm$	0.00011
+5 mm Clad OD	0.00089	$\pm$	0.00019	$\sqrt{3}$	0.00051	$\pm$	0.00011
-1 mm Clad H	0.00082	$\pm$	0.00019	$\sqrt{3}$	0.00047	$\pm$	0.00011
+1 mm Clad H	-0.00102	$\pm$	0.00020	$\sqrt{3}$	-0.00059	$\pm$	0.00011
-10 mm Spine D	-0.00007	$\pm$	0.00020	$\sqrt{3}$	-0.00004	$\pm$	0.00011
+10 mm Spine D	0.00005	$\pm$	0.00020	$\sqrt{3}$	0.00003	$\pm$	0.00011

#### 2.1.2.6 Instrumentation

Insufficient information is available to model and evaluate the uncertainties and biases related to the utility of instrumentation in the HTTR. Neglect of instrumentation inclusion in the model would be a bias; uncertainty in the dimensions and composition of the instrumentation would provide uncertainty in that bias or uncertainty in the model should it have been included in the benchmark model.

#### 2.1.2.7 Graphite Blocks

##### **Block Dimensions**

An uncertainty in the block dimensions was unreported. The uncertainty in the flat-to-flat distance and the height were each varied  $\pm 1$  mm (bounding limit) from their nominal values of 360 and 580 cm (Tables 1.13 and 1.16), respectively, and their effects on the uncertainty of  $k_{\text{eff}}$  were determined. The average gap between columns was reported as approximately 2 mm (Ref. 2, p. 13), allowing room for the flat-to-flat uncertainty. Results are shown in Table 2.14.

The total number of graphite blocks used in the core is approximately 549. For determining the random component of the uncertainty, the results in Table 2.14 would be divided by  $\sqrt{549}$ .

Table 2.14. Effect of Uncertainty in Graphite Block Dimensions.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm Flat-to-Flat Distance	-0.00277	$\pm$	0.00019	$\sqrt{3}$	-0.00160	$\pm$	0.00011
+1 mm Flat-to-Flat Distance	0.00252	$\pm$	0.00020	$\sqrt{3}$	0.00145	$\pm$	0.00011
-1 mm Block H	0.00097	$\pm$	0.00020	$\sqrt{3}$	0.00056	$\pm$	0.00011
+1 mm Block H	-0.00183	$\pm$	0.00020	$\sqrt{3}$	-0.00106	$\pm$	0.00011

**Dowel/Socket Dimensions**

Insufficient information is available to completely model and evaluate the uncertainties and biases related to the incorporation of dowels and sockets in the HTTR. Uncertainty in the volume fraction will be included as part of the assessment of the uncertainty in the total density of the graphite blocks.

**Coolant Channel Diameter**

An assumed variation of  $\pm 1$  mm (bounding limit) in the diameter of the coolant channels of the fuel blocks (nominally 41 mm, Table 1.13) and reflector blocks (nominally 23 mm) in the fuel columns was performed to determine the effective uncertainty in  $k_{\text{eff}}$ . Results are shown in Table 2.15.

Insufficient information was available to determine the dimensions of the coolant channels of the lowest reflector blocks. They were modeled similar to the other reflector blocks utilized in the fuel columns. No bias or biased uncertainty was assessed.

The total number of fuel and reflector coolant channels used in the core are approximately 4,770 and 3,816, respectively. For determining the random component of the uncertainty, the results in Table 2.15 would be divided by  $\sqrt{4,770}$  and  $\sqrt{3,816}$ , respectively.

Table 2.15. Effect of Uncertainty in Coolant Channel Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm Fueled Channel D	0.00488	$\pm$	0.00019	$\sqrt{3}$	0.00282	$\pm$	0.00011
+1 mm Fueled Channel D	-0.00499	$\pm$	0.00019	$\sqrt{3}$	-0.00288	$\pm$	0.00011
-1 mm Reflector Channel D	0.00000	$\pm$	0.00020	$\sqrt{3}$	0.00000	$\pm$	0.00011
+1 mm Reflector Channel D	-0.00002	$\pm$	0.00020	$\sqrt{3}$	-0.00001	$\pm$	0.00011

**Fuel and Coolant Channel Pitch**

An uncertainty in the fuel and coolant channel pitches in the fuel columns was not reported. A variation of  $\pm 0.1$  mm ( $1\sigma$ ) from the nominal pitch of 51.5 mm (Figures 1.52 and 1.53) was assumed and the effects on the uncertainty of  $k_{\text{eff}}$  were determined. Results are shown in Table 2.16.

The total number of pitch positions used in the core is approximately 8,586. For determining the random component of the uncertainty, the results in Table 2.16 would be divided by  $\sqrt{8,586}$ .

Table 2.16. Effect of Uncertainty in Fuel and Coolant Channel Pitch.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm ( $10\sigma$ )	-0.00154	$\pm$	0.00020	10	-0.00015	$\pm$	0.00002
+1 mm ( $10\sigma$ )	0.00153	$\pm$	0.00020	10	0.00015	$\pm$	0.00002

**Handling Socket Dimensions**

The handling sockets were not included in the model. The calculated volume of the socket (based upon dimensions in Figure 1.52) is roughly 0.5 vol. % of the complete block envelope. A bias has not been assessed. Uncertainty in the volume fraction will be included as part of the assessment of the uncertainty in the total density of the graphite blocks.

**Column Pitch**

An uncertainty in column pitch was assumed based upon the average distance between blocks of approximately 2 mm. A variation of  $\pm 1$  mm ( $1\sigma$ ) from the nominal value of 362 mm was analyzed and the effects on the uncertainty of  $k_{\text{eff}}$  were determined. Results are shown in Table 2.17.

The total number of columns used in the core is approximately 61. For determining the random component of the uncertainty, the results in Table 2.17 would be divided by  $\sqrt{61}$ .

Table 2.17. Effect of Uncertainty in Column Pitch.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm ( $1\sigma$ )	0.00225	$\pm$	0.00020	1	0.00225	$\pm$	0.00020
+1 mm ( $1\sigma$ )	-0.00265	$\pm$	0.00020	1	-0.00265	$\pm$	0.00020

**Control-Rod Channel Diameter**

An assumed variation of  $\pm 1$  mm ( $1\sigma$ ) in the diameter of the control-rod coolant channels (nominal value of 123 mm, Figure 1.64) in the control block columns was performed to determine the effective uncertainty in  $k_{\text{eff}}$ . Results are shown in Table 2.18.

The total number of control-rod channels (including empty instrumentation channels) used in the core is approximately 57. For determining the random component of the uncertainty, the results in Table 2.18 would be divided by  $\sqrt{57}$ .

Table 2.18. Effect of Uncertainty in CR Channel Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm ( $1\sigma$ )	0.00052	$\pm$	0.00020	1	0.00052	$\pm$	0.00020
+1 mm ( $1\sigma$ )	-0.00056	$\pm$	0.00020	1	-0.00056	$\pm$	0.00020

**Control-Rod Channel Pitch**

An uncertainty in control-rod coolant channel pitch in the fuel and reflector blocks was not reported. A variation of  $\pm 1$  mm ( $1\sigma$ ) from the nominal distance of 108 mm from the block axis (Figure 1.64) was assumed and the effects on the uncertainty of  $k_{\text{eff}}$  were determined. Results are shown in Table 2.19.

The total number of control-rod channels (including empty instrumentation channels) used in the core is approximately 57. For determining the random component of the uncertainty, the results in Table 2.19 would be divided by  $\sqrt{57}$ .

Table 2.19. Effect of Uncertainty in Control-Rod Channel Pitch.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-1 mm ( $1\sigma$ )	0.00023	$\pm$	0.00020	1	0.00023	$\pm$	0.00020
+1 mm ( $1\sigma$ )	-0.00009	$\pm$	0.00020	1	-0.00009	$\pm$	0.00020

**2.1.2.8 Permanent Reflectors**

Insufficient information is available to model in detail the permanent reflector of the HTTR. A bias was not assessed for any simplification of the permanent reflector.

A radial representation of the permanent reflector had the outer diameter varied  $\pm 5$  cm ( $1\sigma$ ) to determine the effective change in  $\Delta k$ . The overall diameter and length of the permanent reflector is 4250 mm and 5250 mm, respectively. Results are shown in Table 2.20.

The permanent reflector is comprised of 12 circumferential segments in eight axial layers for a total of 96 blocks. However, the uncertainty in the diameter of the model's permanent reflector is not adjusted for random uncertainty and treated as 100% systematic because of the uncertainty in the overall detail of the permanent reflector.

Table 2.20. Effect of Uncertainty in Permanent Reflector Diameter.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-5 cm ( $1\sigma$ )	-0.00028	$\pm$	0.00020	1	-0.00028	$\pm$	0.00020
+5 cm ( $1\sigma$ )	0.00061	$\pm$	0.00020	1	0.00061	$\pm$	0.00020



#### 2.1.2.9 Dummy Blocks

Insufficient information is available to model and evaluate the uncertainties and biases related to the utility of dummy blocks in the HTTR. The 30-fuel-column core, however, does not contain dummy blocks. Additional information would be necessary to evaluate the annular HTTR core criticals.

### 2.1.3 Compositional Variations

#### 2.1.3.1 Coated Fuel Particles

##### Uranium Enrichment

The concentration of  $^{234}\text{U}$  expected in the TRISO fuel had to be determined, as it was not provided. First the weight fractions of isotopes in natural uranium dioxide were determined. Then the enriched weight percent of  $^{235}\text{U}$  was multiplied by the natural weight percent of  $^{234}\text{U}$  (0.0055 at.%) and divided by the natural weight percent of  $^{235}\text{U}$  (0.72 at.%). Thus an approximate concentration of “enriched”  $^{234}\text{U}$  content could be determined for this evaluation.

Information was not provided regarding the uncertainty in the uranium enrichment of the TRISO kernels. Each of the twelve enrichments were varied an assumed  $\pm 0.1$  wt.% in  $^{235}\text{U}$  ( $1\sigma$ ), with the  $^{234}\text{U}$  content adjusted to match the effective increase or decrease in enrichment of  $^{235}\text{U}$ , to determine the effective uncertainty in  $k_{\text{eff}}$ . The nominal enrichment values are shown in Figure 1.46 and Table 1.11. Results are shown in Table 2.21.

The uncertainty in the uranium enrichment is considered all systematic with no random component.

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Table 2.21. Effect of Uncertainty in Uranium Enrichment.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.1 wt.% (1 $\sigma$ ) 3.4 wt.%	-0.00162	$\pm$	0.00020	1	-0.00162	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 3.4 wt.%	0.00161	$\pm$	0.00019	1	0.00161	$\pm$	0.00019
-0.1 wt.% (1 $\sigma$ ) 3.9 wt.%	-0.00096	$\pm$	0.00019	1	-0.00096	$\pm$	0.00019
+0.1 wt.% (1 $\sigma$ ) 3.9 wt.%	0.00151	$\pm$	0.00020	1	0.00151	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 4.3 wt.%	-0.00209	$\pm$	0.00019	1	-0.00209	$\pm$	0.00019
+0.1 wt.% (1 $\sigma$ ) 4.3 wt.%	0.00195	$\pm$	0.00020	1	0.00195	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 4.8 wt.%	-0.00050	$\pm$	0.00020	1	-0.00050	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 4.8 wt.%	0.00073	$\pm$	0.00020	1	0.00073	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 5.2 wt.%	-0.00040	$\pm$	0.00020	1	-0.00040	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 5.2 wt.%	0.00052	$\pm$	0.00020	1	0.00052	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 5.9 wt.%	-0.00067	$\pm$	0.00019	1	-0.00067	$\pm$	0.00019
+0.1 wt.% (1 $\sigma$ ) 5.9 wt.%	0.00042	$\pm$	0.00020	1	0.00042	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 6.3 wt.%	-0.00005	$\pm$	0.00020	1	-0.00005	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 6.3 wt.%	0.00029	$\pm$	0.00020	1	0.00029	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 6.7 wt.%	-0.00026	$\pm$	0.00019	1	-0.00026	$\pm$	0.00019
+0.1 wt.% (1 $\sigma$ ) 6.7 wt.%	0.00018	$\pm$	0.00020	1	0.00018	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 7.2 wt.%	0.00001	$\pm$	0.00020	1	0.00001	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 7.2 wt.%	0.00008	$\pm$	0.00019	1	0.00008	$\pm$	0.00019
-0.1 wt.% (1 $\sigma$ ) 7.9 wt.%	-0.00023	$\pm$	0.00020	1	-0.00023	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 7.9 wt.%	0.00015	$\pm$	0.00020	1	0.00015	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 9.4 wt.%	-0.00013	$\pm$	0.00020	1	-0.00013	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 9.4 wt.%	0.00028	$\pm$	0.00020	1	0.00028	$\pm$	0.00020
-0.1 wt.% (1 $\sigma$ ) 9.9 wt.%	-0.00018	$\pm$	0.00020	1	-0.00018	$\pm$	0.00020
+0.1 wt.% (1 $\sigma$ ) 9.9 wt.%	0.00017	$\pm$	0.00020	1	0.00017	$\pm$	0.00020

**Oxygen to Uranium Ratio**

The oxygen to uranium ratio was varied by  $\pm 0.02$  ( $1\sigma$ ) from the nominal value of 2.00 to determine the effective uncertainty in  $k_{\text{eff}}$ . Results are shown in Table 2.22.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.22 would be divided by  $\sqrt{861,140,000}$ .

Table 2.22. Effect of Uncertainty in Oxygen to Uranium Ratio.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.02 ( $1\sigma$ )	-0.00040	$\pm$	0.00019	1	-0.00040	$\pm$	0.00019
+0.02 ( $1\sigma$ )	0.00009	$\pm$	0.00020	1	0.00009	$\pm$	0.00020

**UO<sub>2</sub> Density**

The kernel density was varied  $\pm 0.26$  g/cm<sup>3</sup> from a selected value of 10.39 g/cm<sup>3</sup> (based on the average uranium mass per fuel rod) to determine the effective uncertainty in  $k_{\text{eff}}$ . It is a bounding limit. The listed UO<sub>2</sub> density is 10.41 or 10.63 g/cm<sup>3</sup> (Tables 1.13 or 1.14, respectively). However, to achieve the expected fuel content of 188.58 g per fuel rod, the selected density would need to be chosen, with the nominal kernel diameter of 600  $\mu\text{m}$ . The selected density for this evaluation is within the statistical uncertainty provided in Table 1.14. Results are shown in Table 2.23.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.23 would be divided by  $\sqrt{861,140,000}$ .

Because of the overspecification of the TRISO particles in Table 1.14 and the correlation of uranium kernel diameter, density, TRISO packing fraction, and mass, the effect of the uncertainty in the kernel density is not included in the total uncertainty (See Section 2.1.6).

Table 2.23. Effect of Uncertainty in UO<sub>2</sub> Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.26 g/cm <sup>3</sup>	-0.00374	$\pm$	0.00019	$\sqrt{3}$	-0.00216	$\pm$	0.00011
+0.26 g/cm <sup>3</sup>	0.00369	$\pm$	0.00020	$\sqrt{3}$	0.00213	$\pm$	0.00011

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The kernel impurity was varied from 0-3 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The maximum limit was multiplied 10-fold so as to quantify the effective upper uncertainty in the UO<sub>2</sub> impurity. The average value is 1.5 ppm by weight (Tables 1.13 and 1.14). Results are shown in Table 2.24.

The uncertainty in the UO<sub>2</sub> impurity is considered all systematic with no random component.

Table 2.24. Effect of Uncertainty in UO<sub>2</sub> Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00030	$\pm$	0.00019	$\sqrt{3}$	0.00017	$\pm$	0.00011
3 ppm	-0.00008	$\pm$	0.00020	$\sqrt{3}$	-0.00005	$\pm$	0.00011
30 ppm	-0.00030	$\pm$	0.00020	$10\sqrt{3}$	-0.00002	$\pm$	0.00001

**Buffer Density**

The buffer density was varied  $\pm 0.30 \text{ g/cm}^3$  from the nominal value of  $1.10 \text{ g/cm}^3$  (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is three times the bounding limit. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.25.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.25 would be divided by  $\sqrt{861,140,000}$ .

Table 2.25. Effect of Uncertainty in Buffer Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.30 g/cm <sup>3</sup>	0.00003	$\pm$	0.00020	$3\sqrt{3}$	-0.00001	$\pm$	0.00004
+0.30 g/cm <sup>3</sup>	-0.00008	$\pm$	0.00020	$3\sqrt{3}$	0.00002	$\pm$	0.00004
-0.10 g/cm <sup>3</sup> (PERT)	-0.00003	$\pm$	0.00002	$\sqrt{3}$	-0.00002	$\pm$	0.00001
+0.10 g/cm <sup>3</sup> (PERT)	0.00003	$\pm$	0.00002	$\sqrt{3}$	0.00002	$\pm$	0.00001

## Gas Cooled (Thermal) Reactor - GCR

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The buffer impurity was varied from 0-3 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The average value is 1.5 ppm by weight (Table 1.13). Results are shown in Table 2.26.

The uncertainty in the buffer impurity is considered all systematic with no random component.

Table 2.26. Effect of Uncertainty in Buffer Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00019	$\pm$	0.00019	$\sqrt{3}$	0.00011	$\pm$	0.00011
3 ppm	0.00019	$\pm$	0.00019	$\sqrt{3}$	0.00011	$\pm$	0.00011

**IPyC Density**

The IPyC density was varied +0.30 and -0.15 g/cm<sup>3</sup> from the nominal value of 1.85 g/cm<sup>3</sup> (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . These values are three times the bounding limit. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.27.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.27 would be divided by  $\sqrt{861,140,000}$ .

Table 2.27. Effect of Uncertainty in IPyC Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.15 g/cm <sup>3</sup>	-0.00013	$\pm$	0.00019	$3\sqrt{3}$	-0.00003	$\pm$	0.00004
+0.30 g/cm <sup>3</sup>	0.00001	$\pm$	0.00019	$3\sqrt{3}$	0.00000	$\pm$	0.00004
-0.05 g/cm <sup>3</sup> (PERT)	0.00002	$\pm$	0.00001	$\sqrt{3}$	0.00001	$\pm$	0.00001
+0.10 g/cm <sup>3</sup> (PERT)	-0.00001	$\pm$	0.00001	$\sqrt{3}$	-0.00001	$\pm$	0.00001

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**IPyC Impurity**

The IPyC impurity was varied from 0-3 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The average value is 1.5 ppm by weight (Table 1.13). Results are shown in Table 2.28.

The uncertainty in the IPyC impurity is considered all systematic with no random component.

Table 2.28. Effect of Uncertainty in IPyC Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00011	$\pm$	0.00020	$\sqrt{3}$	0.00006	$\pm$	0.00011
3 ppm	-0.00004	$\pm$	0.00020	$\sqrt{3}$	-0.00002	$\pm$	0.00011

**SiC Density**

The SiC density was increased by 0.06 g/cm<sup>3</sup>, which is typical for SiC material, from the nominal value of 3.20 g/cm<sup>3</sup> (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is three times the bounding limit. The density was not decreased because of the minimum requirement for SiC density. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.29.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.29 would be divided by  $\sqrt{861,140,000}$ .

Table 2.29. Effect of Uncertainty in SiC Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
+0.06 g/cm <sup>3</sup>	-0.00001	$\pm$	0.00019	$3\sqrt{3}$	-0.00000	$\pm$	0.00004
+0.02 g/cm <sup>3</sup> (PERT)	-0.00001	$\pm$	0.00000	$1\sqrt{3}$	-0.00001	$\pm$	0.00000

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**SiC Impurity**

The SiC impurity was varied from 0-3 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The average value is 1.5 ppm by weight (Table 1.13). Results are shown in Table 2.30.

The uncertainty in the SiC impurity is considered all systematic with no random component.

Table 2.30. Effect of Uncertainty in SiC Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00013	$\pm$	0.00020	$\sqrt{3}$	0.00008	$\pm$	0.00011
3 ppm	-0.00005	$\pm$	0.00019	$\sqrt{3}$	-0.00003	$\pm$	0.00011

**OPyC Density**

The OPyC density was varied +0.30 and -0.15 g/cm<sup>3</sup> (3 $\sigma$ ) from the nominal value of 1.85 g/cm<sup>3</sup> (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . These values are three times the bounding limit. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.31.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.31 would be divided by  $\sqrt{861,140,000}$ .

Table 2.31. Effect of Uncertainty in OPyC Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.15 g/cm <sup>3</sup>	-0.00028	$\pm$	0.00019	$3\sqrt{3}$	-0.00005	$\pm$	0.00004
+0.30 g/cm <sup>3</sup>	0.00021	$\pm$	0.00020	$3\sqrt{3}$	0.00004	$\pm$	0.00004
-0.05 g/cm <sup>3</sup> (PERT)	0.00004	$\pm$	0.00002	$\sqrt{3}$	0.00002	$\pm$	0.00001
+0.10 g/cm <sup>3</sup> (PERT)	-0.00002	$\pm$	0.00001	$\sqrt{3}$	-0.00001	$\pm$	0.00001



**OPyC Impurity**

The OPyC impurity was varied from 0-3 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The average value is 1.5 ppm by weight (Table 1.13). Results are shown in Table 2.32.

The uncertainty in the OPyC impurity is considered all systematic with no random component.

Table 2.32. Effect of Uncertainty in OPyC Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00000	$\pm$	0.00020	$\sqrt{3}$	0.00000	$\pm$	0.00011
3 ppm	0.00028	$\pm$	0.00020	$\sqrt{3}$	0.00016	$\pm$	0.00011

**Overcoat Density**

Because insufficient data is available for the final composition and density of the graphite overcoat, this layer is being treated with equal properties to that of the surrounding compact graphite matrix. The overcoat density was varied  $\pm 0.15 \text{ g/cm}^3$  to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is three times the bounding limit. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.33.

The total number of TRISO particles used in the core is approximately 861,140,000. For determining the random component of the uncertainty, the results in Table 2.33 would be divided by  $\sqrt{861,140,000}$ .

Table 2.33. Effect of Uncertainty in Overcoat Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.15 g/cm <sup>3</sup>	-0.00053	$\pm$	0.00019	$3\sqrt{3}$	-0.00010	$\pm$	0.00004
+0.15 g/cm <sup>3</sup>	0.00051	$\pm$	0.00019	$3\sqrt{3}$	-0.00010	$\pm$	0.00004
-0.05 g/cm <sup>3</sup> (PERT)	0.00013	$\pm$	0.00002	$\sqrt{3}$	0.00008	$\pm$	0.00001
+0.05 g/cm <sup>3</sup> (PERT)	-0.00013	$\pm$	0.00002	$\sqrt{3}$	-0.00008	$\pm$	0.00001

**Overcoat Composition**

Because insufficient data is available for the final composition and density of the graphite overcoat, this layer is being treated with equal properties to that of the surrounding compact graphite matrix. In essence, the uncertainty has already been accounted for, as the surrounding graphite matrix has been characterized for uncertainties, and the overcoat has been demonstrated elsewhere to have negligible impact on the calculation of  $k_{\text{eff}}$  when it is not explicitly modeled, but included in the surrounding matrix.<sup>a</sup>

<sup>a</sup> W. Ji, J. L. Conlin, W. R. Martin, J. C. Lee, and F. B. Brown, "Explicit Modeling of Particle Fuel for the Very-High Temperature Gas-Cooled Reactor," *Trans. Am. Nucl. Soc.*, **92** (June 2005).

**Overcoat Impurity**

The overcoat impurity was varied from 0-5 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The average value of the coated fuel particles is 1.5 ppm by weight (Table 1.13); it is assumed that the overcoat would have a comparable impurity amount. The alternative is to use the impurity of the compact (matrix) material, which is 0.82 ppm by weight (Table 1.27). The larger amount is selected, i.e. 1.5 ppm. Results are shown in Table 2.34.

The uncertainty in the overcoat impurity is considered all systematic with no random component.

Table 2.34. Effect of Uncertainty in Overcoat Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00066	$\pm$	0.00020	$\sqrt{3}$	0.00038	$\pm$	0.00011
5 ppm	-0.00153	$\pm$	0.00020	$\sqrt{3}$	-0.00088	$\pm$	0.00011

**2.1.3.2 Prismatic Fuel Compact****Density**

The compact matrix density was varied  $\pm 0.15 \text{ g/cm}^3$  from the nominal value of  $1.70 \text{ g/cm}^3$  (Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is three times the bounding limit. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.35.

The total number of fuel compacts used in the core is approximately 66,780. For determining the random component of the uncertainty, the results in Table 2.35 would be divided by  $\sqrt{66,780}$ .

Table 2.35. Effect of Uncertainty in Compact Matrix Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.15 g/cm <sup>3</sup>	-0.00023	$\pm$	0.00020	$3\sqrt{3}$	-0.00004	$\pm$	0.00004
+0.15 g/cm <sup>3</sup>	0.00028	$\pm$	0.00020	$3\sqrt{3}$	0.00005	$\pm$	0.00004
-0.05 g/cm <sup>3</sup> (PERT)	0.00005	$\pm$	0.00001	$\sqrt{3}$	0.00003	$\pm$	0.00001
+0.05 g/cm <sup>3</sup> (PERT)	-0.00005	$\pm$	0.00001	$\sqrt{3}$	-0.00003	$\pm$	0.00001

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**Impurity**

The compact matrix impurity was varied from 0-5 ppm by weight of equivalent natural-boron content (Table 1.14) to determine the bounding uncertainty in  $k_{\text{eff}}$ . The nominal impurity is 0.82 ppm of natural boron by weight (Table 1.27). Results are shown in Table 2.36.

The uncertainty in the fuel compact impurity is considered all systematic with no random component.

Table 2.36. Effect of Uncertainty in Compact Matrix Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00012	$\pm$	0.00019	$\sqrt{3}$	0.00007	$\pm$	0.00011
5 ppm	-0.00070	$\pm$	0.00019	$\sqrt{3}$	-0.00040	$\pm$	0.00011

**Free Uranium Content**

The free uranium fraction in the fuel compacts was varied between bounding limits of 0 and 0.00015 with a median of 0.000075 (see Table 1.14) to determine the effective uncertainty in  $k_{\text{eff}}$ . The bounding limit was tripled to 0.00045 in order to assess the uncertainty, but the effects were still negligible. The free uranium was assumed to be 100 %  $^{235}\text{U}$  so as to assess the maximum uncertainty. The free uranium content in the graphite compact was not included in the benchmark model and a bias was not assessed. Results are shown in Table 2.37.

The uncertainty in the free uranium content is considered all systematic with no random component.

Table 2.37. Effect of Uncertainty in Compact Free Uranium Content.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
75 ppm (median composition)	-0.00015	$\pm$	0.00020	1	NA	$\pm$	NA
150 ppm	0.00002	$\pm$	0.00020	$2\sqrt{3}$	0.00001	$\pm$	0.00006
450 ppm	-0.00014	$\pm$	0.00020	$6\sqrt{3}$	-0.00001	$\pm$	0.00002

### 2.1.3.3 Graphite Sleeves

#### **Density**

The graphite sleeve density was varied an assumed  $\pm 0.03 \text{ g/cm}^3$  from the nominal value of  $1.770 \text{ g/cm}^3$  (Table 1.27) to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is the bounding limit. The graphite sleeves are composed of IG-110 graphite, and the assumed uncertainty of  $0.03 \text{ g/cm}^3$  encompasses the range of reported densities for IG-110 graphite throughout Section 1. Results are shown in Table 2.38.

The total number of graphite sleeves used in the core is approximately 4,770. For determining the random component of the uncertainty, the results in Table 2.38 would be divided by  $\sqrt{4,770}$ .

Table 2.38. Effect of Uncertainty in Graphite Sleeve Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.03 g/cm <sup>3</sup>	-0.00043	$\pm$	0.00019	$\sqrt{3}$	-0.00025	$\pm$	0.00011
+0.03 g/cm <sup>3</sup>	0.00037	$\pm$	0.00020	$\sqrt{3}$	0.00021	$\pm$	0.00011

#### **Impurity**

The graphite sleeve impurity was varied from 0-1 ppm by weight of equivalent natural-boron content (Table 1.13) to determine the bounding uncertainty in  $k_{\text{eff}}$ . The nominal impurity is 0.37 ppm of natural boron by weight (Table 1.27). Results are shown in Table 2.39.

The uncertainty in the graphite sleeve impurity is considered all systematic with no random component.

Table 2.39. Effect of Uncertainty in Graphite Sleeve Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00058	$\pm$	0.00019	$\sqrt{3}$	0.00033	$\pm$	0.00011
1 ppm	-0.00036	$\pm$	0.00020	$\sqrt{3}$	-0.00021	$\pm$	0.00011

#### 2.1.3.4 Burnable Poisons

##### **Absorber Density**

The absorber density was varied  $\pm 0.03 \text{ g/cm}^3$  from the average value of  $1.8 \text{ g/cm}^3$  to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is the bounding limit. The density uncertainty is based upon uncertainty typically found in sintered  $\text{B}_4\text{C/C}$  pellets (Table 1.29). Results are shown in Table 2.40.

The total number of burnable poison pellets used in the core is approximately 171,720. For determining the random component of the uncertainty, the results in Table 2.40 would be divided by  $\sqrt{171,720}$ .

Table 2.40. Effect of Uncertainty in BP Absorber Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.03 g/cm <sup>3</sup>	0.00087	$\pm$	0.00020	$\sqrt{3}$	0.00050	$\pm$	0.00011
+0.03 g/cm <sup>3</sup>	-0.00096	$\pm$	0.00020	$\sqrt{3}$	-0.00055	$\pm$	0.00011

##### **Absorber Content**

The uncertainty in the absorber content was not provided and the variation provided in Table 1.29 appears too excessive for the quantities utilized in the BP pellets. A variation of approximately  $\pm 0.25 \%$  by weight was assumed for each of the two absorber pellet types, and the effective uncertainty in  $k_{\text{eff}}$  was determined. This value is treated as a bounding limit. The uncertainty of  $\pm 0.25 \%$  is based upon the assumption that burnable poison pellets with boron contents between 1.75 and 2.25 wt.% would have an average content of 2.00 wt.% and boron contents between 2.25 and 2.75 wt.% would have an average content of 2.50 wt.% bin. Further information would be necessary to reduce the range of the uncertainty. Results are shown in Table 2.41.

The total number of burnable poison pellets with weight percents of 2.00 and 2.50 used in the core are approximately 103,032 and 68,688, respectively. For determining the random component of the uncertainty, the results in Table 2.41 would be divided by  $\sqrt{103,032}$  and  $\sqrt{68,688}$ , respectively.

Table 2.41. Effect of Uncertainty in BP Absorber Content.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.25 wt.% (2.00 wt.%)	0.00605	$\pm$	0.00020	$\sqrt{3}$	0.00349	$\pm$	0.00011
+0.25 wt.% (2.00 wt.%)	-0.00499	$\pm$	0.00020	$\sqrt{3}$	-0.00288	$\pm$	0.00011
-0.25 wt.% (2.50 wt.%)	0.00183	$\pm$	0.00020	$\sqrt{3}$	0.00106	$\pm$	0.00011
+0.25 wt.% (2.50 wt.%)	-0.00152	$\pm$	0.00019	$\sqrt{3}$	-0.00088	$\pm$	0.00011

**Absorber Impurity**

No information was available regarding any impurities present in the sintered B<sub>4</sub>C/C material for the BP pellets. However, the impurity limits provided in Table 1.29 can be applied to approximate a rough estimate of the impact of additional impurities. Sodium and manganese were added with a concentrations of 100 and 10 ppm (by weight), respectively. Concentrations of 1000 ppm (by weight) of aluminum, silicon, calcium, and titanium were also included. The effective uncertainty in  $k_{\text{eff}}$  was determined and is shown in Table 2.42. This value is treated as a bounding limit.

The uncertainty in the absorber impurity is considered all systematic with no random component.

Table 2.42. Effect of Uncertainty in Absorber Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
Added Impurities	-0.00052	$\pm$	0.00020	$\sqrt{3}$	-0.00030	$\pm$	0.00011

**Absorber Isotopic Abundance**

According to the 16<sup>th</sup> edition of the Chart of the Nuclides, the natural isotopic abundance of <sup>10</sup>B has been measured between 19.1 and 20.3 at.% with a nominal value of 19.9 at.%.<sup>a</sup> The abundance of <sup>10</sup>B in the BPs was therefore evaluated at the minimum and maximum bounding values to determine the effective uncertainty in  $k_{\text{eff}}$ . Table 1.28 states that the abundance of <sup>10</sup>B is 18.7 wt.%, which correlates to approximately 20.2 at.%. Results are shown in Table 2.43. The benchmark model was evaluated at 19.9 at.%.

The uncertainty in the absorber isotopic abundance is considered all systematic with no random component.

Table 2.43. Effect of Uncertainty in BP Isotopic Abundance of <sup>10</sup>B.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
19.1%	0.00267	$\pm$	0.00019	$\sqrt{3}$	0.00154	$\pm$	0.00011
20.3%	-0.00093	$\pm$	0.00020	$\sqrt{3}$	-0.00054	$\pm$	0.00011

<sup>a</sup> Nuclides and Isotopes: Chart of the Nuclides, 16<sup>th</sup> edition, (2002).

**Graphite Disk Density**

An uncertainty in the density of the graphite disks was not reported. A variation of  $\pm 0.09 \text{ g/cm}^3$  was assumed and the effects on the uncertainty of  $k_{\text{eff}}$  were determined. This value is three times the bounding limit. The graphite disks are composed of IG-110 graphite, and the assumed uncertainty of  $0.03 \text{ g/cm}^3$  encompasses the range of reported densities for IG-110 graphite throughout Section 1. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.44.

The total number of graphite disks used in the core is approximately 19,080. For determining the random component of the uncertainty, the results in Table 2.44 would be divided by  $\sqrt{19,080}$ .

Table 2.44. Effect of Uncertainty in Graphite Disk Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.09 g/cm <sup>3</sup>	-0.00004	$\pm$	0.00019	$3\sqrt{3}$	-0.00001	$\pm$	0.00004
+0.09 g/cm <sup>3</sup>	-0.00004	$\pm$	0.00020	$3\sqrt{3}$	-0.00001	$\pm$	0.00004
-0.03 g/cm <sup>3</sup> (PERT)	0.00000	$\pm$	0.00000	$\sqrt{3}$	0.00000	$\pm$	0.00000
+0.03 g/cm <sup>3</sup> (PERT)	0.00000	$\pm$	0.00000	$\sqrt{3}$	0.00000	$\pm$	0.00000

**Graphite Disk Impurity**

The assumed graphite disk impurity was varied from 0-5 ppm by weight of equivalent natural-boron content to determine the bounding uncertainty in  $k_{\text{eff}}$ . The nominal impurity is 0.37 ppm of natural boron by weight (Table 1.28). Results are shown in Table 2.45.

The uncertainty in the graphite disk impurity is considered all systematic with no random component.

Table 2.45. Effect of Uncertainty in Graphite Disk Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00014	$\pm$	0.00019	$\sqrt{3}$	0.00008	$\pm$	0.00011
5 ppm	-0.00009	$\pm$	0.00020	$\sqrt{3}$	0.00005	$\pm$	0.00011

## 2.1.3.5 Control Rods

**Absorber Density**

The absorber density was varied  $\pm 0.09 \text{ g/cm}^3$  from the nominal value of  $1.9 \text{ g/cm}^3$  (Table 1.15) to determine the effective uncertainty in  $k_{\text{eff}}$ . This value is three times the bounding limit. The uncertainty is taken from Table 1.29. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7. Results are shown in Table 2.46.

The total number of control rod absorber pellets used in the core is approximately 1,300. The assumption is based on that only 13 control positions are approximately 50% inserted into the core. For determining the random component of the uncertainty, the results in Table 2.46 would be divided by  $\sqrt{1,300}$ .

Table 2.46. Effect of Uncertainty in CR Absorber Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.09 g/cm <sup>3</sup>	0.00028	$\pm$	0.00020	$3\sqrt{3}$	0.00005	$\pm$	0.00004
+0.09 g/cm <sup>3</sup>	-0.00005	$\pm$	0.00020	$3\sqrt{3}$	0.00001	$\pm$	0.00004
-0.03 g/cm <sup>3</sup> (PERT)	0.00007	$\pm$	0.00000	$\sqrt{3}$	0.00004	$\pm$	0.00000
+0.03 g/cm <sup>3</sup> (PERT)	-0.00007	$\pm$	0.00000	$\sqrt{3}$	-0.00004	$\pm$	0.00000

**Absorber Content**

The boron content was varied  $\pm 9 \text{ wt.}\%$  from the nominal value of  $30 \text{ wt.}\%$  (Table 1.15) and the effect on the uncertainty in  $k_{\text{eff}}$  was determined. This value is three times the bounding limit. The uncertainty is taken from Table 1.29. Results are shown in Table 2.47.

The total number of control rod absorber pellets used in the core is approximately 1,300. The assumption is based on that only 13 control positions are approximately 50% inserted into the core. For determining the random component of the uncertainty, the results in Table 2.47 would be divided by  $\sqrt{1,300}$ .

Table 2.47. Effect of Uncertainty in CR Absorber Content.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-9 wt.%	0.00058	$\pm$	0.00020	$3\sqrt{3}$	0.00011	$\pm$	0.00004
+9 wt.%	-0.00063	$\pm$	0.00020	$3\sqrt{3}$	-0.00012	$\pm$	0.00004



**Absorber Impurity**

No information was available regarding any impurities present in the sintered B<sub>4</sub>C/C material for the control rod absorbers. However, the impurity limits provided in Table 1.29 can be applied to approximate a rough estimate of the impact of additional impurities. Sodium and manganese were added with a concentrations of 100 and 10 ppm (by weight), respectively. Concentrations of 1000 ppm (by weight) of aluminum, silicon, calcium, and titanium were also included. The effective uncertainty in  $k_{\text{eff}}$  was determined and is shown in Table 2.48. This value is treated as a bounding limit.

The uncertainty in the absorber impurity is considered all systematic with no random component.

Table 2.48. Effect of Uncertainty in Absorber Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
Added Impurities	-0.00052	$\pm$	0.00019	$\sqrt{3}$	-0.00030	$\pm$	0.00011

**Absorber Isotopic Abundance**

According to the 16<sup>th</sup> edition of the Chart of the Nuclides, the natural isotopic abundance of <sup>10</sup>B has been measured between 19.1 and 20.3% with a nominal value of 19.9%.<sup>a</sup> The abundance of <sup>10</sup>B in the control rods was therefore evaluated at the minimum and maximum bounding values to determine the effective uncertainty in  $k_{\text{eff}}$ . Table 1.28 states that the abundance of <sup>10</sup>B is 18.7 wt.%, which correlates to approximately 20.2 at.%. The model was not evaluated for a <sup>10</sup>B abundance of 18.7 wt.%. Results are shown in Table 2.49. The benchmark model was evaluated at 19.9 at.%.

The uncertainty in the absorber isotopic abundance is considered all systematic with no random component.

Table 2.49. Effect of Uncertainty in CR Isotopic Abundance of <sup>10</sup>B.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
19.1%	0.00018	$\pm$	0.00019	$\sqrt{3}$	0.00010	$\pm$	0.00011
20.3%	-0.00017	$\pm$	0.00020	$\sqrt{3}$	-0.00010	$\pm$	0.00011

<sup>a</sup> Nuclides and Isotopes: Chart of the Nuclides, 16<sup>th</sup> edition, (2002).

**Clad Density**

Two references were compared to confirm the composition and density of the Alloy 800H material.<sup>a,b</sup> The density of Alloy 800H is taken to be 8.03 g/cm<sup>3</sup> with an estimated uncertainty of  $\pm 0.03$  g/cm<sup>3</sup>. This value is the bounding limit. The effective change in  $k_{\text{eff}}$  was determined. Results are shown in Table 2.50.

The total number of control rod sections used in the core is approximately 130. The assumption is based on that only 13 control positions are approximately 50% inserted into the core. For determining the random component of the uncertainty, the results in Table 2.50 would be divided by  $\sqrt{130}$ .

Table 2.50. Effect of Uncertainty in Clad Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.03 g/cm <sup>3</sup> (1 $\sigma$ )	0.00026	$\pm$	0.00020	$\sqrt{3}$	0.00015	$\pm$	0.00011
+0.03 g/cm <sup>3</sup> (1 $\sigma$ )	0.00020	$\pm$	0.00019	$\sqrt{3}$	0.00012	$\pm$	0.00011

**Clad Composition**

The clad composition used in the benchmark model is derived from the aforementioned references for Alloy 800H. A summary of the composition of Alloy 800H is provided in Table 2.51. The nominal distribution is used in the benchmark model. The composition of the clad material was varied from the nominal value. In the first case, the minimum compositions of all elements were used with iron as the remaining balance. The second case uses the maximum weights of all the elements with a reduction in the iron content until the minimum value is achieved; then nickel is reduced to achieve a total of 100 wt.%. The effective change in  $k_{\text{eff}}$  for these two cases was determined. Results are shown in Table 2.52.

The uncertainty in the clad composition is considered all systematic with no random component.

<sup>a</sup> Specification Sheet: Alloy 800, 800H, and 800AT, Sandmeyer Steel Company (April 2004).

<sup>b</sup> Material Characteristics: Alloy 800H, PhilipCornes, <http://www.cornes.com.sg/a800h.htm> (Accessed August 5, 2008).

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Table 2.51. Composition of Alloy 800H.

Element	Minimum wt.%	Maximum wt.%	Nominal wt.%
Carbon	0.06	0.1	0.08
Aluminum	0.15	0.6	0.375
Silicon	--	1	0.35
Phosphorus	--	--	0.02
Sulfur	--	0.015	0.01
Titanium	0.15	0.6	0.375
Chromium	19	23	21
Manganese	--	1.5	1
Iron	39.5	--	43.99
Nickel	30	35	32.5
Copper	--	0.75	0.3
Total	--	--	100.000

Table 2.52. Effect of Uncertainty in Clad Composition.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
Minimum Fe	0.00012	$\pm$	0.00020	$\sqrt{3}$	0.00007	$\pm$	0.00011
Maximum Fe	0.00001	$\pm$	0.00020	$\sqrt{3}$	0.00001	$\pm$	0.00011

**Clad Impurity**

No information was available regarding any impurities present in the Alloy 800H metal. Concentrations of minor and trace elements found commonly in nickel alloy<sup>a</sup> were added to the clad material in the benchmark model (where neutron cross-section libraries were available) to determine the effective uncertainty in  $k_{\text{eff}}$ . A list of the additional impurities is shown in Table 2.53, and results are shown in Table 2.54. This value is treated as a bounding limit.

The uncertainty in the clad impurity is considered all systematic with no random component.

<sup>a</sup> J. H. Zaidi, S. Waheed, and S. Ahmed, "Determination of Trace Impurities in Nickel-Based Alloy using Neutron Activation Analysis," *J. Radioanal. Nucl. Ch.*, **242**: 259-263 (1999).

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Table 2.53. Concentration (by weight) of Minor and Trace Elements in Nickel Alloy.

Element	Concentration	Library Available?	Element	Concentration	Library Available?
As	63.7 ppm	Yes	Mo	9.8 ppm	Yes
Ba	215 ppm	Yes	Na	0.45 %	Yes
Br	8.9 ppm	Yes	Nd	16.2 ppm	Yes
Ca	5.9 %	Yes	Rb	24 ppb	Yes
Ce	0.33 %	No	Sb	2.2 ppm	Yes
Co	9.96 ppm	Yes	Sc	0.13 ppm	Yes
Cs	4.34 ppm	Yes	Se	0.16 ppm	No
Dy	6.12 ppm	No	Sm	0.23 ppm	Yes
Er	2.8 ppm	No	Sn	0.25 ppm	Yes
Eu	1.17 ppm	Yes	Sr	371 ppm	No
Ga	107.5 ppm	Yes	Ta	1.21 ppm	Yes
Gd	1.7 ppm	Yes	Tb	1.24 ppm	No
Hf	19.5 ppm	Yes	Th	18.3 ppm	Yes
Hg	6 ppb	Yes	U	4.3 ppm	Yes
In	245 ppb	Yes	Yb	2.48 ppm	No
La	2.34 ppm	No	V	0.75 %	Yes
Lu	1.16 ppm	Yes	Zn	93.4 ppm	Yes
Mg	2.14 %	Yes			

Table 2.54. Effect of Uncertainty in Clad Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
Added Impurities	-0.00014	$\pm$	0.00020	$\sqrt{3}$	-0.00008	$\pm$	0.00011

### 2.1.3.6 Instrumentation

Insufficient information is available to model and evaluate the uncertainties and biases related to the utility of instrumentation in the HTTR.

### 2.1.3.7 Graphite Blocks

#### **Density**

The IG-110 graphite density was varied  $\pm 0.04 \text{ g/cm}^3$  from a nominal value selected as  $1.76 \text{ g/cm}^3$  (ranging from  $1.75$  to  $1.78 \text{ g/cm}^3$ , Tables 1.13, 1.19, and 1.27) to determine the effective uncertainty in  $k_{\text{eff}}$ . A variation of  $0.03 \text{ g/cm}^3$  accounts for the difference in density between the various samples of IG-110 blocks, and a variation of  $0.01 \text{ g/cm}^3$  is assumed to encompass the variability in the volume fraction caused by eliminating various features such as the dowels, sockets, fuel handling position, and ridged

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features on the fuel rods. The assumed uncertainty of  $0.03 \text{ g/cm}^3$  for the IG-110 graphite encompasses the range of reported densities for throughout Section 1. Results are shown in Table 2.55. This value is treated as a bounding limit.

Previous results<sup>a</sup> state that a graphite weight difference of less than 1% should result in a bias of  $-0.3 \% \Delta k/k$ . Scaling the results provided in Table 2.55 provides an uncertainty of approximately  $0.245 \% \Delta k/k$ , comparable to the previously published information.

The total number of graphite blocks used in the core is approximately 549. For determining the random component of the uncertainty, the results in Table 2.55 would be divided by  $\sqrt{549}$ .

Table 2.55. Effect of Uncertainty in IG-110 Density in Graphite Blocks.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
$-0.04 \text{ g/cm}^3$	-0.00562	$\pm$	0.00020	$\sqrt{3}$	-0.00324	$\pm$	0.00011
$+0.04 \text{ g/cm}^3$	0.00550	$\pm$	0.00019	$\sqrt{3}$	0.00318	$\pm$	0.00011

**Impurity**

The graphite block impurity was varied from 0-1 ppm by weight of equivalent natural-boron content (Table 1.13) to determine the bounding uncertainty in  $k_{\text{eff}}$ . The nominal impurity is 0.40 ppm or 0.37 ppm of natural boron by weight for the fuel/control blocks and reflector blocks, respectively (Table 1.27). The 0.40 ppm value was selected for the average block composition in IG-110 graphite in the core. Results are shown in Table 2.56.

It has been reported that the estimated air content in the graphite blocks would provide  $-0.4 \% \Delta k/k$  to the computational model.<sup>a</sup> The inclusion of air in the benchmark model did not produce a noticeable change in reactivity. However, it is unclear exactly how much air could be entrapped within the graphite blocks. Air can be entrapped during the graphitization process or absorbed onto the graphite surface. Typically, significant contribution to the equivalent boron content in a graphite block is caused by impurities in the graphite. Methods to measure the impurity content in graphite have improved over the past several years, and the impurity content of the HTTR graphite may need to be reassessed.<sup>b</sup>

The uncertainty in the graphite block impurity is considered all systematic with no random component.

Table 2.56. Effect of Uncertainty in IG-110 Impurity in Graphite Blocks.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00542	$\pm$	0.00019	$\sqrt{3}$	0.00313	$\pm$	0.00011
1 ppm	-0.00770	$\pm$	0.00020	$\sqrt{3}$	-0.00445	$\pm$	0.00011

<sup>a</sup> Fujimoto, N., Nakano, M., Takeuchi, M., Fujisaki, S., and Yamashita, K., "Start-Up Core Physics Tests of High Temperature Engineering Test Reactor (HTTR), (II): First Criticality by an Annular Form Fuel Loading and Its Criticality Prediction Method," *J. Atomic Energy Society Japan*, **42**(5), 458-464 (2000).

<sup>b</sup> Private communication with Rob Bratton at Idaho National Laboratory (November 20, 2008).

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## 2.1.3.8 Permanent Reflectors

**Density**

The PGX graphite density was varied  $\pm 0.04 \text{ g/cm}^3$  from a nominal value selected as  $1.73 \text{ g/cm}^3$  (ranging from  $1.73$  to  $1.74 \text{ g/cm}^3$ , Tables 1.19, 1.20, and 1.27) to determine the effective uncertainty in  $k_{\text{eff}}$ . A variation of  $0.03 \text{ g/cm}^3$  accounts for the PGX, and a variation of  $0.01 \text{ g/cm}^3$  is assumed to encompass the variability in the volume fraction, which is not provided. The assumed uncertainty of  $0.03 \text{ g/cm}^3$  for the PGX graphite encompasses the range of reported densities for throughout Section 1. Results are shown in Table 2.57. This value is treated as a bounding limit.

The total number of permanent reflector blocks used in the core is approximately 96. For determining the random component of the uncertainty, the results in Table 2.57 would be divided by  $\sqrt{96}$ .

Table 2.57. Effect of Uncertainty in PGX Density in Permanent Reflector.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
$-0.04 \text{ g/cm}^3$	-0.00031	$\pm$	0.00020	$\sqrt{3}$	-0.00018	$\pm$	0.00011
$+0.04 \text{ g/cm}^3$	0.00057	$\pm$	0.00020	$\sqrt{3}$	0.00033	$\pm$	0.00011

**Impurity**

The permanent reflector impurity was varied from 0-5 ppm by weight of equivalent natural-boron content (Table 1.16) to determine the bounding uncertainty in  $k_{\text{eff}}$ . The nominal impurity is 1.91 ppm of natural boron by weight (Table 1.27). Results are shown in Table 2.58.

The uncertainty in the permanent reflector impurity is considered all systematic with no random component.

Table 2.58. Effect of Uncertainty in PGX Impurity in Permanent Reflector.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
0 ppm	0.00411	$\pm$	0.00019	$\sqrt{3}$	0.00237	$\pm$	0.00011
5 ppm	-0.00518	$\pm$	0.00020	$\sqrt{3}$	-0.00299	$\pm$	0.00011

## 2.1.3.9 Dummy Blocks

The 30-fuel-column core does not contain dummy blocks.

## 2.1.3.10 Helium Coolant

**Density**

The density of helium gas was evaluated using the ideal gas law,  $PV=nRT$ , at a pressure of 1 atm and temperature of  $25^\circ\text{C}$ . The helium coolant was modeled with a atom density of  $2.4616 \times 10^{-4} \text{ atoms/b-cm}$  (mass density of  $1.6361 \times 10^{-4} \text{ g/cm}^3$ ).

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The helium density was varied approximately  $\pm 3.5\%$  ( $1\sigma$ ) to determine the effective uncertainty in  $k_{\text{eff}}$ . Results are shown in Table 2.59. The perturbation card, PERT, in MCNP was also employed in an independent analysis to determine the effective uncertainty. The larger uncertainty in  $\Delta k_{\text{eff}}$  will be applied towards the total uncertainty analysis in Section 2.1.7.

The effect of neglecting the helium content and replacing it with an empty void was also performed. The effective change in change in  $k_{\text{eff}}$  was determined. Results are shown in Table 2.59.

The uncertainty in the helium density is considered all systematic with no random component.

Table 2.59. Effect of Uncertainty in Helium Coolant Density.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
void	-0.00035	$\pm$	0.00020	1	NA	$\pm$	NA
-3.5% ( $1\sigma$ )	-0.00030	$\pm$	0.00020	1	-0.00030	$\pm$	0.00020
+3.5% ( $1\sigma$ )	0.00007	$\pm$	0.00020	1	0.00007	$\pm$	0.00020
void (PERT)	-0.00002	$\pm$	0.00001	1	NA	$\pm$	NA
-3.5% ( $1\sigma$ , PERT)	0.00000	$\pm$	0.00000	1	0.00000	$\pm$	0.00000
+3.5% ( $1\sigma$ , PERT)	0.00000	$\pm$	0.00000	1	0.00000	$\pm$	0.00000

## Impurity

The upper impurity limit in the helium coolant was included in the model to determine its effect upon the uncertainty in  $k_{\text{eff}}$ . Upper concentration limits from Table 1.10 were included although the impurity content in helium has at room temperature would be considerably less; therefore it will be treated as a bounding limit. The bounding limit was tripled for each component in order to assess the uncertainty, but the effects were still negligible. Results are shown in Table 2.60.

The uncertainty in the helium impurity is considered all systematic with no random component.

Table 2.60. Effect of Uncertainty in Helium Coolant Impurity.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
max impurities	-0.00008	$\pm$	0.00019	$2\sqrt{3}$	-0.00002	$\pm$	0.00006
3×maximum	-0.00004	$\pm$	0.00019	$6\sqrt{3}$	0.00000	$\pm$	0.00002

## 2.1.4 Computational Analyses

### 2.1.4.1 Room Return

Insufficient information is available to model and evaluate the uncertainties and biases related to any room return effects in the HTTR. Shielding plugs, plates, and blocks are incorporated within the HTTR vessel and would considerably reduce room return effects from the surrounding reactor vessel, reactor internals, and HTTR infrastructure and facility.

A conservative analysis of the HTTR benchmark model surrounded by shielding material, steel, and concrete provided an insignificant effect because the slight increase in the effective multiplication factor was below the statistical uncertainty of the analysis. Actual room return effects would be assumed quite negligible.

#### 2.1.4.2 Stochastic TRISO Sampling

Currently MCNP has a limited capability for modeling stochastic geometries. The HTTR core is unique in the fact that it has 12 different enrichments throughout the core. MCNP could effectively model randomness for only two types of TRISO particles at once because the URAN card used to create stochastic models has a maximum limit of two universes per model. Therefore a complete analysis of the stochastic nature of the fuel particles in a complete core model cannot be currently evaluated.

To provide an idea of the effects of TRISO placement, the fuel compacts were modeled with a uniform cell lattice of TRISO particles (including partial particles). Figures 2.1 and 2.2 show a cross section of the fuel compacts with ordered and uniformly-filled TRISO distributions, respectively. The effective difference in the multiplication factor is shown in Table 2.61. The 30 vol. % packing fraction and uranium mass per fuel rod is conserved in this comparison. The reduction in resonance shielding effects in partial particles (i.e., fuel kernels not completely surrounded by graphite coatings) along the edges is not considered in this analysis.

An analysis performed at the University of Michigan to look at the effects of explicitly modeling particle fuel in a very-high temperature gas-cooled reactor found that in a full core model, the effect of modeling with a uniform lattice and clipped TRISO particles would have a reduced  $k_{\text{eff}}$  of approximately 0.1 %.<sup>a</sup> The results shown in Table 2.61 are slightly greater than 0.1 % and demonstrate the same trend, where the uniform lattice model reports a slightly lower  $k_{\text{eff}}$  value.

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<sup>a</sup> W. Ji, J. L. Conlin, W. R. Martin, J. C. Lee, and F. B. Brown, "Explicit Modeling of Particle Fuel for the Very-High Temperature Gas-Cooled Reactor," *Trans. Am. Nucl. Soc.*, **92** (June 2005).



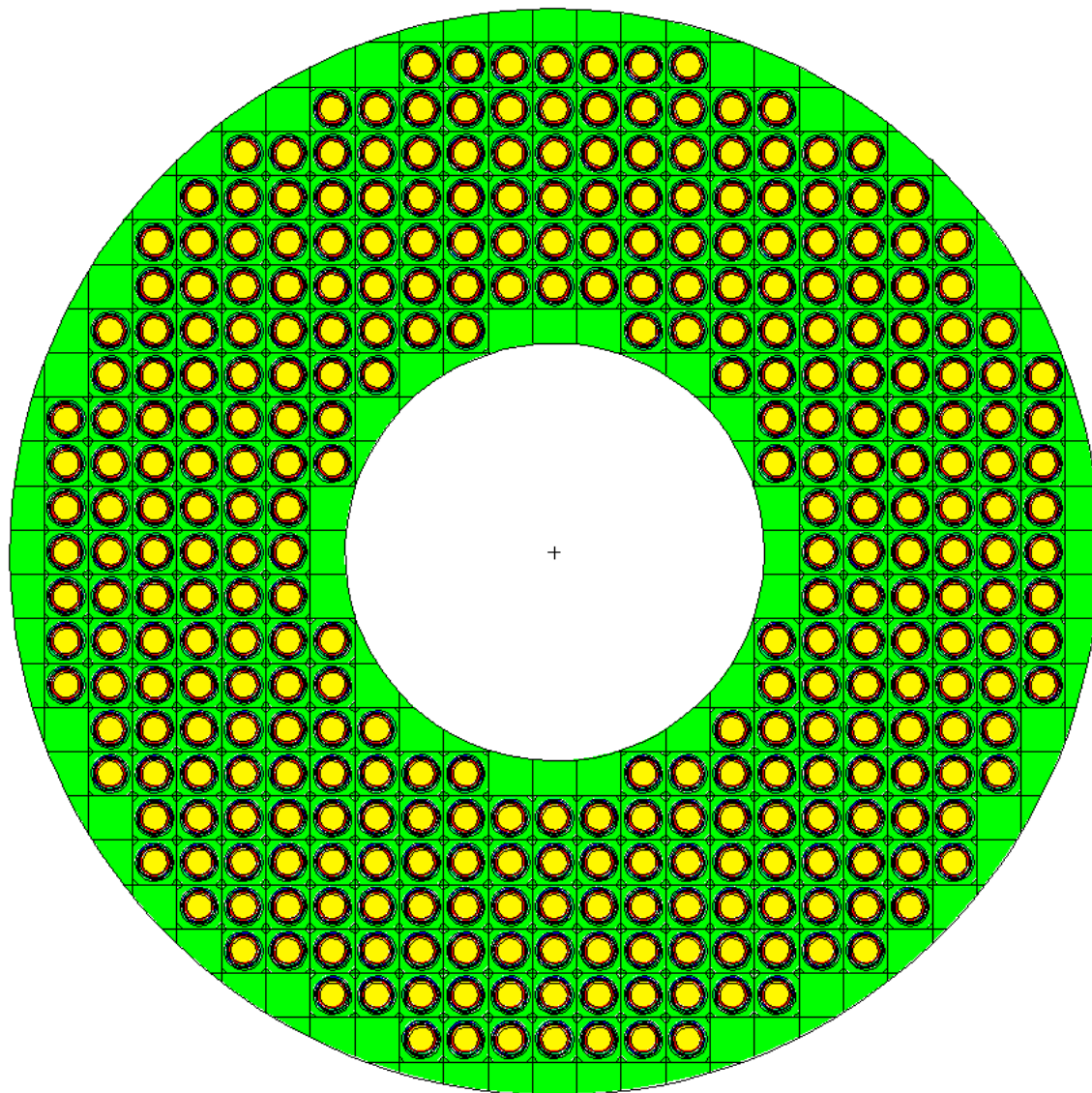


Figure 2.1. MCNP Ordered TRISO Lattice within the Fuel Compacts.

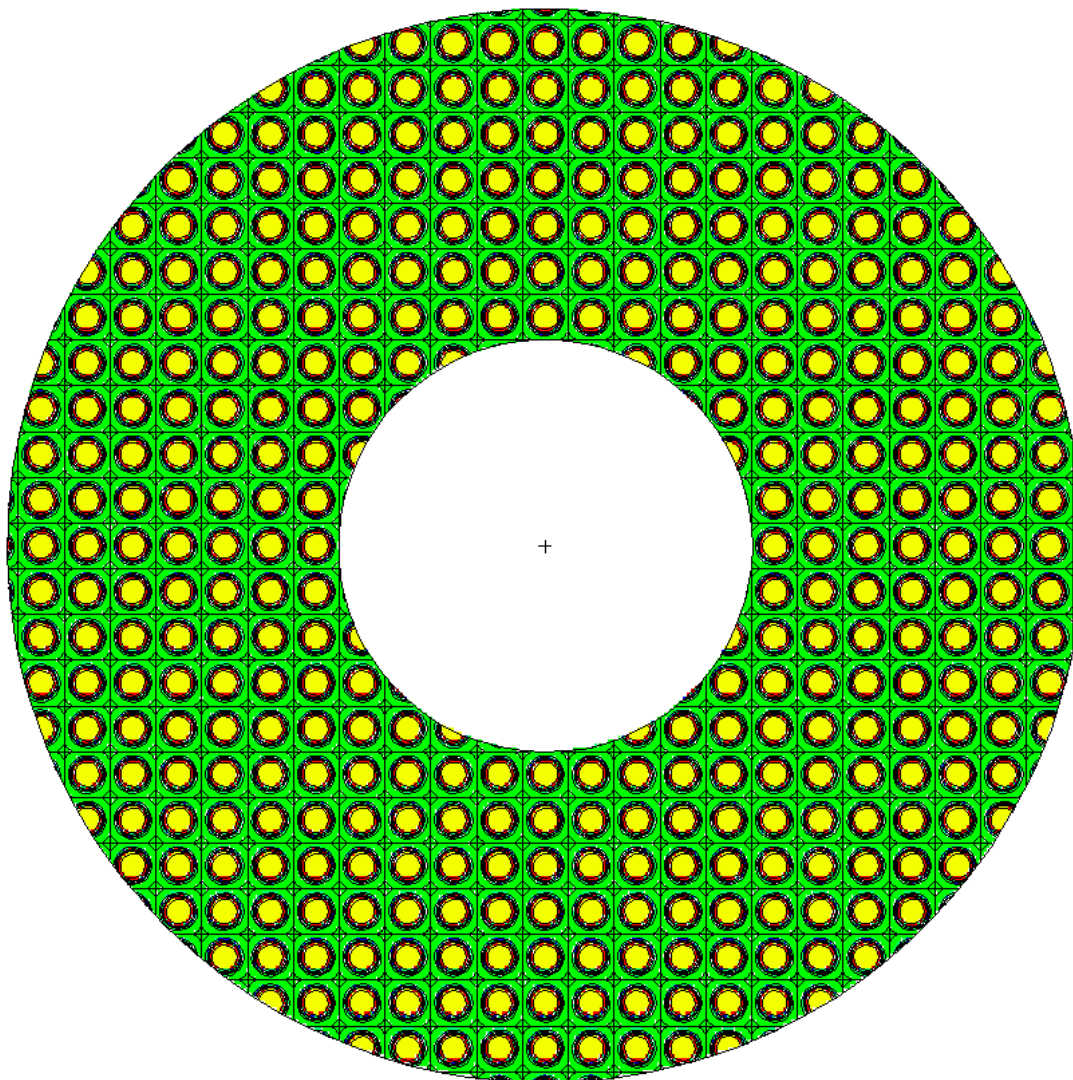


Figure 2.2. MCNP Uniformly-Filled TRISO Lattice within the Fuel Compacts.

Table 2.61. Comparison of Uniform and Organized TRISO Fill in Fuel Compacts.

Organized Distribution (Figure 2.1)			Uniform Fill (Figure 2.2)			Effective Difference		
$k_{\text{eff}}$	$\pm$	$\sigma_k$	$k_{\text{eff}}$	$\pm$	$\sigma_k$	$\Delta k$	$\pm$	$\sigma_{\Delta k}$
1.02532	$\pm$	0.00014	1.02367	$\pm$	0.00014	0.00165	$\pm$	0.00020

### 2.1.4.3 Stochastic Block Sampling

As discussed in the previous section, MCNP has a limited capability for modeling stochastic geometries for systems with multiple unique components. Whereas the HTTR core model is comprised of 61 columns each containing 9 bricks, a comprehensive analysis of the stochastic nature of block stacking could not be evaluated.

### 2.1.4.4 MCNP Random Number Generation

The random number seed was changed for the benchmark model and compared with the original to determine the uncertainty in utilizing Monte Carlo analysis methods. Results are shown in Table 2.62.

Table 2.62. Analysis of the Effect of Random Number Generation in MCNP.

Seed	$k_{\text{eff}}$	$\pm$	$\sigma_{k_{\text{eff}}}$
123456787	1.02544	$\pm$	0.00014
9876543279	1.02504	$\pm$	0.00014
198765432799	1.02545	$\pm$	0.00013
17623486105893	1.02559	$\pm$	0.00014
19073486328125 <sup>(a)</sup>	1.02532	$\pm$	0.00014
32160231045432797	1.02545	$\pm$	0.00014
Average	1.02538	$\pm$	0.00019

(a) Primary seed value for evaluation.

### 2.1.4.5 Simplification Biases and Uncertainties

Whereas insufficient information is publicly available, a comprehensive analysis of simplification biases and their respective uncertainties could not be appropriately assessed. As additional information becomes available, highly detailed and simplified benchmark models can be generated and their biases can be adequately determined.

## 2.1.5 Systematic Biases and Uncertainties

There were no systematic biases or uncertainties reported for these experiments.

As discussed at the beginning of Section 2, all uncertainties are treated as 25% deterministic (or systematic) with no reduction in uncertainty due to the multiplicity of core components and as 75% random.

## 2.1.6 Analysis of HTTR Uranium Content

Because of the overspecification of the TRISO particles in Table 1.14 and the correlation of uranium kernel diameter, density, and mass, the effect of the uncertainty in the kernel diameter and density is not included in the total uncertainty. The uranium content of the fuel rods of  $188.58 \pm 5.66$  g (Table 1.14) is the parameter most likely known with the greatest accuracy. Therefore, the diameter of the kernels will be fixed at 600  $\mu\text{m}$ , and the density will be varied  $\pm 0.32$  g/cm<sup>3</sup> (bounding limit) from a nominal value of 10.39 g/cm<sup>3</sup> to determine the effective uncertainty in  $k_{\text{eff}}$  due to the uranium mass uncertainty. Results derived from those shown in Table 2.23 are shown in Table 2.63.

The total number of fuel rods used in the core is approximately 4,770. For determining the random component of the uncertainty, the results in Table 2.63 would be divided by  $\sqrt{4,770}$ .

Table 2.63. Effect of Uncertainty in Uranium Mass.

Deviation	$\Delta k$	$\pm$	$\sigma_{\Delta k}$	Scaling Factor	$\Delta k_{\text{eff}} (1\sigma)$	$\pm$	$\sigma_{\Delta k_{\text{eff}}}$
-0.26 g/cm <sup>3</sup>	-0.00374	$\pm$	0.00019	$\times (0.32/0.26)/\sqrt{3}$	-0.00266	$\pm$	0.00014
+0.26 g/cm <sup>3</sup>	0.00369	$\pm$	0.00020	$\times (0.32/0.26)/\sqrt{3}$	0.00262	$\pm$	0.00014

### 2.1.7 Total Experimental Uncertainty

A compilation of the total evaluated uncertainty in the HTTR model is shown in Table 2.64 for the 30-fuel-column core design. As discussed earlier, each of the evaluated uncertainties is divided into a systematic component (25 %) and random component (75 %) where appropriate. The random component is then divided by the square-root of the number of random objects. The root-mean square of each subcomponent of the total uncertainty is determined and the total uncertainty is the root-mean square of all the random and systematic uncertainties.

Uncertainties less than 0.00001 are reported as negligible (neg). When calculated uncertainties in  $\Delta k_{\text{eff}}$  are less than their statistical uncertainties, the statistical uncertainties are used in the calculation of the total uncertainty. Table listings where calculations were not performed or otherwise not available are labeled with 'NA'. For uncertainties where a random component is not applicable, the uncertainty is denoted with '--'.

The most significant contributions to the overall uncertainty from the systematic uncertainties include the impurities in the IG-110 graphite blocks and impurities in the PGX graphite blocks. All uncertainties providing at least 0.1 %  $\Delta k_{\text{eff}}$  are highlighted in gray in Table 2.64. All of the random uncertainties are less than 0.1 %  $\Delta k_{\text{eff}}$ . The overall uncertainty is less than 1%  $\Delta k_{\text{eff}}$ ; it is expected that the total uncertainty will be reduced as additional parameters that characterize the HTTR are obtained.

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Table 2.64. Total Experimental Uncertainty of the Fully-Loaded 30-Fuel-Column Core.

Varied Parameter	Systematic Uncertainty		Random Uncertainty	
	$-\Delta k_{\text{eff}} (1\sigma)$	$+\Delta k_{\text{eff}} (1\sigma)$	$-\Delta k_{\text{eff}} (1\sigma)$	$+\Delta k_{\text{eff}} (1\sigma)$
Temperature	0.00066	-0.00066	--	--
Control Rod Positions	-0.00013	0.00016	-0.00007	0.00009
Measured Value of $k_{\text{eff}}$	NA	NA	NA	NA
Kernel Diameter	See Section 2.1.6			
Buffer Diameter	0.00003	neg	neg	neg
IPyC Diameter	neg	neg	neg	neg
SiC Diameter	0.00013	-0.00009	neg	neg
OPyC Diameter	neg	neg	neg	neg
Overcoat Diameter	NA	NA	NA	NA
Compact Inner Diameter	0.00006	0.00007	neg	neg
Compact Outer Diameter	-0.0005	0.00015	neg	neg
Compact Height	neg	neg	neg	neg
Compact Packing Fraction	See Section 2.1.6			
Sleeve Inner Diameter	0.00022	-0.00023	neg	neg
Sleeve Outer Diameter	-0.00120	0.00124	-0.00005	0.00005
Sleeve Height	0.00010	0.00005	neg	neg
BP Diameter	0.00197	-0.00183	0.00001	-0.00001
BP Stack Height	0.00019	-0.00016	neg	neg
BP Pin Hole Diameter	0.00016	-0.00015	neg	neg
Graphite Disk Diameter	0.00003	neg	neg	neg
Disk Stack Height	neg	0.00005	neg	neg
CR Absorber Inner Diameter	-0.00003	0.00005	neg	neg
CR Absorber Outer Diameter	0.00025	-0.00025	0.00002	-0.00002
CR Absorber Height	0.00012	-0.00012	0.00003	-0.00003
CR Clad Inner Diameter	neg	0.00003	neg	neg
CR Clad Outer Diameter	-0.00016	0.00013	-0.00004	0.00003
CR Clad Height	0.00012	-0.00015	0.00003	-0.00004
CR Spine Diameter	0.00003	neg	neg	neg
Instrumentation Dimensions	NA	NA	NA	NA
Block Flat-to-Flat Distance	-0.00040	0.00036	-0.00005	0.00005
Graphite Block Height	0.00014	-0.00026	0.00002	-0.00003
Dowel/Socket Dimensions	NA	NA	NA	NA
Fuel Channel Diameter	0.00070	-0.00072	0.00003	-0.00003
Reflector Channel Diameter	neg	neg	neg	neg
Channel Pitch	-0.00004	0.00004	neg	neg

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Table 2.64 (cont.). Total Experimental Uncertainty of the Fully-Loaded 30-Fuel-Column Core.

Varied Parameter	Systematic Uncertainty		Random Uncertainty	
	$-\Delta k_{\text{eff}}(1\sigma)$	$+\Delta k_{\text{eff}}(1\sigma)$	$-\Delta k_{\text{eff}}(1\sigma)$	$+\Delta k_{\text{eff}}(1\sigma)$
Handling Socket Dimensions	NA	NA	NA	NA
Column Pitch	0.00056	-0.00066	0.00022	-0.00025
CR Channel Diameter	0.00013	-0.00014	0.00005	-0.00006
CR Channel Pitch	0.00006	0.00005	0.00002	neg
Permanent Reflector Diameter	-0.00028	0.00061	--	--
Dummy Block Dimensions	NA	NA	NA	NA
3.4 wt.% Enrichment	-0.00162	0.00161	--	--
3.9 wt.% Enrichment	-0.00096	0.00151	--	--
4.3 wt.% Enrichment	-0.00209	0.00195	--	--
4.8 wt.% Enrichment	-0.00050	0.00073	--	--
5.2 wt.% Enrichment	-0.00040	0.00052	--	--
5.9 wt.% Enrichment	-0.00067	0.00042	--	--
6.3 wt.% Enrichment	0.00020	0.00029	--	--
6.7 wt.% Enrichment	-0.00026	0.00020	--	--
7.2 wt.% Enrichment	0.00020	0.00019	--	--
7.9 wt.% Enrichment	-0.00023	0.00020	--	--
9.4 wt.% Enrichment	0.00020	0.00028	--	--
9.9 wt.% Enrichment	0.00020	0.00020	--	--
Oxygen to Uranium Ratio	-0.00010	0.00005	--	--
UO <sub>2</sub> Density	See Section 2.1.6			
UO <sub>2</sub> Impurity	0.00017	-0.00002	--	--
Buffer Density	neg	neg	neg	neg
Buffer Impurity	0.00011	0.00011	--	--
IPyC Density	neg	neg	neg	neg
IPyC Impurity	0.00011	0.00011	--	--
SiC Density	NA	neg	NA	neg
SiC Impurity	0.00011	0.00011	--	--
OPyC Density	-0.00001	0.00001	neg	neg
OPyC Impurity	neg	0.00016	--	--
Overcoat Density	-0.00003	0.00002	neg	neg
Overcoat Composition	NA	NA	NA	NA
Overcoat Impurity	0.00038	-0.00088	--	--
Compact Density	-0.00001	0.00001	neg	neg
Compact Impurity	0.00011	-0.00040	--	--
Compact Free U Content	NA	0.00002	--	--

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Table 2.64 (cont.). Total Experimental Uncertainty of the Fully-Loaded 30-Fuel-Column Core.

Varied Parameter	Systematic Uncertainty		Random Uncertainty	
	$-\Delta k_{\text{eff}}(1\sigma)$	$+\Delta k_{\text{eff}}(1\sigma)$	$-\Delta k_{\text{eff}}(1\sigma)$	$+\Delta k_{\text{eff}}(1\sigma)$
Sleeve Density	-0.00006	0.00005	neg	neg
Sleeve Impurity	0.00033	-0.00021	--	--
BP Absorber Density	0.00013	-0.00014	neg	neg
BP Absorber Content, 2.0 wt.%	0.00087	-0.00072	neg	neg
BP Absorber Content, 2.5 wt.%	0.00026	-0.00022	neg	neg
BP Absorber Impurity	NA	-0.00030	--	--
BP Isotopic Abundance of $^{10}\text{B}$	0.00154	-0.00054	--	--
Graphite Disk Density	neg	neg	neg	neg
Graphite Disk Impurity	0.00011	0.00011	--	--
CR Absorber Density	0.00001	neg	neg	neg
CR Absorber Content	0.00003	-0.00003	neg	neg
CR Absorber Impurity	NA	-0.00030	--	--
CR Isotopic Abundance of $^{10}\text{B}$	0.00011	0.00011	--	--
CR Clad Density	0.00004	0.00003	neg	neg
CR Clad Composition	0.00011	neg	--	--
CR Clad Impurity	NA	0.00011	--	--
Instrumentation Composition	NA	NA	NA	NA
IG-110 Density in Blocks	-0.00081	0.00079	-0.00010	0.00010
IG-110 Impurity in Blocks	0.00313	-0.00445	--	--
PGX Density	-0.00004	0.00008	-0.00001	0.00003
PGX Impurity	0.00237	-0.00299	--	--
Dummy Block Composition	NA	NA	NA	NA
Helium Coolant Density	-0.00030	0.00020	--	--
Helium Coolant Impurity	NA	neg	--	--
Room Return	NA	NA	NA	NA
Stochastic TRISO Sampling	0.00041	0.00041	neg	neg
Stochastic Block Sampling	NA	NA	NA	NA
MCNP Random Number Seed	--	--	0.00019	0.00019
Uranium Fuel Mass (Sec. 2.1.6)	-0.00066	0.00066	-0.00003	0.00003
Total Uncertainty of Components	0.00605	0.00702	0.00034	0.00037
<b>Total Evaluation Uncertainty</b>	<b>0.00606</b>	<b>0.00703</b>		

## **2.2 Evaluation of Buckling and Extrapolation Length Data**

Buckling and extrapolation length measurements were not made.

## **2.3 Evaluation of Spectral Characteristics Data**

Spectral characteristics measurements were not made.

## **2.4 Evaluation of Reactivity Effects Data**

Reactivity effects measurements were not evaluated.

## **2.5 Evaluation of Reactivity Coefficient Data**

Reactivity coefficient measurements were not evaluated.

## **2.6 Evaluation of Kinetics Measurements Data**

Kinetics measurements were not evaluated.

## **2.7 Evaluation of Reaction-Rate Distributions**

Reaction-rate distribution measurements were not evaluated.

## **2.8 Evaluation of Power Distribution Data**

Power distribution measurements were not made.

## **2.9 Evaluation of Isotopic Measurements**

Isotopic measurements were not made.

## **2.10 Evaluation of Other Miscellaneous Types of Measurements**

Other miscellaneous types of measurements were not made.



### 3.0 BENCHMARK SPECIFICATIONS

#### 3.1 Benchmark-Model Specifications for Critical and / or Subcritical Measurements

Whereas insufficient information is publicly available, a finely-detailed benchmark model could not be established. A mid-grade benchmark of the HTTR was prepared and analyzed. The simplification bias for this model could also not be appropriately determined.

Currently only the fully-loaded 30-fuel-column core critical is modeled.

##### 3.1.1 Description of the Benchmark Model Simplifications

Significant simplifications were incurred to develop a benchmark model of the HTTR because of a lack of information publicly available to determine dimensions and compositions. Said simplifications will be discussed where applicable in the descriptions of the dimension and material properties of the model.

As stated previously, biases of the model were not assessed but will be addressed as additional HTTR information becomes available. Biases that have been partially investigated are listed in Section 3.1.1.1.

The fuel handling positions, dowels, and sockets were not included in the model but were accounted for with a void fraction of 0.5 % reduction in graphite density (based upon volume calculations using dimensions provided in Figure 1.52). The burnable poison insertion holes were placed on the same pitch as the fuel channels so as to simplify the model.

It is apparent from a comparison of Figures 1.65 and 1.67 that the depth to which the control rod, RSS, and instrumentation holes are drilled varies. A depth of 1060 mm above the bottom of the core was selected for all positions so as to simplify the model.

Insufficient information was available to model the bottom-most reflector block according to actual design; therefore it was modeled with the same design as the two top reflector blocks and the other bottom reflector block. The top and bottom of each coolant channel is expected to taper from the 21-mm diameter to the 41-mm diameter of the fuel assemblies, but information was unavailable to describe the model.

Individual sections of the permanent reflector were not modeled.

Insufficient information was available to model the shielding blocks surrounding the core and shielding plugs in the core. Therefore, they were not included in the benchmark model. It is assumed that all neutrons reaching the core boundaries are lost and not scattered back by the shielding material. A conservative estimate of room-return effects demonstrated a negligible change in  $k_{\text{eff}}$ .

In the materials section, impurity contents in the materials are based upon natural boron equivalency. In the model, however, only the  $^{10}\text{B}$  component is included, as the effect of the  $^{11}\text{B}$  content would be insignificant.

The density is the same ( $1.80 \text{ g/cm}^3$ ) for both types of burnable poison pellets. The boron content in the pellets are based on the reported weight percents instead of the atomic percents.

##### 3.1.1.1 Assessed Biases

Although some biases have been partially investigated, there is incomplete information regarding the HTTR to properly address simplification biases in order to adjust the benchmark  $k_{\text{eff}}$ . As stated previously, a conservative estimate of potential room-return effects provided negligible results. As shown in Section 2.1.3.2 in Table 2.37, the effect of neglecting the free uranium content of the fuel

compacts was negligible. Finally, the effect of modeling the helium coolant as void material was also negligible (as shown in Section 2.1.3.10, Table 2.59).

### 3.1.2 Dimensions

#### 3.1.2.1 Prismatic Pin-in-Block Fuel

##### TRISO Particles

The basic ingredient for HTTR fuel is the TRISO particle. A  $\text{UO}_2$  kernel is surrounded by four coatings: a low density porous pyrolytic carbon (PyC) buffer layer, a high density inner isotropic PyC layer, a SiC layer, and a final outer PyC layer. A resinated graphite overcoat is then deposited around each TRISO particle. Figure 3.1 depicts the TRISO layers and their respective dimensions.

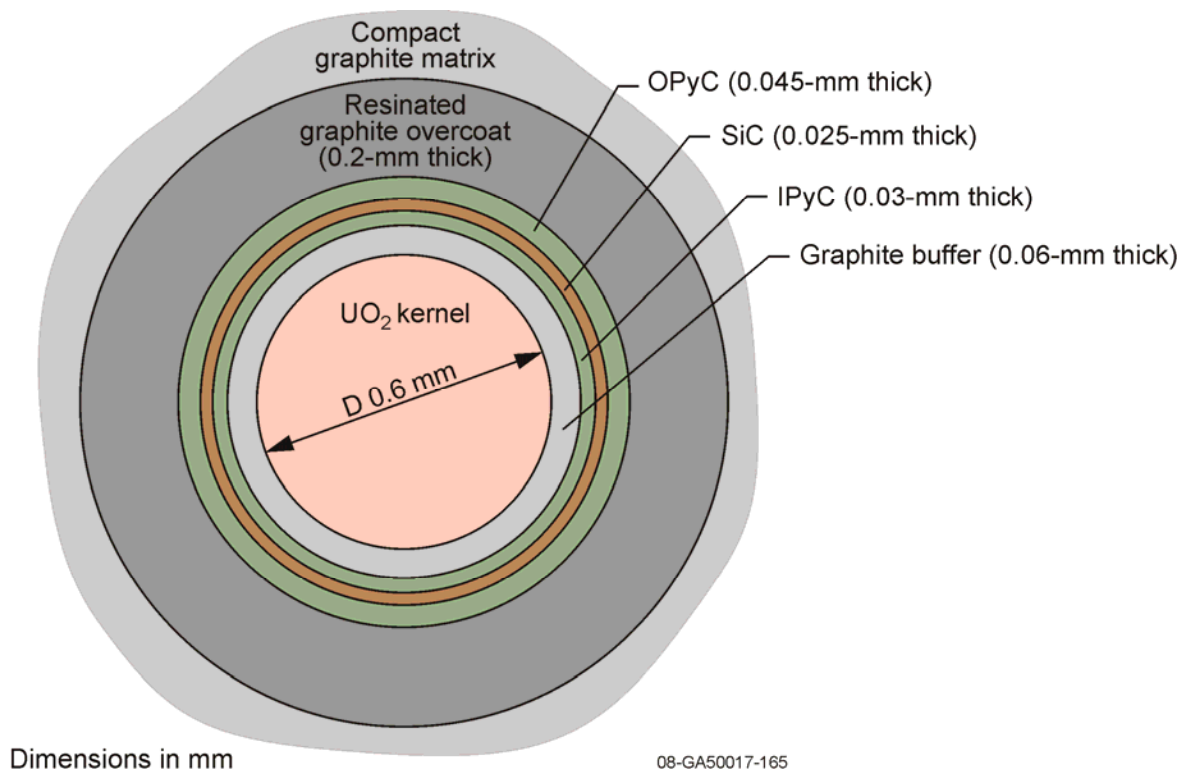


Figure 3.1. TRISO-Coated Fuel Particle.

##### Compacts

All fourteen fuel compacts in a single fuel pin are modeled as a single unit filled with the TRISO lattice. The stacked compacts have an inner diameter of 1 cm, an outer diameter of 2.6 cm, and an overall height of 54.6 cm.

A horizontal cross section of the compacts is shown in Figure 3.2. In the benchmark model, 12,987 TRISO particles are randomly distributed throughout the compact matrix in a single compact. While the benchmark model retains randomness in distribution, many computer codes cannot properly model such configurations. It is up to the user to determine which method is most appropriate while accounting for its impact on the reactivity of the model. Example means for analyzing this model are provided in

Section 4.1. The difference in methods for accurately modeling random TRISO particles in a full-core reactor is on the order of  $\sim 0.1\%$  of  $k_{\text{eff}}$ .<sup>a</sup>

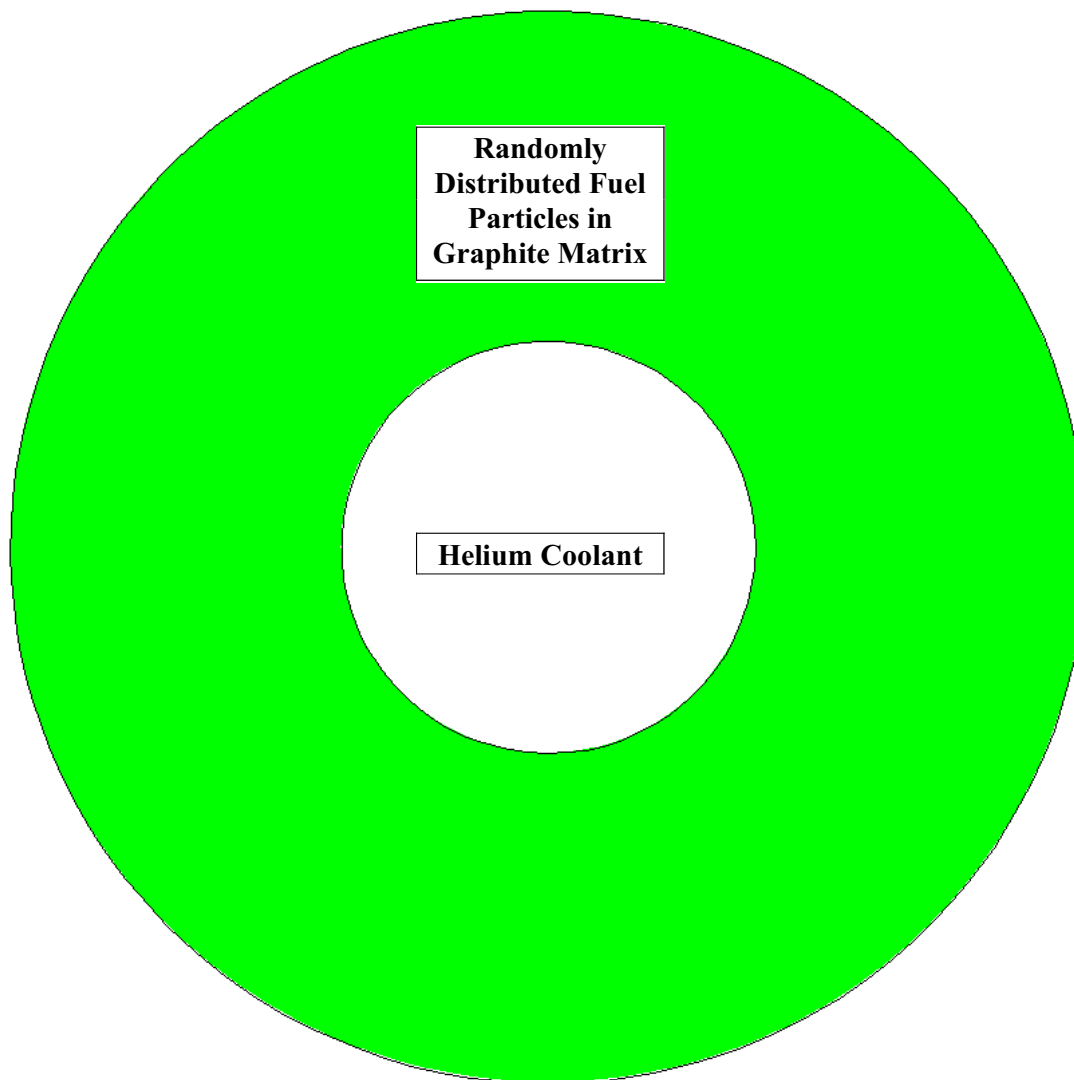


Figure 3.2. Fuel Compact (Filled with Randomly Distributed TRISO Particles).

<sup>a</sup> W. Ji, J. L. Conlin, W. R. Martin, J. C. Lee, and F. B. Brown, "Explicit Modeling of Particle Fuel for the Very-High Temperature Gas-Cooled Reactor," *Trans. Am. Nucl. Soc.*, **92** (June 2005).

**Fuel Element**

An approximate description of the HTTR fuel element is modeled (Figure 3.3).

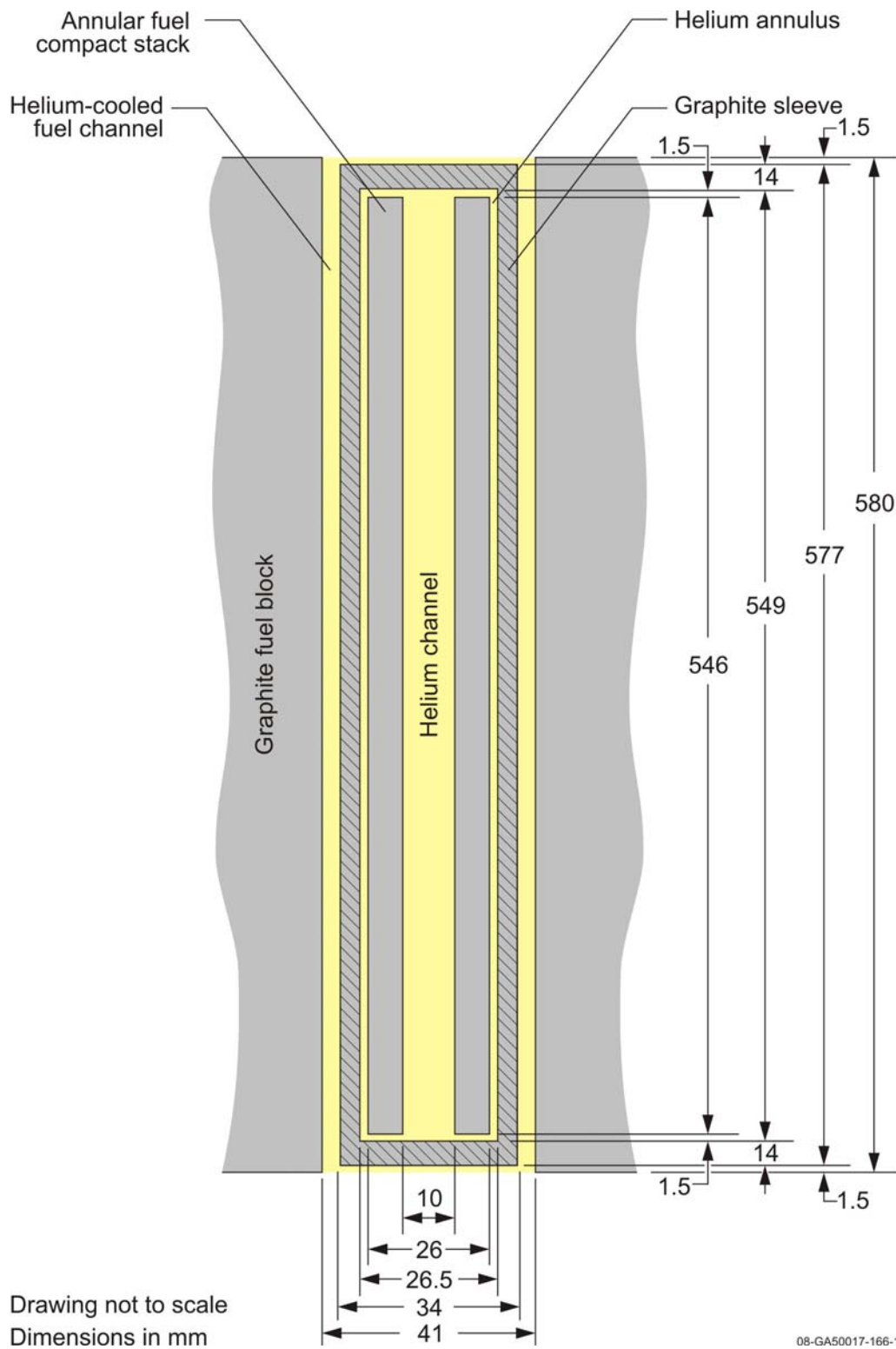
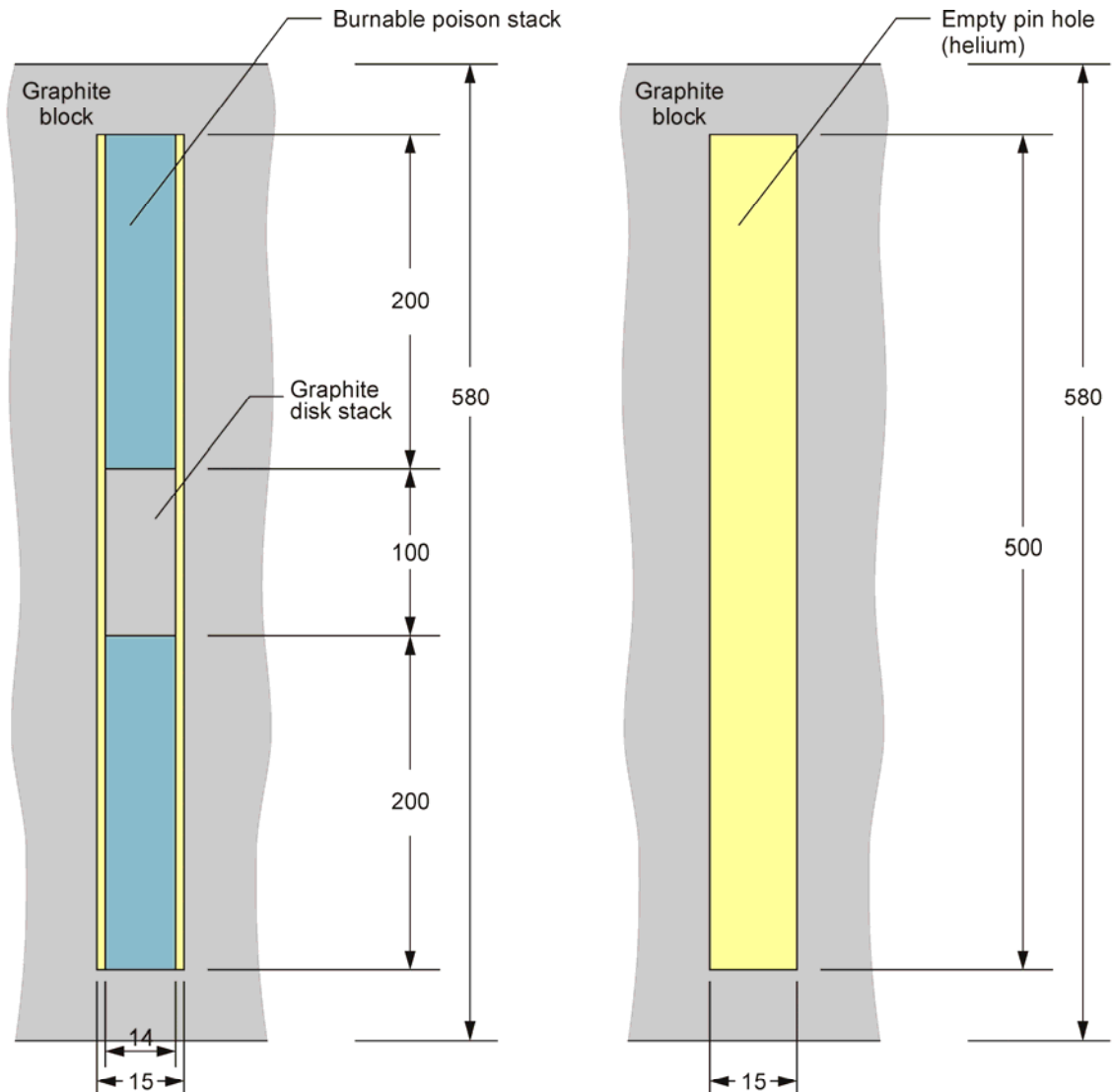


Figure 3.3. Benchmark HTTR Fuel Element.

08-GA50017-166-1

### 3.1.2.2 Burnable Poisons

The burnable poison pellets and graphite disks were modeled as individual stacks contained within a pin position in the fuel blocks (Figure 3.4). Each fuel block contained two BP pins and one empty pin position.



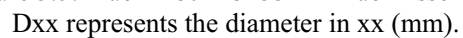
Drawing not to scale  
Dimensions in mm

Figure 3.4. Burnable Poison Pin (Left) and Empty Pin Position (Right).

08-GA50017-166-2

### 3.1.2.3 Fuel Blocks

The HTTR contains two types of regular hexagonal fuel blocks: 33-pin (Zones 1 and 2) and 31-pin (Zones 3 and 4). Diagrams of each fuel block design implemented in the benchmark model are shown in Figures 3.5 and 3.6, respectively. The pitch for all positions is 51.5 mm.



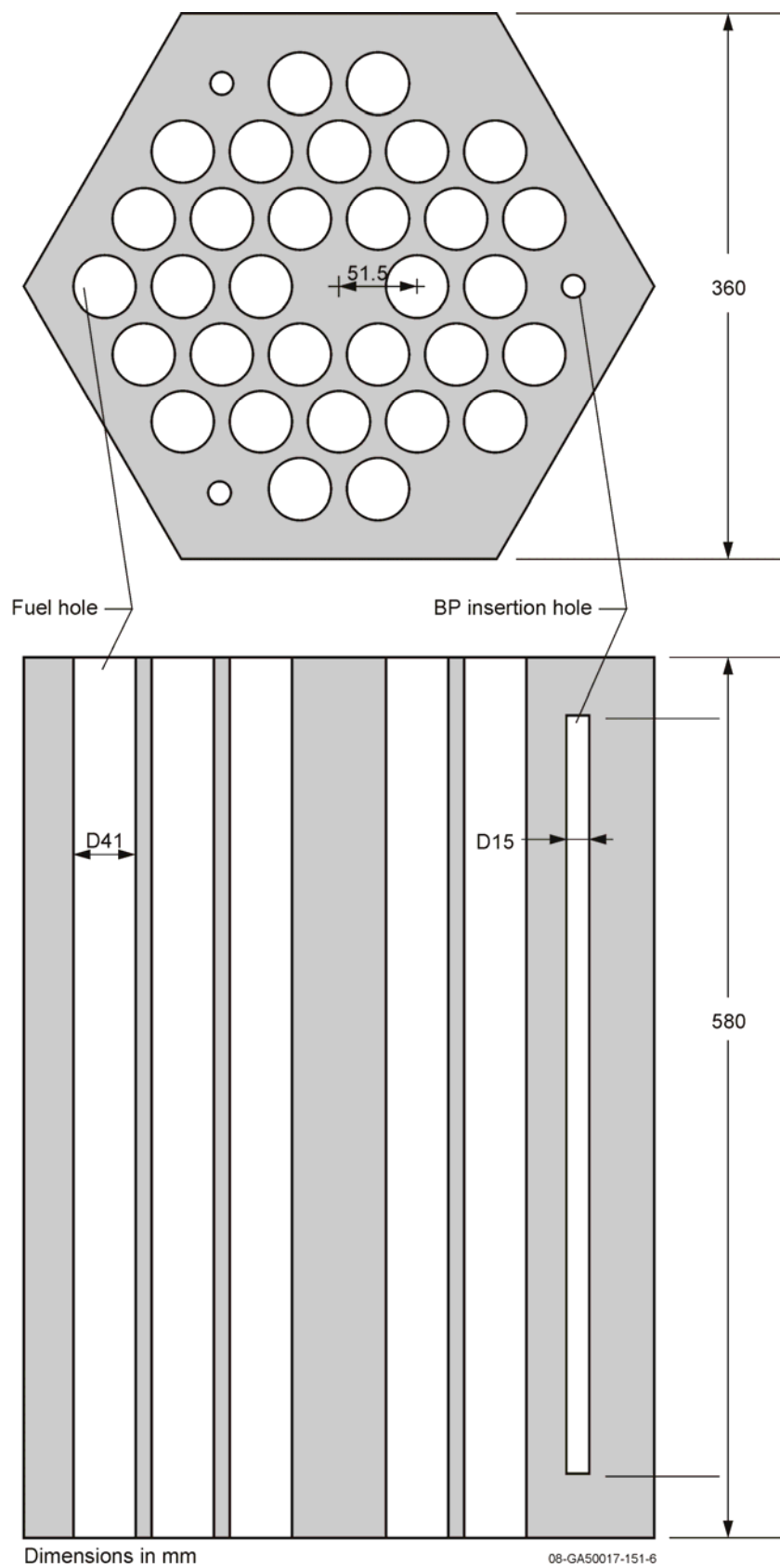


Figure 3.6. Fuel Block for 31-Pin Fuel Assembly.  
Dxx represents the diameter in xx (mm).

### 3.1.2.4 Dummy Blocks

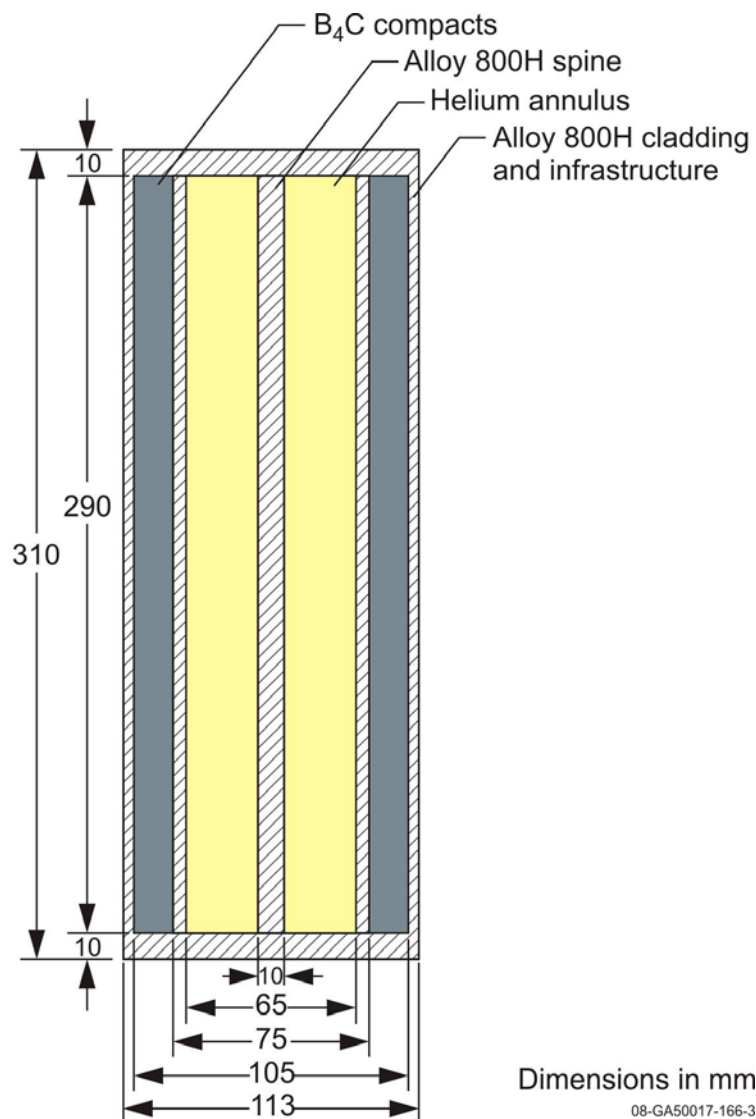
The 30-fuel-column core does not contain dummy blocks.

### 3.1.2.5 Control Rod System

#### **Control Rods**

A diagram of a control rod section is shown in Figure 3.7. The absorber compacts are modeled as a single unit. Detailed dimensions regarding the cladding infrastructure for each section was unavailable, and the clad is therefore modeled without detail. A single control rod is comprised of ten sections (Figure 3.8) with a total height of 3.1 m.

The control rods are divided up into four sets: center position (C), ring 1 (R1), ring 2 (R2), and ring 3 (R3). The center position contains two control rods. The other rings are comprised of six, six, and three positions, containing a total of twelve, twelve, and six control rods, respectively. Control rods in each set are synchronously moved.





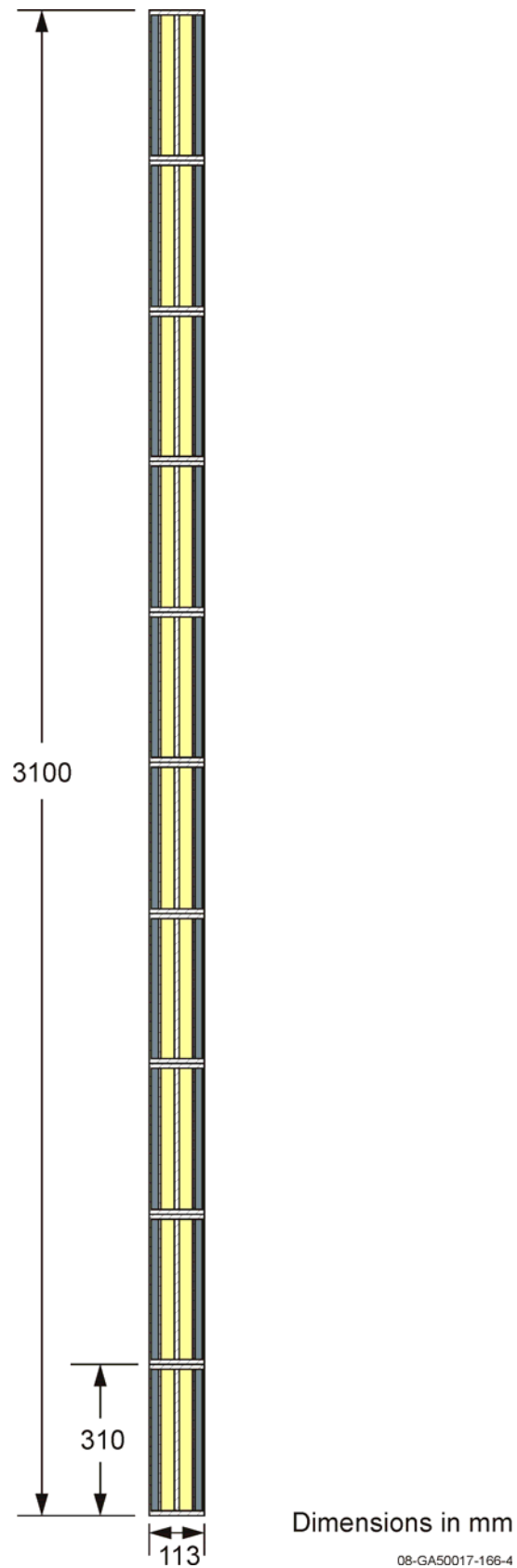


Figure 3.8. Control Rod Comprised of Ten Sections.

**Control Rod Columns**

Individual control rod blocks were not modeled. A single control rod column was modeled with three holes to accommodate two control rods and an empty position (for the reserved shutdown system). A diagram of a generic control column (without control rods) is shown in Figure 3.9. The holes in the control rod and instrumentation columns are equidistant from each other, with an angle of  $120^\circ$ .

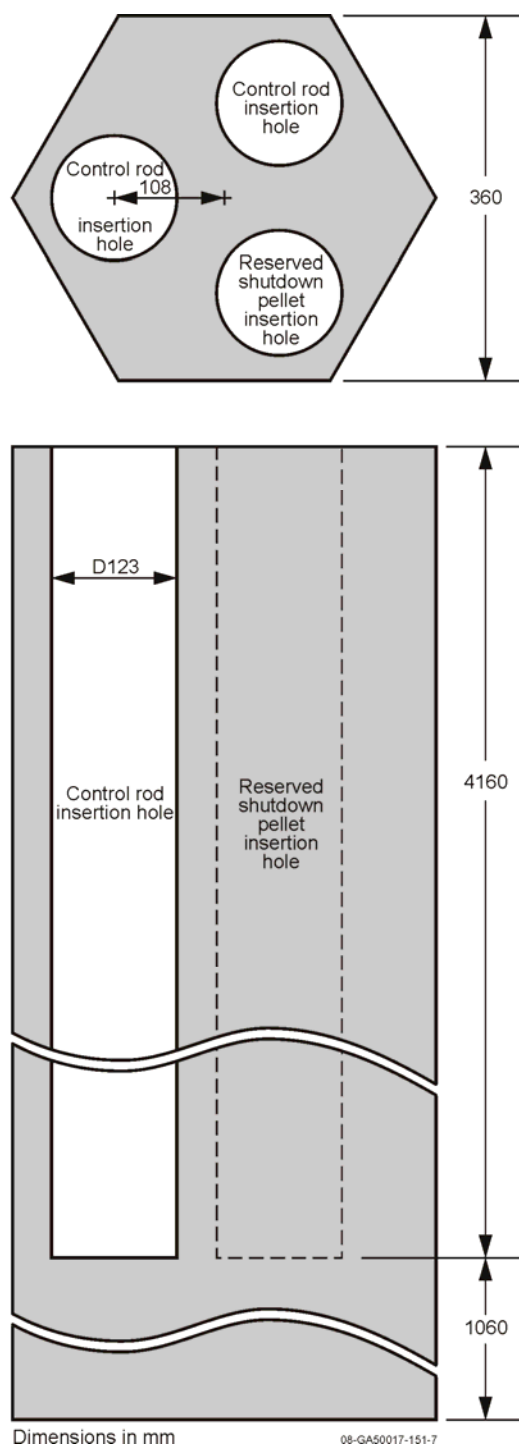


Figure 3.9. Control Rod Column.  
Dxx represents the diameter in xx (mm).

### 3.1.2.6 Instrumentation

#### Instrumentation Components

Insufficient information was available to model instrumentation in the HTTR or evaluate biases related to the removal of said instrumentation from the model.

#### Instrumentation Columns

Instrumentation columns are modeled as a single unit without blocks, similar to the control rod columns. However, all three positions are empty (Figure 3.10).

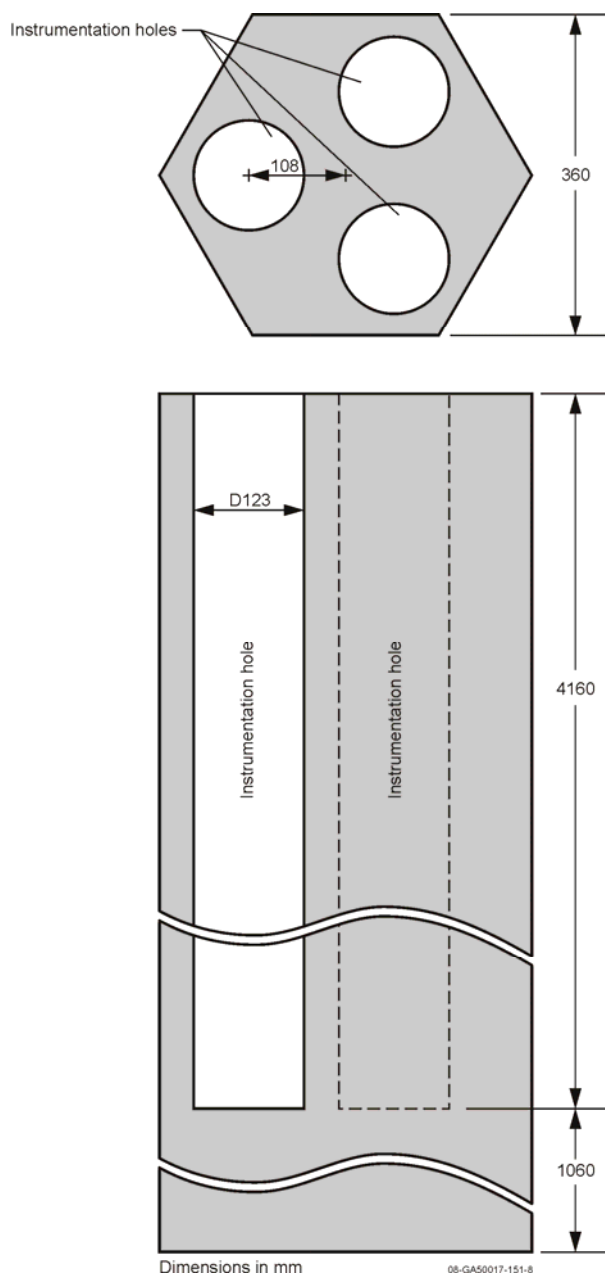


Figure 3.10. Instrumentation Column.  
Dxx represents the diameter in xx (mm).

### 3.1.2.7 Replaceable Reflector Columns

The replaceable reflector columns are modeled as a solid unit and not as individual blocks, similar to the control rod and instrumentation columns but without hollow sections (Figure 3.11). The fuel handling positions, dowels, and sockets were not included in the model but were accounted for with a void fraction of 0.5%; a bias was not assessed.

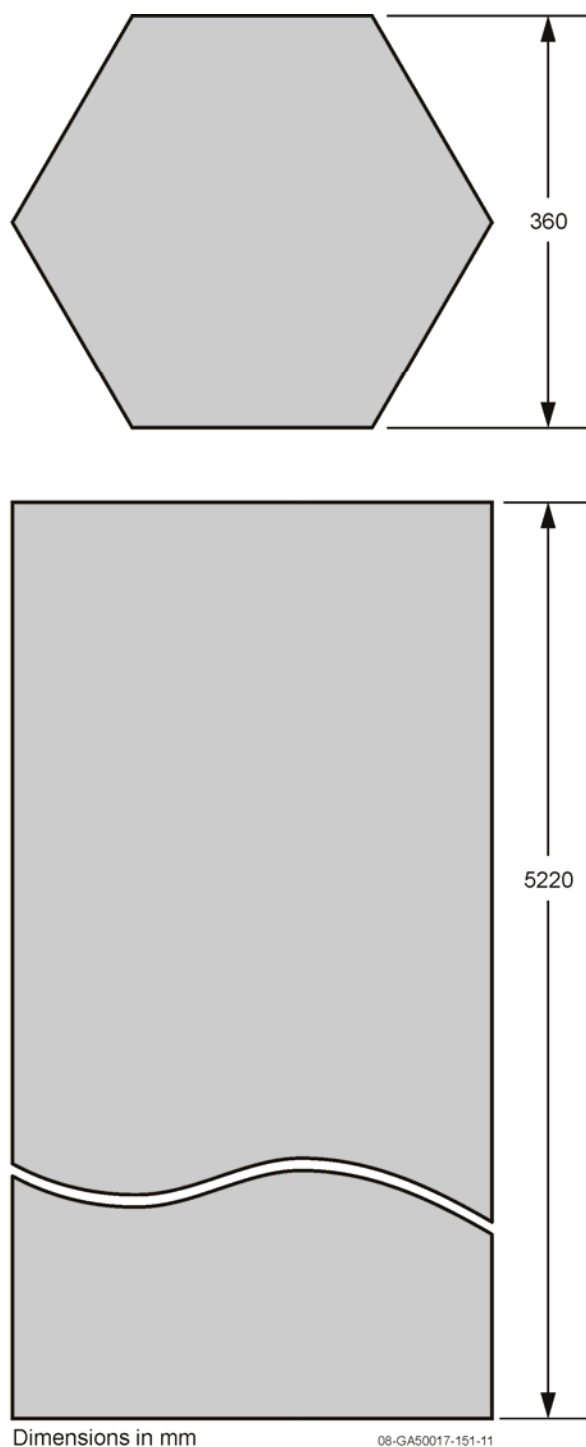


Figure 3.11. Replaceable Reflector Column.

### 3.1.2.8 Replaceable Reflector Blocks in Fuel Columns

The replaceable reflector blocks, located at the top and the bottom fuel columns, are shown in Figures 3.12 and 3.13, for the 33-pin and 31-pin fuel assemblies, respectively. The replaceable reflector blocks have the same regular hexagonal shape and pitch as described for the fuel blocks.

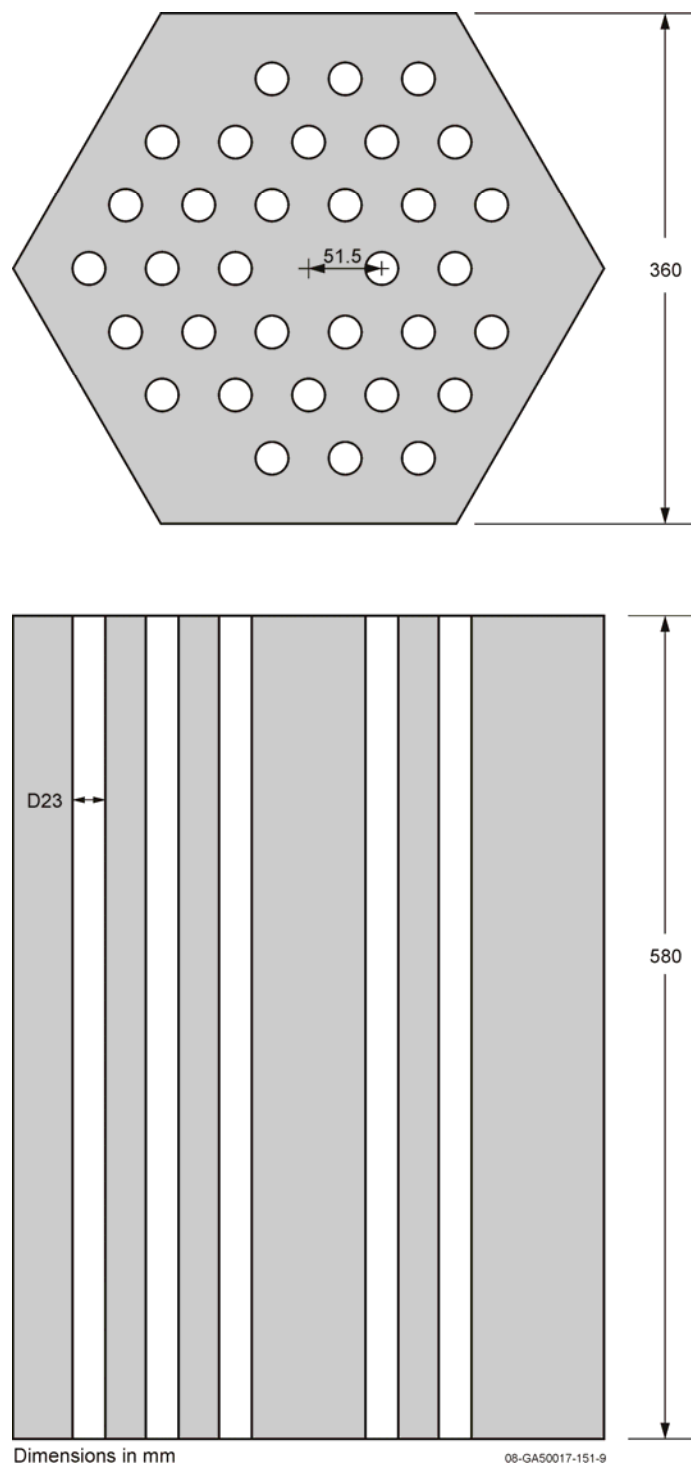
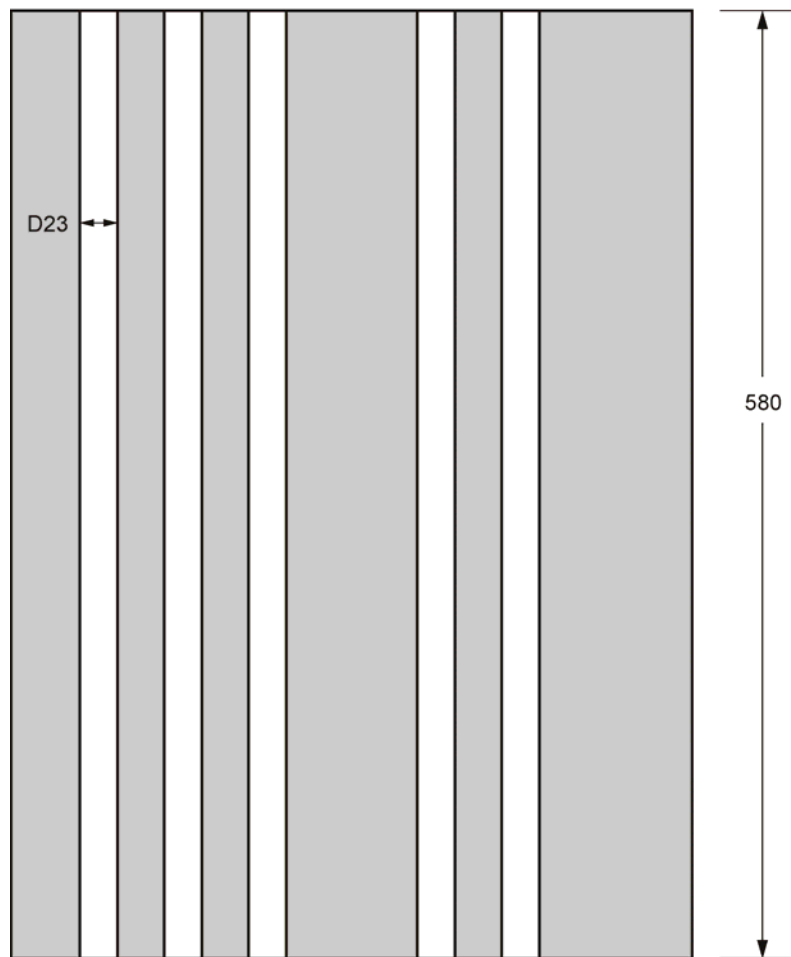
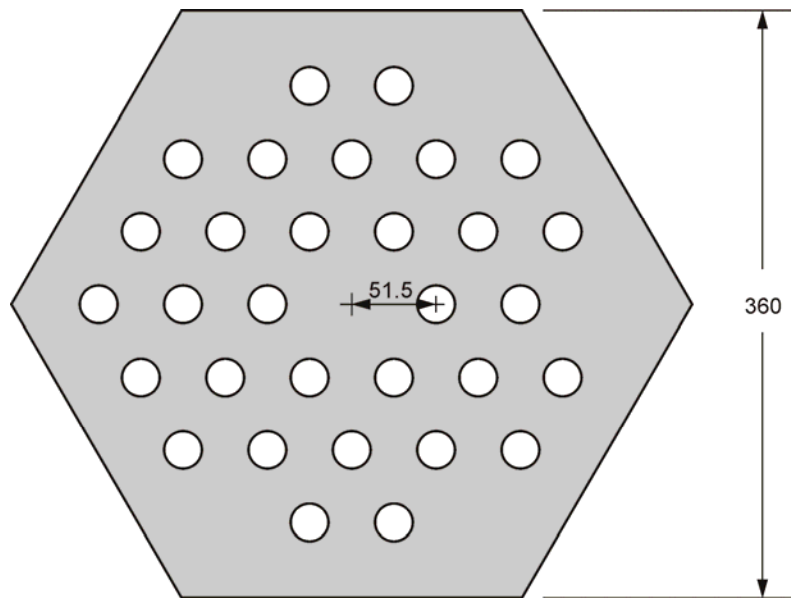


Figure 3.12. Replaceable Reflector Block for 33-Pin Fuel Assembly.  
Dxx represents the diameter in xx (mm).



Dimensions in mm

08-GA50017-151-10

Figure 3.13. Replaceable Reflector Block for 31-Pin Fuel Assembly.  
Dxx represents the diameter in xx (mm).

## 3.1.2.9 Fuel Columns

The fuel columns are separated into four zones (as shown in Figure 1.46). Each zone has a specified pattern of uranium enrichment. Each column contains two top replaceable reflector blocks, five fuel blocks, and two bottom replaceable reflector blocks. The second and third fuel blocks from the top contain burnable poison pellets that are more enriched than the pellets in the other three positions. Figure 3.14 shows the enrichment of the uranium (wt.%) in the TRISO fuel (upper left) and the natural boron content (wt.%) in the burnable pellets (lower right).

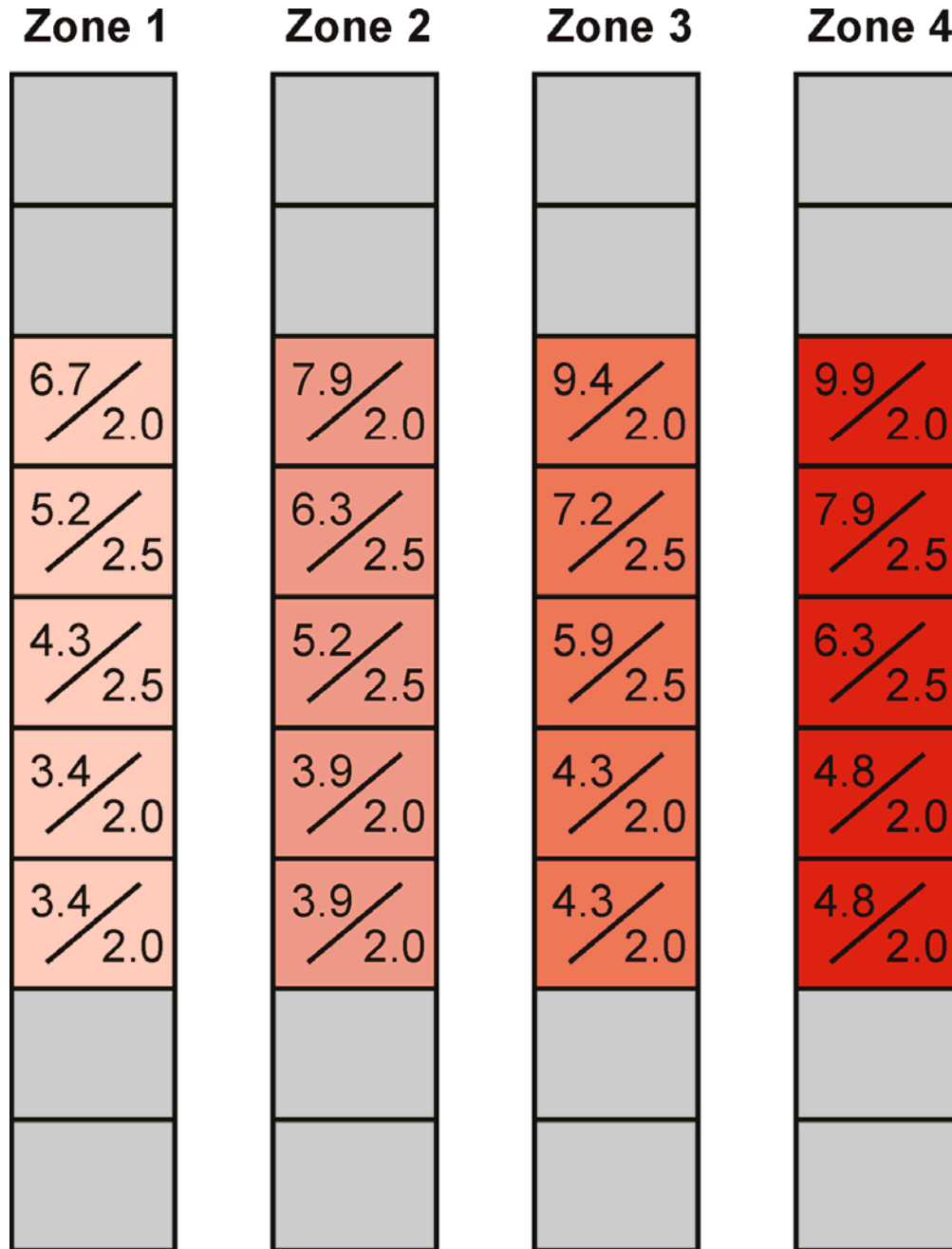


Figure 3.14. HTTR Fuel Zones.

08-GA50017-167-1

### 3.1.2.10 Reactor Core Configuration

The HTTR core configuration is shown in Figures 3.15, 3.16, 3.17. The first figure identifies the positions in the core for a given column type. The second figure provides the orientation of each column within its respective position in the core. The third figure shows the column identification number for each position in the core. Figure 3.18 shows a basic cross section of the HTTR core generated in MCNP.

The permanent reflector surrounding the core has been circularized with a radius of 2125 mm and height of 5220 mm.

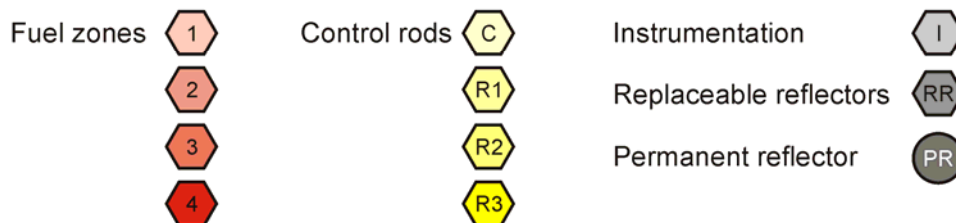
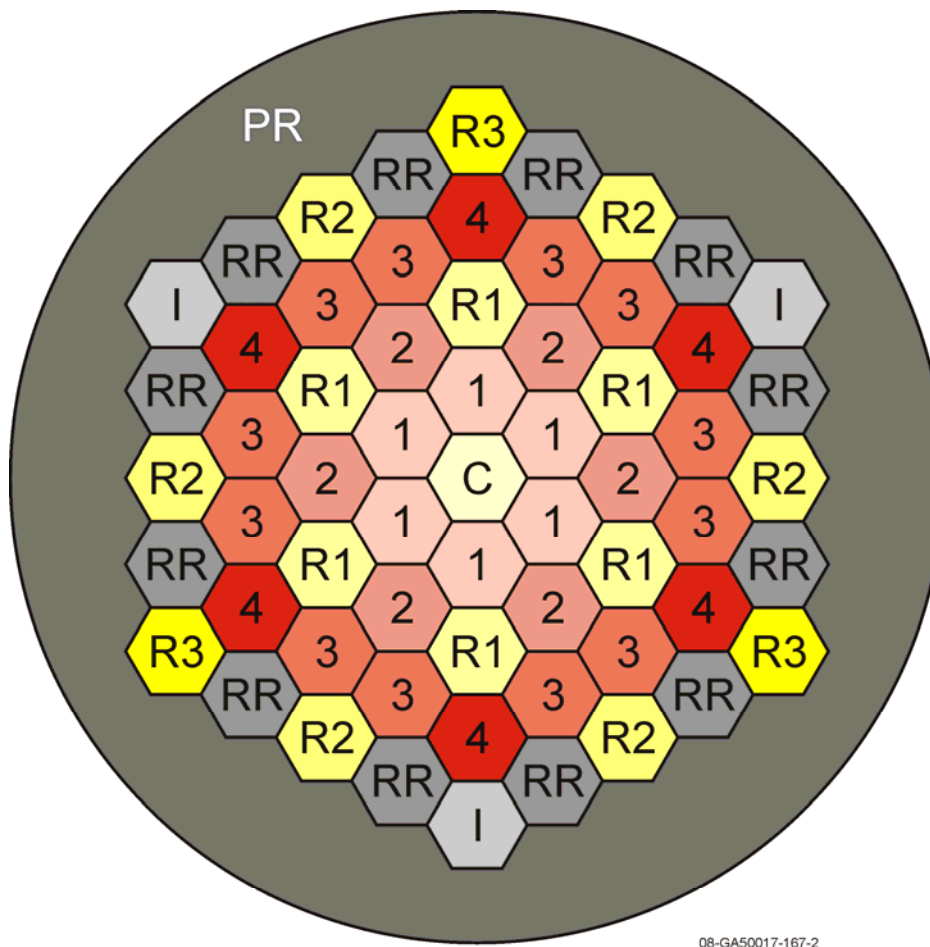


Figure 3.15. HTTR Core Positions.



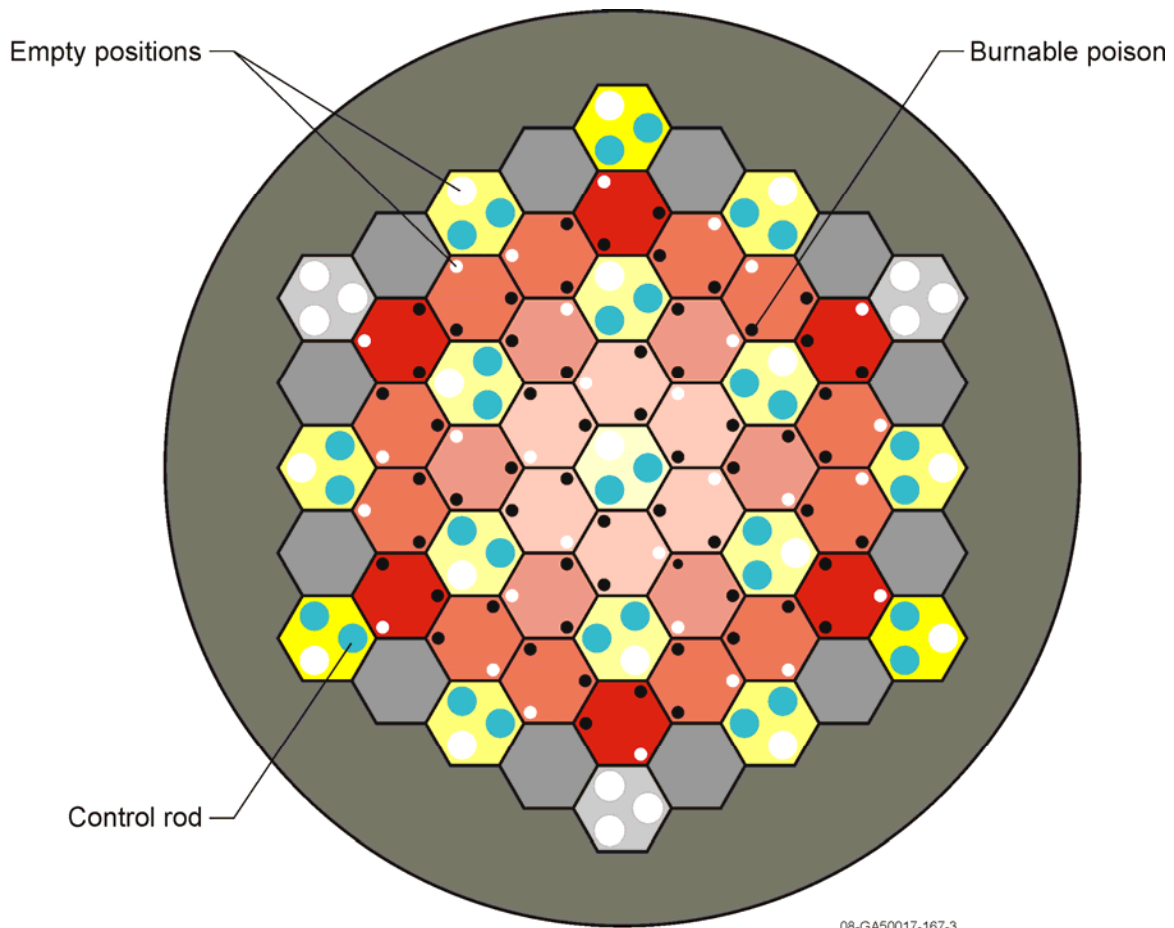


Figure 3.16. Fuel and Control Rod Column Orientations.

08-GA50017-167-3

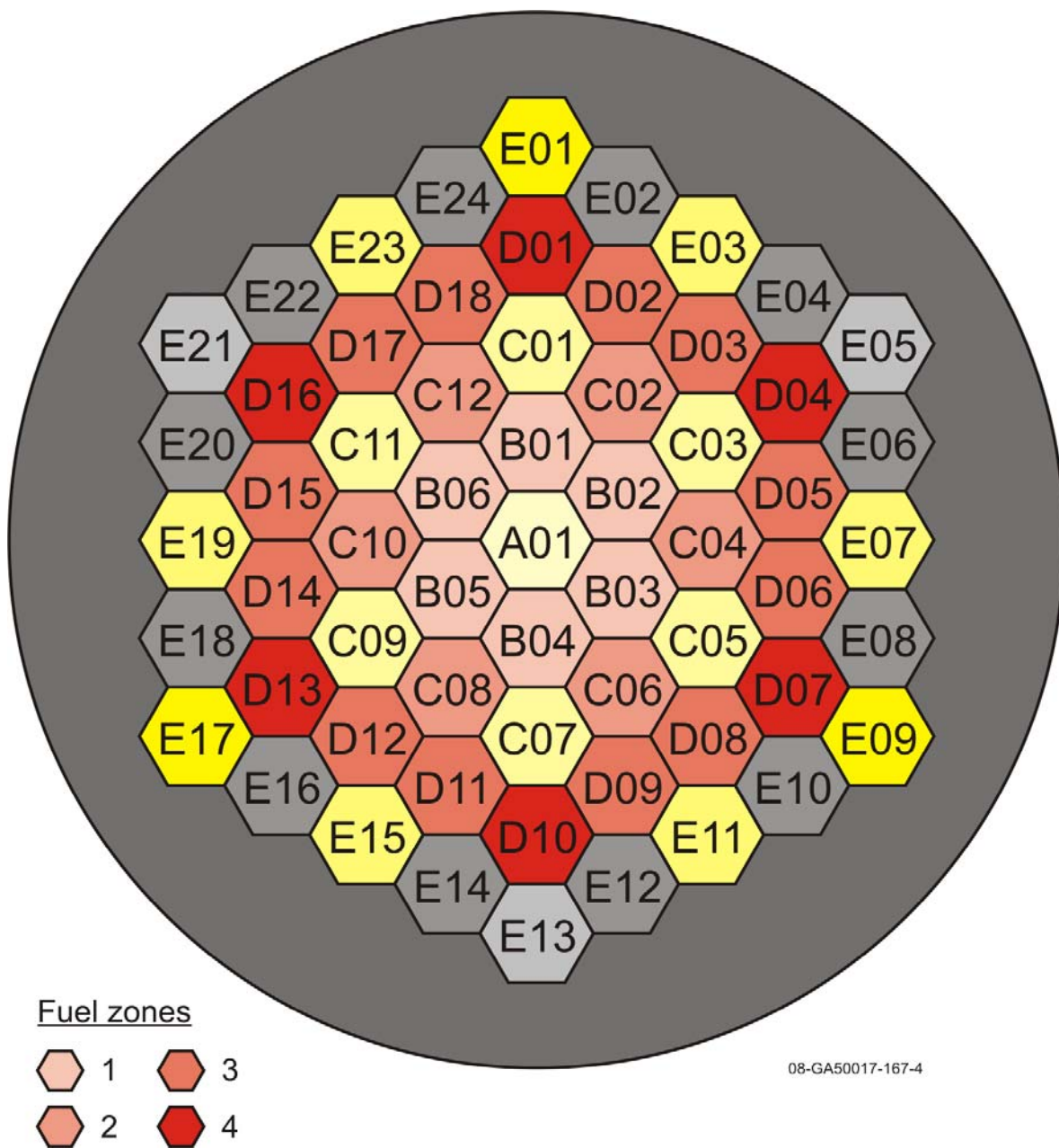


Figure 3.17. HTTR Column Identification.

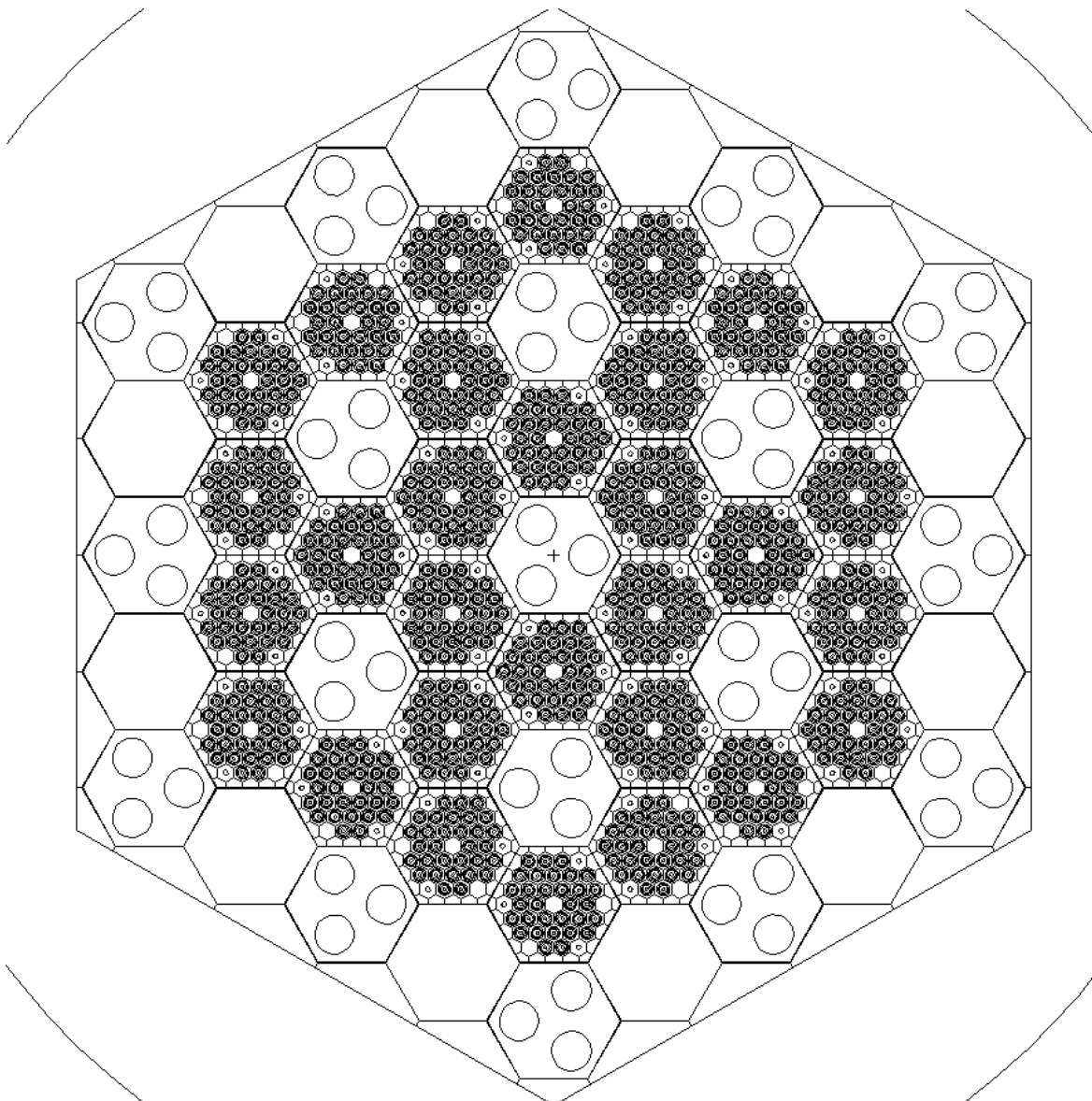


Figure 3.18. Cross Section of the HTTR Core (MCNP Visual Editor).<sup>a</sup>

### 3.1.2.11 Critical Rod Positions

Control rod positions are described with the zero position defined as level with the bottom place of the lowest fuel block (i.e., 1160 mm from the bottom of the core graphite).

The critical rod positions for the 30-fuel-column core are 1775 mm for the C, R1, and R2 positions, and 4049 mm for the R3 positions. All control rod positions are distances above the bottom of the lowest fuel block. Figure 3.19 provides reference between the various column types in the core and the control rod positions.

<sup>a</sup> L. L. Carter and R. A. Schwarz, "MCNP Visual Editor Computer Code Manual for Vised Version 19K,"

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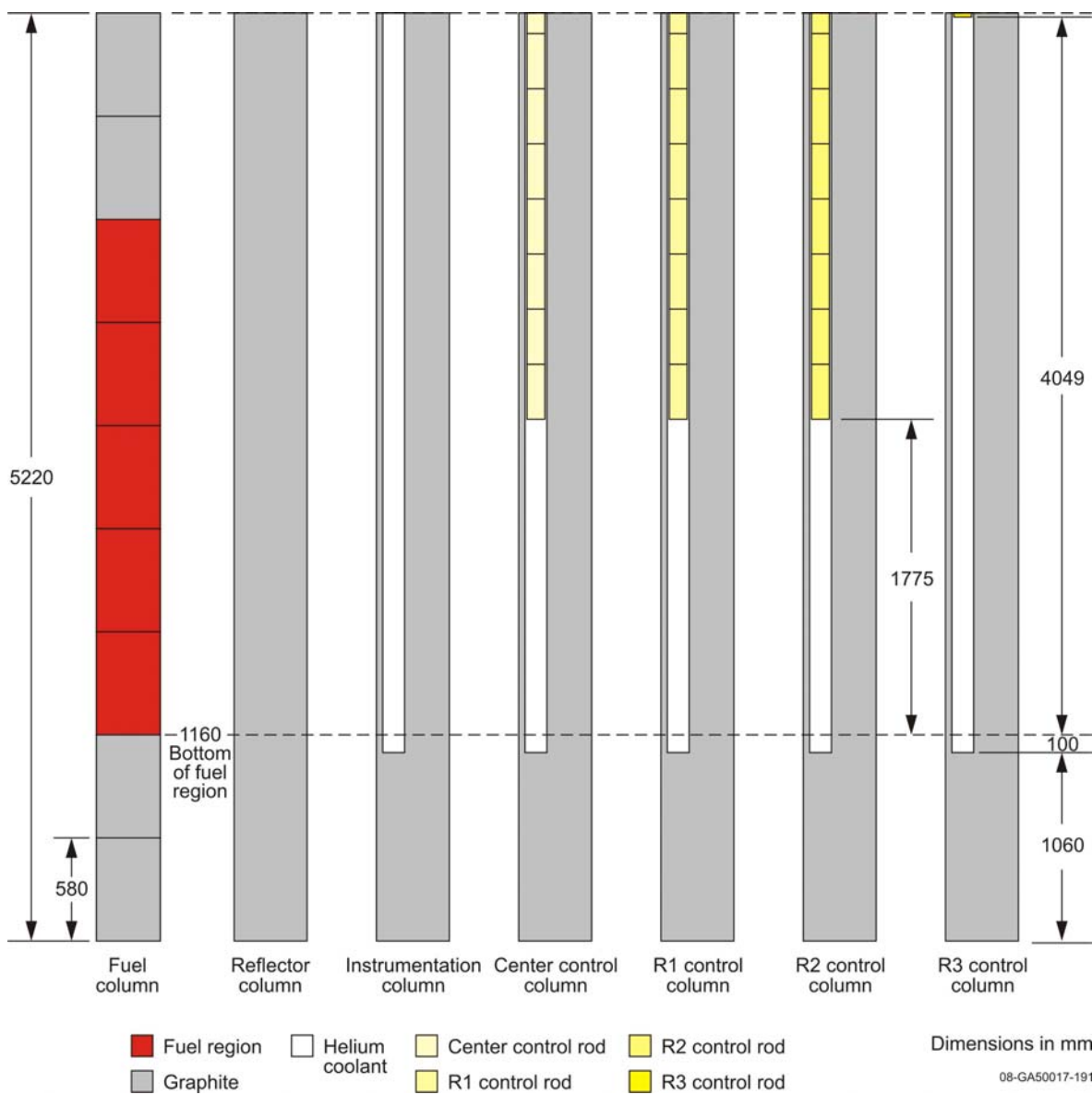
HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

Figure 3.19. Axial Profile of Columns and Control Rod Positions.

**3.1.3 Material Data****3.1.3.1 Pin-in-Block Fuel****TRISO Particles**

The mass density of the TRISO-coated  $\text{UO}_2$  fuel kernels is  $10.39 \text{ g/cm}^3$  with an O/U ratio of 2.00 and an equivalent natural-boron impurity content of 1.5 ppm by weight. A summary of the atomic densities and compositions for the twelve enrichments found throughout the core are provided in Table 3.1.

Table 3.1. Atomic Densities (atoms/b-cm) of the  $\text{UO}_2$  Kernels for Varying Enrichments.

Isotope	3.40 wt.%	3.90 wt.%	4.30 wt.%	4.80 wt.%	5.20 wt.%	5.90 wt.%
$^{10}\text{B}$	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07
O	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02
$^{234}\text{U}$	6.0946E-06	6.9909E-06	7.7079E-06	8.6041E-06	9.3211E-06	1.0576E-05
$^{235}\text{U}$	7.9784E-04	9.1517E-04	1.0090E-03	1.1264E-03	1.2202E-03	1.3845E-03
$^{238}\text{U}$	2.2376E-02	2.2259E-02	2.2165E-02	2.2049E-02	2.1955E-02	2.1792E-02
Total	6.9523E-02	6.9524E-02	6.9526E-02	6.9527E-02	6.9528E-02	6.9530E-02

Table 3.1 (cont.). Atomic Densities (atoms/b-cm) of the  $\text{UO}_2$  Kernels for Varying Enrichments.

Isotope	6.30 wt.%	6.70 wt.%	7.20 wt.%	7.90 wt.%	9.40 wt.%	9.90 wt.%
$^{10}\text{B}$	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07	1.7276E-07
O	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02	4.6343E-02
$^{234}\text{U}$	1.1293E-05	1.2010E-05	1.2906E-05	1.4161E-05	1.6850E-05	1.7746E-05
$^{235}\text{U}$	1.4783E-03	1.5722E-03	1.6895E-03	1.8538E-03	2.2058E-03	2.3231E-03
$^{238}\text{U}$	2.1699E-02	2.1605E-02	2.1489E-02	2.1325E-02	2.0975E-02	2.0858E-02
Total	6.9532E-02	6.9533E-02	6.9534E-02	6.9536E-02	6.9541E-02	6.9542E-02

The material properties of the TRISO layers and graphite overcoat are provided in Table 3.2.

Table 3.2. Material Properties of the TRISO Coatings and Graphite Overcoat.

Property	Buffer	IPyC	SiC	OPyC	Overcoat
Mass Density (g/cm <sup>3</sup> )	1.1	1.85	3.2	1.85	1.7
B-nat Impurity (wppm)	1.5	1.5	1.5	1.5	1.5
Atomic Density (atoms/b-cm)	5.5153E-02	9.2758E-02	1.1453E-01	9.2758E-02	8.5237E-02
<sup>10</sup> B	1.8290E-08	3.0761E-08	5.3208E-08	3.0761E-08	2.8267E-08
C-nat	5.5153E-02	9.2758E-02	8.0223E-02	9.2758E-02	8.5237E-02
Si	--	--	3.4307E-02	--	--

### **Compacts**

The mass density of the fuel compact graphite matrix is 1.7 g/cm<sup>3</sup> with an equivalent natural-boron impurity content of 0.82 ppm by weight. The atomic density and composition of the compact matrix is shown in Table 3.3.

Table 3.3. Atomic Densities of the Fuel Compact Graphite Matrix.

Isotope	Atoms/b-cm
<sup>10</sup> B	1.5452E-08
C-nat	8.5237E-02
Total	8.5237E-02

### **Fuel Element**

The IG-110 graphite sleeve and end caps for the fuel pins have a mass density of 1.77 g/cm<sup>3</sup> and an equivalent natural-boron content of 0.37 ppm by weight. The atomic density and composition of the graphite used in the fuel element is shown in Table 3.4.

Table 3.4. Atomic Densities of the Graphite Fuel Sleeve.

Isotope	Atoms/b-cm
<sup>10</sup> B	7.2596E-09
C-nat	8.8747E-02
Total	8.8747E-02

### 3.1.3.2 Burnable Poisons

The burnable poison pellets have a mass density of 1.80 g/cm<sup>3</sup>; a summary of the atomic densities and compositions for the two natural-boron concentrations employed in the core are provided in Table 3.5. The mass density of the graphite disks used to separate the burnable poison pellets is 1.77 g/cm<sup>3</sup> with an equivalent natural-boron content of 0.37 ppm by weight. The atomic density and composition of the graphite disks is also found in Table 3.5

Table 3.5. Atomic Densities (atoms/b-cm) of the Burnable Poison Pellets and Graphite Disks.

Isotope	2.00 wt.%	2.50 wt.%	Disks
<sup>10</sup> B	3.9906E-04	4.9882E-04	7.2596E-09
<sup>11</sup> B	1.6063E-03	2.0078E-03	--
C-nat	8.8446E-02	8.7995E-02	8.8747E-02
Total	9.0451E-02	9.0501E-02	8.8747E-02

### 3.1.3.3 Fuel Blocks

The IG-110 graphite fuel blocks have a mass density of 1.7512 g/cm<sup>3</sup> (1.76 g/cm<sup>3</sup> base density decreased by a 0.5% void fraction) and an equivalent natural-boron content of 0.4 ppm by weight. The atomic density and composition of the graphite fuel blocks is shown in Table 3.6.

Table 3.6. Atomic Densities of the Graphite Fuel Blocks.

Isotope	Atoms/b-cm
<sup>10</sup> B	7.7648E-09
C-nat	8.7804E-02
Total	8.7804E-02

### 3.1.3.4 Dummy Blocks

The 30-fuel-column core does not contain dummy blocks.

### 3.1.3.5 Control Rod System

**Control Rods**

The absorber compacts have a mass density of  $1.9 \text{ g/cm}^3$  and have a composition and atomic density as described in Table 3.7.

Table 3.7. Atomic Densities of the Absorber Compacts.

Isotope	Atoms/b-cm
$^{10}\text{B}$	6.8220E-03
$^{11}\text{B}$	2.4974E-02
C-nat	6.6685E-02
Total	6.6685E-02

The Alloy 800H cladding of the control rods has a mass density of  $8.03 \text{ g/cm}^3$  with a composition and atomic density as shown in Table 3.8.

Table 3.8. Atomic Densities of the Alloy 800H Clad.

Isotope	Atoms/b-cm
C-nat	3.2210E-04
Al	6.7209E-04
Si	6.0263E-04
P	3.1225E-05
S	1.5081E-05
Ti	3.7884E-04
Cr	1.9530E-02
Mn	8.8022E-04
Fe	3.8092E-02
Ni	2.6777E-02
Cu	2.2830E-04
Total	8.7530E-02

**Control Rod Columns**

The IG-110 graphite fuel columns are modeled with the same physical properties as the fuel blocks in Section 3.1.3.3 and Table 3.6.



### 3.1.3.6 Instrumentation

#### **Instrumentation Components**

Insufficient information was available to model instrumentation in the HTTR or evaluate biases related to the removal of said instrumentation from the model.

#### **Instrumentation Columns**

The IG-110 graphite instrumentation columns are modeled with the same physical properties as the fuel blocks in Section 3.1.3.3 and Table 3.6.

### 3.1.3.7 Replaceable Reflector Columns

The IG-110 replaceable reflector columns are modeled with the same physical properties as the fuel blocks in Section 3.1.3.3 and Table 3.6.

### 3.1.3.8 Replaceable Reflectors Blocks in Fuel Columns

The IG-110 replaceable reflector blocks are modeled with the same physical properties as the fuel blocks in Section 3.1.3.3 and Table 3.6.

### 3.1.3.9 Permanent Reflector

The PGX graphite permanent reflector has a mass density of 1.71789 g/cm<sup>3</sup> (1.76 g/cm<sup>3</sup> base density decreased by a 0.7% void fraction) and an equivalent natural-boron content of 1.91 ppm by weight. The atomic density and composition of the permanent reflector is shown in Table 3.9.

Table 3.9. Atomic Densities of Permanent Reflector.

Isotope	Atoms/b-cm
<sup>10</sup> B	3.6372E-08
C-nat	8.6134E-02
Total	8.6134E-02

### 3.1.3.10 Helium Coolant

The helium coolant has an atomic density of 2.4616E-05 atoms/b-cm (mass density of  $1.6361 \times 10^{-4}$  g/cm<sup>3</sup>). No impurities are modeled in the coolant.

## 3.1.4 Temperature Data

The benchmark model temperature is 20.5 °C.

## 3.1.5 Experimental and Benchmark-Model $k_{\text{eff}}$ and / or Subcritical Parameters

The experimental  $k_{\text{eff}}$  was approximately 1.0000, made to delayed critical. A comprehensive bias assessment could not be performed; therefore, the experimental  $k_{\text{eff}}$  was not adjusted. Furthermore, the uncertainty in the benchmark model is the same as the uncertainty evaluated for the experimental model.

The unbiased benchmark model  $k_{\text{eff}}$  is 1.00000 with an evaluated uncertainty between -0.00606 and +0.00703.

### **3.2 Benchmark-Model Specifications for Buckling and Extrapolation-Length Measurements**

Buckling and extrapolation length measurements were not made.

### **3.3 Benchmark-Model Specifications for Spectral Characteristics Measurements**

Spectral characteristics measurements were not made.

### **3.4 Benchmark-Model Specifications for Reactivity Effects Measurements**

Reactivity effects measurements were not evaluated.

### **3.5 Benchmark-Model Specifications for Reactivity Coefficient Measurements**

Reactivity coefficient measurements were not evaluated.

### **3.6 Benchmark-Model Specifications for Kinetics Measurements**

Kinetics measurements were not evaluated.

### **3.7 Benchmark-Model Specifications for Reaction-Rate Distribution Measurements**

Reaction-rate distribution measurements were not evaluated.

### **3.8 Benchmark-Model Specifications for Power Distribution Measurements**

Power distribution measurements were not made.

### **3.9 Benchmark-Model Specifications for Isotopic Measurements**

Isotopic measurements were not made.

### **3.10 Benchmark-Model Specifications for Other Miscellaneous Types of Measurements**

Other miscellaneous types of measurements were not made.

## 4.0 RESULTS OF SAMPLE CALCULATIONS

### 4.1 Results of Calculations of the Critical or Subcritical Configurations

Random particles cannot be easily modeled in MCNP. Therefore an ordered and uniformly-filled lattice approach for modeling the benchmark was implemented. Results are provided for both cases. It is up to the user to determine which method or model is most appropriate for analyzing this benchmark; the effective impact on the reactivity will then need to be accounted for.

Currently the calculations using the benchmark model calculate a  $k_{\text{eff}}$  2 % greater than the experimental benchmark value. It is believed that a systematic bias causes this difference, which is on the order of  $\geq 3\sigma$  from the expected benchmark value. It is currently difficult to obtain the necessary information to improve the confidence in the benchmark model; the necessary data is controlled because the configuration of the HTTR analyzed in this benchmark is the same core currently in operation. Once this information is made available, the HTTR benchmark can be adjusted as appropriate.

#### 4.1.1 Ordered TRISO Lattice within the Fuel Compacts

The TRISO particles are modeled in rectangular lattices with the dimensions of 0.106 cm (length)  $\times$  0.106 cm (width)  $\times$  0.1 cm (height) to generate a volumetric packing fraction of 30 % (not including the graphite overcoat) when only complete particles are placed within the compact. A cross-sectional view of the TRISO lattice block is shown in Figure 4.1. The graphite overcoat isn't completely represented in the lattice.

A horizontal cross section of the compacts is shown in Figure 4.2. As can be seen, selective placement of TRISO particles was necessary to obtain the desired packing fraction of 30 vol. %. For the current configuration, 12,987 TRISO particles are present within a standard fuel compact; this value is close to the reported value of approximately 13,000.

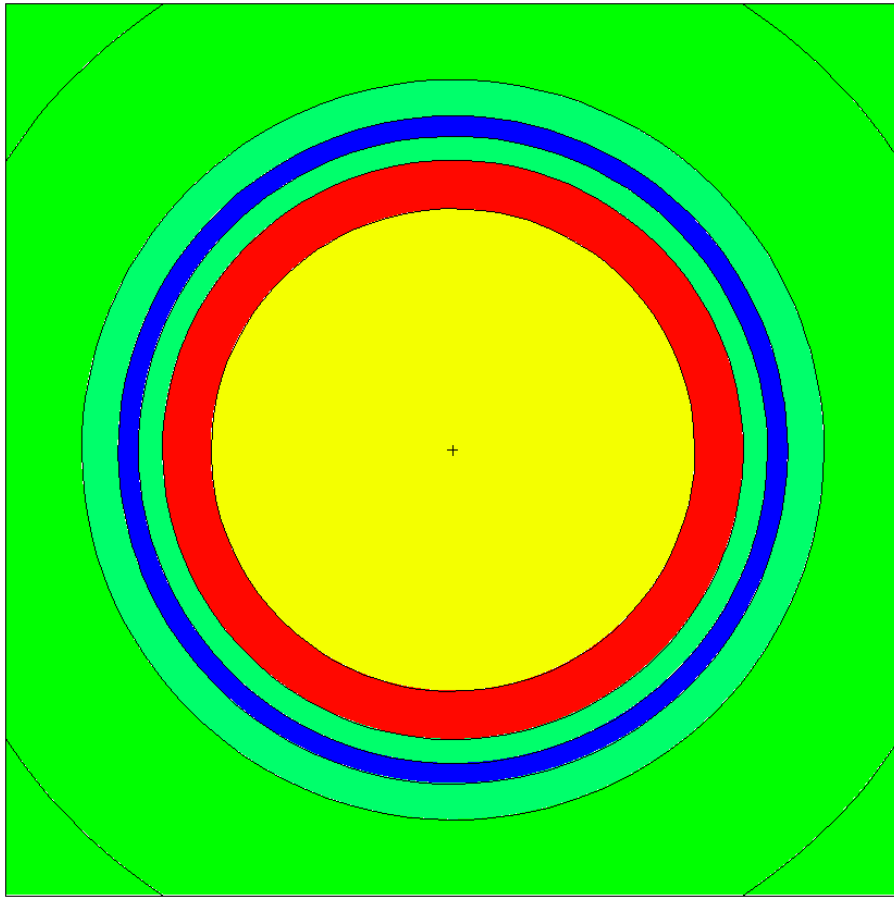


Figure 4.1. MCNP TRISO Lattice Unit Cell.

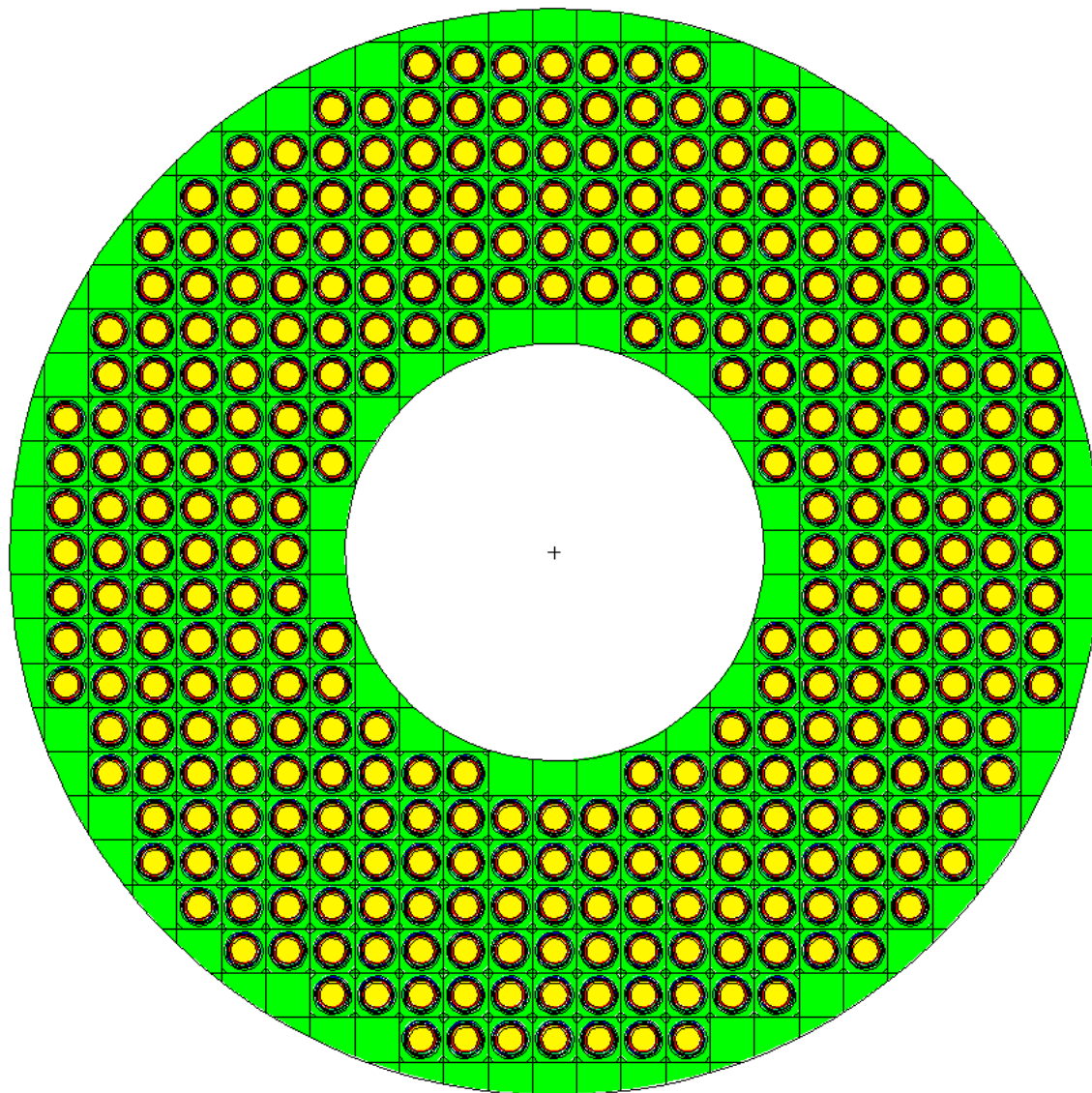


Figure 4.2. MCNP Ordered TRISO Lattice within the Fuel Compacts.

The computed  $k_{\text{eff}}$  values for the benchmark model of the fully-loaded 30-fuel-column core evaluated with MCNP using the ENDF/B-V.2, -VI.8, and -VII.0, JEFF-3.1, and JENDL-3.3 cross section libraries are listed in Table 4.1. All benchmark model calculations are compared against the expected benchmark value reported in Section 3.1.5. The total uncertainty in the expected value of  $k_{\text{eff}}$  is taken from Section 2.1.6. Included in the reported results is a comparison of the ENDF/B-VI.8 analysis without accounting for the neutron cross-section adjustment for neutron upscatter at thermal energies and scattering data for elements bound in materials. This correction is referred to as  $S(\alpha, \beta)$ . A JENDL-3.3 analysis was also performed with the inclusion of ENDF/B-VI.8 thermal neutron scattering data to demonstrate its importance in the prediction of  $k_{\text{eff}}$ . The ENDF/B-VI.8 benchmark model was also analyzed using MCNPX. The MCNP5 calculations were performed with 3,550 generations (skipping the first 50) and 10,000 neutrons per generation.

Calculated values of  $k_{\text{eff}}$  differ from the benchmark model values by more than  $3\sigma$ , except for the case using the JENDL-3.3 library with ENDF/B-VI.8 thermal scattering treatment. This latter case is within  $3\sigma$  of the benchmark  $k_{\text{eff}}$ . Properly accounting for all biases in the model will not fix the discrepancy. It

is most likely that the content of neutron absorbing impurities in the graphite blocks is greater than actually reported. Increased impurity content would significantly decrease the calculated values for  $k_{\text{eff}}$ . Methods for accurately determining the equivalent boron content in graphite have evolved since the original construction of the HTTR. Reevaluation of the core graphite might be necessary to improve the quality of this benchmark.

Table 4.1. Final Results for the HTTR Benchmark Model Evaluation using an Ordered Lattice.

Neutron Cross-Section Library	Calculated			Benchmark			(C-E)/E (%)
	$k_{\text{eff}}$	$\pm$	$\sigma$	$k_{\text{eff}}$	$-\sigma$ (%)	$+\sigma$ (%)	
ENDF/B-V.2	1.02327	$\pm$	0.00013	1.00000 <sup>(a)</sup>	0.606	0.703	2.327
ENDF/B-VI.8	1.02532	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.532
ENDF/B-VI.8 <sup>(b)</sup> [no S( $\alpha,\beta$ )]	1.04329	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	4.329
END/B-VII.0	1.02600	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.600
JEFF-3.1	1.02711	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.711
JENDL-3.3	1.03981	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	3.981
JENDL-3.3 with ENDF/B-VI.8 S( $\alpha,\beta$ ) <sup>(b)</sup>	1.02160	$\pm$	0.00013	1.00000 <sup>(a)</sup>	0.606	0.703	2.160
ENDF/B-VI.8 (MCNPX)	1.02542	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.542

(a) No biases have been currently evaluated for correcting the expected experimental  $k_{\text{eff}}$ .

(b) The JENDL-3.3 library did not include additional cross-section data for thermal scattering treatment. Cases were performed for ENDF/B-VI.8 without thermal neutron scattering correction to compare with the JENDL-3.3 results, and JENDL-3.3 was also analyzed using thermal neutron scattering correction from the ENDF/B-VI.8 library.

#### 4.1.2 Uniformly-Filled TRISO Lattice within the Fuel Compacts

The TRISO particles are modeled in rectangular lattices with the dimensions of 0.1107678 cm (length)  $\times$  0.1107678 cm (width)  $\times$  0.10910802 cm (height) to generate a volumetric packing fraction of approximately 30 % (not including the graphite overcoat) while conserving mass. A cross-sectional view of the TRISO lattice block is very similar to that already shown in Figure 4.1. The graphite overcoat isn't completely represented in the lattice.

A horizontal cross section of the compacts is shown in Figure 4.3. As can be seen, a disadvantage of using a uniformly-filled lattice is the inclusion of partial TRISO particles. For the current configuration, an equivalent of 12,987 TRISO particles are present within a standard fuel compact; this value is close to the reported value of approximately 13,000. The packing fraction is 30.5 vol. %.

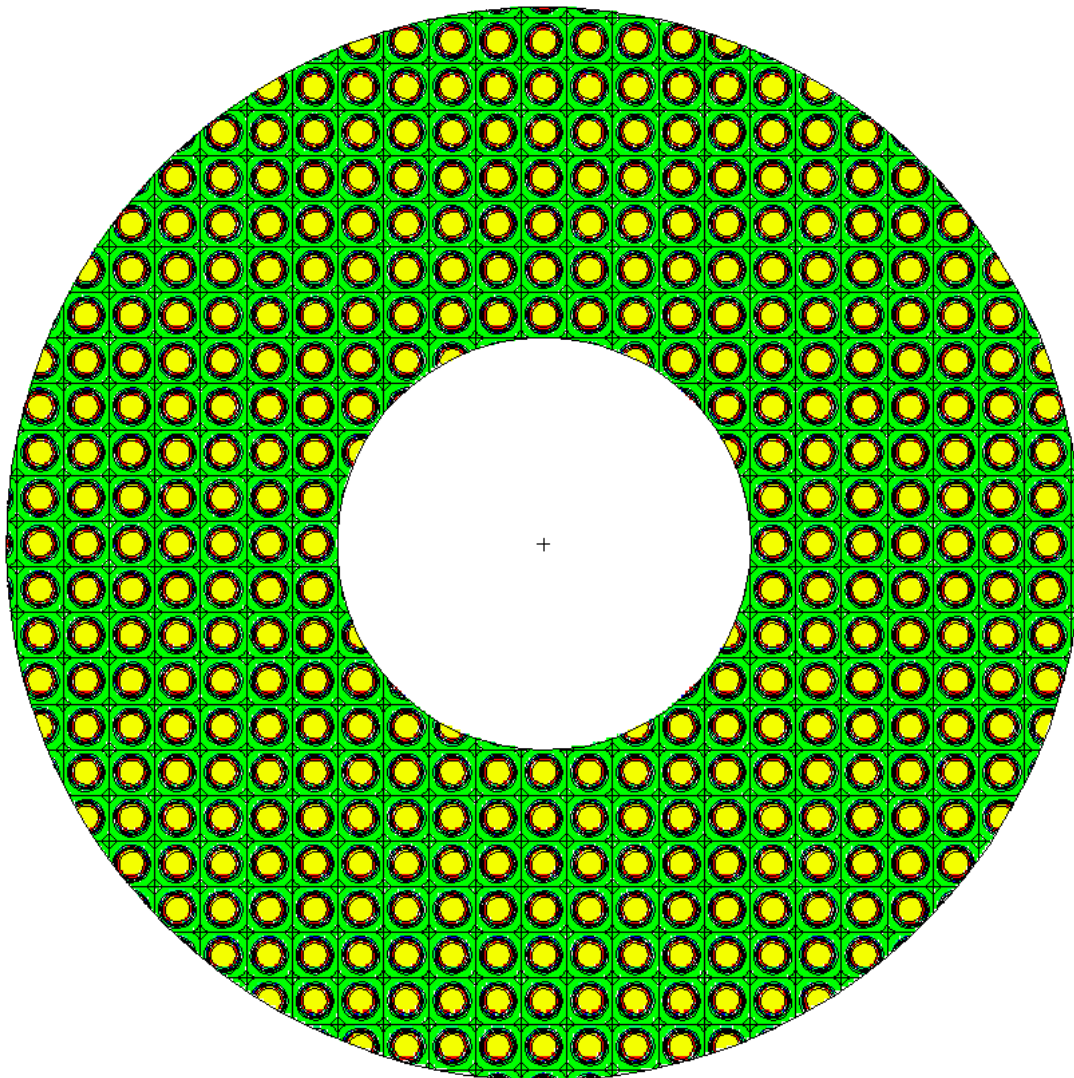


Figure 4.3. MCNP Uniformly-Filled TRISO Lattice within the Fuel Compacts.

The computed  $k_{\text{eff}}$  values for the benchmark model of the fully-loaded 30-fuel-column core evaluated with MCNP using the ENDF/B-V.2, -VI.8, and -VII.0, JEFF-3.1, and JENDL-3.3 cross section libraries are listed in Table 4.2. All benchmark model calculations are compared against the expected benchmark value reported in Section 3.1.5. The total uncertainty in the expected value of  $k_{\text{eff}}$  is taken from Section 2.1.6. Included in the reported results is a comparison of the ENDF/B-VI.8 analysis without accounting for the neutron cross-section adjustment for neutron upscatter at thermal energies and scattering data for elements bound in materials. This correction is referred to as  $S(\alpha, \beta)$ . A JENDL-3.3 analysis was also performed with the inclusion of ENDF/B-VI.8 thermal neutron scattering data to demonstrate its importance in the prediction of  $k_{\text{eff}}$ . The ENDF/B-VI.8 benchmark model was also analyzed using MCNPX. The MCNP5 calculations were performed with 3,550 generations (skipping the first 50) and 10,000 neutrons per generation.

Calculated values of  $k_{\text{eff}}$  differ from the benchmark model values by more than  $3\sigma$ , except for the case using the JENDL-3.3 library with ENDF/B-VI.8 thermal scattering treatment. This latter case is within  $3\sigma$  of the benchmark  $k_{\text{eff}}$ . Properly accounting for all biases in the model might reduce this difference

some. It is most likely that the content of neutron absorbing impurities in the graphite blocks is greater than actually reported. Increased impurity content would significantly decrease the calculated values for  $k_{\text{eff}}$ . Methods for accurately determining the equivalent boron content in graphite have evolved since the original construction of the HTTR. Reevaluation of the core graphite might be necessary to improve the quality of this benchmark. Generally, results in Table 4.2 are approximately 0.2 %  $\Delta k$  lower than those in Table 4.1.

Table 4.2. Final Results for the HTTR Benchmark Model Evaluation using a Uniformly-Filled Lattice.

Neutron Cross-Section Library	Calculated			Benchmark			(C-E)/E (%)
	$k_{\text{eff}}$	$\pm$	$\sigma$	$k_{\text{eff}}$	$-\sigma$ (%)	$+\sigma$ (%)	
ENDF/B-V.2	1.02305	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.305
ENDF/B-VI.8	1.02367	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.367
ENDF/B-VI.8 <sup>(b)</sup> [no S( $\alpha,\beta$ )]	1.04142	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	4.142
END/B-VII.0	1.02418	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	2.418
JEFF-3.1	1.02521	$\pm$	0.00013	1.00000 <sup>(a)</sup>	0.606	0.703	2.521
JENDL-3.3	1.03781	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	3.781
JENDL-3.3 with ENDF/B-VI.8 S( $\alpha,\beta$ ) <sup>(b)</sup>	1.01995	$\pm$	0.00014	1.00000 <sup>(a)</sup>	0.606	0.703	1.995
ENDF/B-VI.8 (MCNPX)	1.02363	$\pm$	0.00013	1.00000 <sup>(a)</sup>	0.606	0.703	2.363

(a) No biases have been currently evaluated for correcting the expected experimental  $k_{\text{eff}}$ .

(b) The JENDL-3.3 library did not include additional cross-section data for thermal scattering treatment. Cases were performed for ENDF/B-VI.8 without thermal neutron scattering correction to compare with the JENDL-3.3 results, and JENDL-3.3 was also analyzed using thermal neutron scattering correction from the ENDF/B-VI.8 library.

#### 4.2 Results of Buckling and Extrapolation Length Calculations

Buckling and extrapolation length measurements were not made.

#### 4.3 Results of Spectral-Characteristics Calculations

Spectral characteristics measurements were not made.

#### 4.4 Results of Reactivity-Effects Calculations

Reactivity effects measurements were not evaluated.

#### 4.5 Results of Reactivity Coefficient Calculations

Reactivity coefficient measurements were not evaluated.



#### **4.6    Results of Kinetics Parameter Calculations**

Kinetics measurements were not evaluated.

#### **4.7    Results of Reaction-Rate Distribution Calculations**

Reaction-rate distribution measurements were not evaluated.

#### **4.8    Results of Power Distribution Calculations**

Power distribution measurements were not made.

#### **4.9    Results of Isotopic Calculations**

Isotopic measurements were not made.

#### **4.10   Results of Calculations for Other Miscellaneous Types of Measurements**

Other miscellaneous types of measurements were not made.

**5.0 REFERENCES**

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**APPENDIX A: COMPUTER CODES, CROSS SECTIONS, AND TYPICAL INPUT LISTINGS****A.1 Critical Configuration****A.1.1 Name(s) of code system(s) used.**

1. Monte Carlo n-Particle, version 5.1.40 (MCNP5)
2. Monte Carlo n-Particle Extensions, version 2.5.0 (MCNPX)

**A.1.2 Bibliographic references for the codes used.**

1. X-5 Monte Carlo Team, "MCNP – a General Monte Carlo n-Particle Transport Code, version 5," LA-UR-03-1987, Los Alamos National Laboratory (2003).
2. J.S. Hendricks, et al., "MCNPX Extensions," LA-UR-05-2675, Los Alamos National Laboratory (April 2005).

**A.1.3 Origin of cross-section data.**

The Evaluated Neutron Data File library, ENDF/B-VI.8, was utilized in the benchmark model analysis. Other versions, including ENDF/B-V.2<sup>a</sup> and ENDF/B-VII.0<sup>b</sup>, were used with the benchmark model for a baseline comparison. The European Joint Evaluated Fission and Fusion File, JEFF-3.1<sup>c</sup> and the Japanese Evaluated Nuclear Data Library, JENDL-3.3,<sup>d</sup> were also included for a basic evaluative comparison. Such comparisons are typical.<sup>e,f</sup> Included in the reported results is a comparison of the ENDF/B-VI.8 analysis without accounting for the neutron cross-section adjustment for neutron upscatter at thermal energies. This correction is referred to as  $S(\alpha, \beta)$ . A JENDL-3.3 analysis was also performed with the inclusion of ENDF/B-VI.8 thermal neutron scattering data to demonstrate its importance in the prediction of  $k_{\text{eff}}$ . The JENDL-3.3 library did not include additional cross-section data for thermal scattering treatment.

**A.1.4 Spectral calculations and data reduction methods used.**

Not applicable

**A.1.5 Number of energy groups or if continuous-energy cross sections are used in the different phases of the calculation.**

Continuous-energy cross sections

---

<sup>a</sup> R. Kinsey, Ed., ENDF/B Summary Documentation, BNL-NCS-17542 (ENDF-201), 3<sup>rd</sup> ed., Brookhaven National Laboratory (1979).

<sup>b</sup> M. B. Chadwick, et al., "ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology," *Nucl. Data Sheets*, **107**: 2931-3060 (2006).

<sup>c</sup> A. Koning, R. Forrest, M. Kellett, R. Mills, H. Henriksson, and Y. Rugama, "The JEFF-3.1 Nuclear Data Library," JEFF Report 21, Organisation for Economic Co-operation and Development, Paris (2006).

<sup>d</sup> K. Shibata, et al., "Japanese Evaluated Nuclear Data Library Version 3 Revision-3: JENDL-3.3," *J. Nucl. Sci. Tech.*, **39**(11): 1125-1136 (November 2002).

<sup>e</sup> A. C. Kahler, "Monte Carlo Eigenvalue Calculations with ENDF/B-VI.8, JEFF-3.0, and JENDL-3.3 Cross Sections for a Selection of International Criticality Safety Benchmark Evaluation Project Handbook Benchmarks," *Nucl. Sci. Eng.*, **145**: 213-224 (2003).

<sup>f</sup> M. Goto, N. Nojiri, and S. Shimakawa, "Neutronics Calculations of HTTR with Several Nuclear Data Libraries," *J. Nucl. Sci. Tech.*, **43**(10): 1237-1244 (2006).

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE

**A.1.6 Component calculations.**

- Type of cell calculation – Reactor core and reflectors
- Geometry – Cylindrical
- Theory used – Not applicable
- Method used – Monte Carlo
- Calculation characteristics – histories/cycles/cycles skipped = 10,000/3,550/50 continuous-energy cross sections

**A.1.7 Other assumptions and characteristics.**

Not applicable

**A.1.8 Typical input listings.**

*MCNP5 and MCNPX Input Deck for the fully-loaded 30-fuel-column core of the HTTR (ordered TRISO lattice):*

```
HTTR Start-Up Core Critical (30 fuel columns) --
c
c John Darrell Bess - Idaho National Laboratory
c Last Updated: September 23, 2008
c
c Cell Cards *****
c --- Fuel Column -----
c ----- TRISO Particles -----
1  1  6.9523E-02  -1 imp:n=1 u=13 $ 3.4% kernel
2  13 5.5153E-02  1 -2 imp:n=1 u=13 $ buffer
3  14 9.2758E-02  2 -3 imp:n=1 u=13 $ IPyC
4  15 1.1453E-01  3 -4 imp:n=1 u=13 $ SiC
5  16 9.2758E-02  4 -5 imp:n=1 u=13 $ OPyC
6  17 8.5237E-02  5 -6 imp:n=1 u=13 $ overcoat
7  18 8.5237E-02  6 901 -902 903 -904 905 -906 imp:n=1 u=13 $ compact fill
11 like 1 but mat=2 u=14 rho=6.9524E-02 $ 3.9% kernel ---
12 like 2 but u=14 $ buffer
13 like 3 but u=14 $ IPyC
14 like 4 but u=14 $ SiC
15 like 5 but u=14 $ OPyC
16 like 6 but u=14 $ overcoat
17 like 7 but u=14 $ compact fill
21 like 1 but mat=3 u=15 rho=6.9526E-02 $ 4.3% kernel ---
22 like 2 but u=15 $ buffer
23 like 3 but u=15 $ IPyC
24 like 4 but u=15 $ SiC
25 like 5 but u=15 $ OPyC
26 like 6 but u=15 $ overcoat
27 like 7 but u=15 $ compact fill
31 like 1 but mat=4 u=16 rho=6.9527E-02 $ 4.8% kernel ---
32 like 2 but u=16 $ buffer
33 like 3 but u=16 $ IPyC
34 like 4 but u=16 $ SiC
35 like 5 but u=16 $ OPyC
36 like 6 but u=16 $ overcoat
37 like 7 but u=16 $ compact fill
41 like 1 but mat=5 u=17 rho=6.9528E-02 $ 5.2% kernel ---
42 like 2 but u=17 $ buffer
43 like 3 but u=17 $ IPyC
44 like 4 but u=17 $ SiC
45 like 5 but u=17 $ OPyC
46 like 6 but u=17 $ overcoat
47 like 7 but u=17 $ compact fill
51 like 1 but mat=6 u=18 rho=6.9530E-02 $ 5.9% kernel ---
52 like 2 but u=18 $ buffer
53 like 3 but u=18 $ IPyC
54 like 4 but u=18 $ SiC
55 like 5 but u=18 $ OPyC
56 like 6 but u=18 $ overcoat
57 like 7 but u=18 $ compact fill
61 like 1 but mat=7 u=19 rho=6.9532E-02 $ 6.3% kernel ---
```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

62 like 2 but u=19 $ buffer
63 like 3 but u=19 $ IPyC
64 like 4 but u=19 $ SiC
65 like 5 but u=19 $ OPyC
66 like 6 but u=19 $ overcoat
67 like 7 but u=19 $ compact fill
71 like 1 but mat=8 u=20 rho=6.9533E-02 $ 6.7% kernel ---
72 like 2 but u=20 $ buffer
73 like 3 but u=20 $ IPyC
74 like 4 but u=20 $ SiC
75 like 5 but u=20 $ OPyC
76 like 6 but u=20 $ overcoat
77 like 7 but u=20 $ compact fill
81 like 1 but mat=9 u=21 rho=6.9534E-02 $ 7.2% kernel ---
82 like 2 but u=21 $ buffer
83 like 3 but u=21 $ IPyC
84 like 4 but u=21 $ SiC
85 like 5 but u=21 $ OPyC
86 like 6 but u=21 $ overcoat
87 like 7 but u=21 $ compact fill
91 like 1 but mat=10 u=22 rho=6.9536E-02 $ 7.9% kernel ---
92 like 2 but u=22 $ buffer
93 like 3 but u=22 $ IPyC
94 like 4 but u=22 $ SiC
95 like 5 but u=22 $ OPyC
96 like 6 but u=22 $ overcoat
97 like 7 but u=22 $ compact fill
101 like 1 but mat=11 u=23 rho=6.9541E-02 $ 9.4% kernel ---
102 like 2 but u=23 $ buffer
103 like 3 but u=23 $ IPyC
104 like 4 but u=23 $ SiC
105 like 5 but u=23 $ OPyC
106 like 6 but u=23 $ overcoat
107 like 7 but u=23 $ compact fill
111 like 1 but mat=12 u=24 rho=6.9542E-02 $ 9.9% kernel ---
112 like 2 but u=24 $ buffer
113 like 3 but u=24 $ IPyC
114 like 4 but u=24 $ SiC
115 like 5 but u=24 $ OPyC
116 like 6 but u=24 $ overcoat
117 like 7 but u=24 $ compact fill
c ----- Compacts -----
120 18 8.5237E-02 901 -902 903 -904 905 -906 imp:n=1 u=300 $ compact fill
163 0 911 -912 913 -914 915 -916 imp:n=1 u=25 lat=1 fill=-13:13 -13:13 0:0
    300 26r 300 26r 300 9r 13 6r 300 9r 300 7r 13 10r 300 7r
    300 5r 13 14r 300 5r 300 4r 13 16r 300 4r 300 3r 13 18r 300 3r
    300 3r 13 18r 300 3r 300 2r 13 8r 300 2r 13 8r 300 2r
    300 1r 13 7r 300 6r 13 6r 300 2r 300 1r 13 6r 300 8r 13 6r 300 1r
    300 1r 13 6r 300 8r 13 6r 300 1r 300 1r 13 5r 300 10r 13 5r 300 1r
    300 1r 13 5r 300 10r 13 5r 300 1r 300 1r 13 5r 300 10r 13 5r 300 1r
    300 1r 13 6r 300 8r 13 6r 300 1r 300 1r 13 6r 300 8r 13 6r 300 1r
    300 2r 13 6r 300 6r 13 6r 300 2r 300 2r 13 8r 300 2r 13 8r 300 2r
    300 3r 13 18r 300 3r 300 3r 13 18r 300 3r 300 4r 13 16r 300 4r
    300 5r 13 14r 300 5r 300 7r 13 10r 300 7r 300 9r 13 6r 300 9r
    300 26r 300 26r
164 like 163 but u=26 fill=-13:13 -13:13 0:0
    300 26r 300 26r 300 9r 14 6r 300 9r 300 7r 14 10r 300 7r
    300 5r 14 14r 300 5r 300 4r 14 16r 300 4r 300 3r 14 18r 300 3r
    300 3r 14 18r 300 3r 300 2r 14 8r 300 2r 14 8r 300 2r
    300 1r 14 7r 300 6r 14 6r 300 2r 300 1r 14 6r 300 8r 14 6r 300 1r
    300 1r 14 6r 300 8r 14 6r 300 1r 300 1r 14 5r 300 10r 14 5r 300 1r
    300 1r 14 5r 300 10r 14 5r 300 1r 300 1r 14 5r 300 10r 14 5r 300 1r
    300 1r 14 6r 300 8r 14 6r 300 1r 300 1r 14 6r 300 8r 14 6r 300 1r
    300 2r 14 6r 300 6r 14 6r 300 2r 300 2r 14 8r 300 2r 14 8r 300 2r
    300 3r 14 18r 300 3r 300 3r 14 18r 300 3r 300 4r 14 16r 300 4r
    300 5r 14 14r 300 5r 300 7r 14 10r 300 7r 300 9r 14 6r 300 9r
    300 26r 300 26r
165 like 163 but u=27 fill=-13:13 -13:13 0:0
    300 26r 300 26r 300 9r 15 6r 300 9r 300 7r 15 10r 300 7r
    300 5r 15 14r 300 5r 300 4r 15 16r 300 4r 300 3r 15 18r 300 3r
    300 3r 15 18r 300 3r 300 2r 15 8r 300 2r 15 8r 300 2r
    300 1r 15 7r 300 6r 15 6r 300 2r 300 1r 15 6r 300 8r 15 6r 300 1r
    300 1r 15 6r 300 8r 15 6r 300 1r 300 1r 15 5r 300 10r 15 5r 300 1r
    300 1r 15 5r 300 10r 15 5r 300 1r 300 1r 15 5r 300 10r 15 5r 300 1r
    300 1r 15 6r 300 8r 15 6r 300 1r 300 1r 15 6r 300 8r 15 6r 300 1r
    300 2r 15 6r 300 6r 15 6r 300 2r 300 2r 15 8r 300 2r 15 8r 300 2r
    300 3r 15 18r 300 3r 300 3r 15 18r 300 3r 300 4r 15 16r 300 4r

```

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HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

300 5r 15 14r 300 5r 300 7r 15 10r 300 7r 300 9r 15 6r 300 9r
300 26r 300 26r
166 like 163 but u=28 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 16 6r 300 9r 300 7r 16 10r 300 7r
300 5r 16 14r 300 5r 300 4r 16 16r 300 4r 300 3r 16 18r 300 3r
300 3r 16 18r 300 3r 300 2r 16 8r 300 2r 16 8r 300 2r
300 1r 16 7r 300 6r 16 6r 300 2r 300 1r 16 6r 300 8r 16 6r 300 1r
300 1r 16 6r 300 8r 16 6r 300 1r 300 1r 16 5r 300 10r 16 5r 300 1r
300 1r 16 5r 300 10r 16 5r 300 1r 300 1r 16 5r 300 10r 16 5r 300 1r
300 1r 16 6r 300 8r 16 6r 300 1r 300 1r 16 6r 300 8r 16 6r 300 1r
300 2r 16 6r 300 6r 16 6r 300 2r 300 2r 16 8r 300 2r 16 8r 300 2r
300 3r 16 18r 300 3r 300 3r 16 18r 300 3r 300 4r 16 16r 300 4r
300 5r 16 14r 300 5r 300 7r 16 10r 300 7r 300 9r 16 6r 300 9r
300 26r 300 26r
167 like 163 but u=29 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 17 6r 300 9r 300 7r 17 10r 300 7r
300 5r 17 14r 300 5r 300 4r 17 16r 300 4r 300 3r 17 18r 300 3r
300 3r 17 18r 300 3r 300 2r 17 8r 300 2r 17 8r 300 2r
300 1r 17 7r 300 6r 17 6r 300 2r 300 1r 17 6r 300 8r 17 6r 300 1r
300 1r 17 6r 300 8r 17 6r 300 1r 300 1r 17 5r 300 10r 17 5r 300 1r
300 1r 17 5r 300 10r 17 5r 300 1r 300 1r 17 5r 300 10r 17 5r 300 1r
300 1r 17 6r 300 8r 17 6r 300 1r 300 1r 17 6r 300 8r 17 6r 300 1r
300 2r 17 6r 300 6r 17 6r 300 2r 300 2r 17 8r 300 2r 17 8r 300 2r
300 3r 17 18r 300 3r 300 3r 17 18r 300 3r 300 4r 17 16r 300 4r
300 5r 17 14r 300 5r 300 7r 17 10r 300 7r 300 9r 17 6r 300 9r
300 26r 300 26r
168 like 163 but u=30 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 18 6r 300 9r 300 7r 18 10r 300 7r
300 5r 18 14r 300 5r 300 4r 18 16r 300 4r 300 3r 18 18r 300 3r
300 3r 18 18r 300 3r 300 2r 18 8r 300 2r 18 8r 300 2r
300 1r 18 7r 300 6r 18 6r 300 2r 300 1r 18 6r 300 8r 18 6r 300 1r
300 1r 18 6r 300 8r 18 6r 300 1r 300 1r 18 5r 300 10r 18 5r 300 1r
300 1r 18 5r 300 10r 18 5r 300 1r 300 1r 18 5r 300 10r 18 5r 300 1r
300 1r 18 6r 300 8r 18 6r 300 1r 300 1r 18 6r 300 8r 18 6r 300 1r
300 2r 18 6r 300 6r 18 6r 300 2r 300 2r 18 8r 300 2r 18 8r 300 2r
300 3r 18 18r 300 3r 300 3r 18 18r 300 3r 300 4r 18 16r 300 4r
300 5r 18 14r 300 5r 300 7r 18 10r 300 7r 300 9r 18 6r 300 9r
300 26r 300 26r
169 like 163 but u=31 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 19 6r 300 9r 300 7r 19 10r 300 7r
300 5r 19 14r 300 5r 300 4r 19 16r 300 4r 300 3r 19 18r 300 3r
300 3r 19 18r 300 3r 300 2r 19 8r 300 2r 19 8r 300 2r
300 1r 19 7r 300 6r 19 6r 300 2r 300 1r 19 6r 300 8r 19 6r 300 1r
300 1r 19 6r 300 8r 19 6r 300 1r 300 1r 19 5r 300 10r 19 5r 300 1r
300 1r 19 5r 300 10r 19 5r 300 1r 300 1r 19 5r 300 10r 19 5r 300 1r
300 1r 19 6r 300 8r 19 6r 300 1r 300 1r 19 6r 300 8r 19 6r 300 1r
300 2r 19 6r 300 6r 19 6r 300 2r 300 2r 19 8r 300 2r 19 8r 300 2r
300 3r 19 18r 300 3r 300 3r 19 18r 300 3r 300 4r 19 16r 300 4r
300 5r 19 14r 300 5r 300 7r 19 10r 300 7r 300 9r 19 6r 300 9r
300 26r 300 26r
180 like 163 but u=32 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 20 6r 300 9r 300 7r 20 10r 300 7r
300 5r 20 14r 300 5r 300 4r 20 16r 300 4r 300 3r 20 18r 300 3r
300 3r 20 18r 300 3r 300 2r 20 8r 300 2r 20 8r 300 2r
300 1r 20 7r 300 6r 20 6r 300 2r 300 1r 20 6r 300 8r 20 6r 300 1r
300 1r 20 6r 300 8r 20 6r 300 1r 300 1r 20 5r 300 10r 20 5r 300 1r
300 1r 20 5r 300 10r 20 5r 300 1r 300 1r 20 5r 300 10r 20 5r 300 1r
300 1r 20 6r 300 8r 20 6r 300 1r 300 1r 20 6r 300 8r 20 6r 300 1r
300 2r 20 6r 300 6r 20 6r 300 2r 300 2r 20 8r 300 2r 20 8r 300 2r
300 3r 20 18r 300 3r 300 3r 20 18r 300 3r 300 4r 20 16r 300 4r
300 5r 20 14r 300 5r 300 7r 20 10r 300 7r 300 9r 20 6r 300 9r
300 26r 300 26r
181 like 163 but u=33 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 21 6r 300 9r 300 7r 21 10r 300 7r
300 5r 21 14r 300 5r 300 4r 21 16r 300 4r 300 3r 21 18r 300 3r
300 3r 21 18r 300 3r 300 2r 21 8r 300 2r 21 8r 300 2r
300 1r 21 7r 300 6r 21 6r 300 2r 300 1r 21 6r 300 8r 21 6r 300 1r
300 1r 21 6r 300 8r 21 6r 300 1r 300 1r 21 5r 300 10r 21 5r 300 1r
300 1r 21 5r 300 10r 21 5r 300 1r 300 1r 21 5r 300 10r 21 5r 300 1r
300 1r 21 6r 300 8r 21 6r 300 1r 300 1r 21 6r 300 8r 21 6r 300 1r
300 2r 21 6r 300 6r 21 6r 300 2r 300 2r 21 8r 300 2r 21 8r 300 2r
300 3r 21 18r 300 3r 300 3r 21 18r 300 3r 300 4r 21 16r 300 4r
300 5r 21 14r 300 5r 300 7r 21 10r 300 7r 300 9r 21 6r 300 9r
300 26r 300 26r
182 like 163 but u=34 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 22 6r 300 9r 300 7r 22 10r 300 7r
300 5r 22 14r 300 5r 300 4r 22 16r 300 4r 300 3r 22 18r 300 3r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

300 3r 22 18r 300 3r 300 2r 22 8r 300 2r 22 8r 300 2r
300 1r 22 7r 300 6r 22 6r 300 2r 300 1r 22 6r 300 8r 22 6r 300 1r
300 1r 22 6r 300 8r 22 6r 300 1r 300 1r 22 5r 300 10r 22 5r 300 1r
300 1r 22 5r 300 10r 22 5r 300 1r 300 1r 22 5r 300 10r 22 5r 300 1r
300 1r 22 6r 300 8r 22 6r 300 1r 300 1r 22 6r 300 8r 22 6r 300 1r
300 2r 22 6r 300 6r 22 6r 300 2r 300 2r 22 8r 300 2r 22 8r 300 2r
300 3r 22 18r 300 3r 300 3r 22 18r 300 3r 300 4r 22 16r 300 4r
300 5r 22 14r 300 5r 300 7r 22 10r 300 7r 300 9r 22 6r 300 9r
300 26r 300 26r
183 like 163 but u=35 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 23 6r 300 9r 300 7r 23 10r 300 7r
300 5r 23 14r 300 5r 300 4r 23 16r 300 4r 300 3r 23 18r 300 3r
300 3r 23 18r 300 3r 300 2r 23 8r 300 2r 23 8r 300 2r
300 1r 23 7r 300 6r 23 6r 300 2r 300 1r 23 6r 300 8r 23 6r 300 1r
300 1r 23 6r 300 8r 23 6r 300 1r 300 1r 23 5r 300 10r 23 5r 300 1r
300 1r 23 5r 300 10r 23 5r 300 1r 300 1r 23 5r 300 10r 23 5r 300 1r
300 1r 23 6r 300 8r 23 6r 300 1r 300 1r 23 6r 300 8r 23 6r 300 1r
300 2r 23 6r 300 6r 23 6r 300 2r 300 2r 23 8r 300 2r 23 8r 300 2r
300 3r 23 18r 300 3r 300 3r 23 18r 300 3r 300 4r 23 16r 300 4r
300 5r 23 14r 300 5r 300 7r 23 10r 300 7r 300 9r 23 6r 300 9r
300 26r 300 26r
184 like 163 but u=36 fill=-13:13 -13:13 0:0
300 26r 300 26r 300 9r 24 6r 300 9r 300 7r 24 10r 300 7r
300 5r 24 14r 300 5r 300 4r 24 16r 300 4r 300 3r 24 18r 300 3r
300 3r 24 18r 300 3r 300 2r 24 8r 300 2r 24 8r 300 2r
300 1r 24 7r 300 6r 24 6r 300 2r 300 1r 24 6r 300 8r 24 6r 300 1r
300 1r 24 6r 300 8r 24 6r 300 1r 300 1r 24 5r 300 10r 24 5r 300 1r
300 1r 24 5r 300 10r 24 5r 300 1r 300 1r 24 5r 300 10r 24 5r 300 1r
300 1r 24 6r 300 8r 24 6r 300 1r 300 1r 24 6r 300 8r 24 6r 300 1r
300 2r 24 6r 300 6r 24 6r 300 2r 300 2r 24 8r 300 2r 24 8r 300 2r
300 3r 24 18r 300 3r 300 3r 24 18r 300 3r 300 4r 24 16r 300 4r
300 5r 24 14r 300 5r 300 7r 24 10r 300 7r 300 9r 24 6r 300 9r
300 26r 300 26r
c
9163 0 921 -922 923 -924 925 -926 imp:n=1 u=925 lat=1 fill=25
9164 like 9163 but u=926 fill=26
9165 like 9163 but u=927 fill=27
9166 like 9163 but u=928 fill=28
9167 like 9163 but u=929 fill=29
9168 like 9163 but u=930 fill=30
9169 like 9163 but u=931 fill=31
9180 like 9163 but u=932 fill=32
9181 like 9163 but u=933 fill=33
9182 like 9163 but u=934 fill=34
9183 like 9163 but u=935 fill=35
9184 like 9163 but u=936 fill=36
185 0 12 -13 imp:n=1 u=37 fill=925
186 like 185 but u=38 fill=926
187 like 185 but u=39 fill=927
188 like 185 but u=40 fill=928
189 like 185 but u=41 fill=929
190 like 185 but u=42 fill=930
191 like 185 but u=43 fill=931
192 like 185 but u=44 fill=932
193 like 185 but u=45 fill=933
194 like 185 but u=46 fill=934
195 like 185 but u=47 fill=935
196 like 185 but u=48 fill=936
c ----- Fuel Pins -----
251 27 2.4616E-05 -12 imp:n=1 u=37 $ central hole
252 like 251 but u=38
253 like 251 but u=39
254 like 251 but u=40
255 like 251 but u=41
256 like 251 but u=42
257 like 251 but u=43
258 like 251 but u=44
259 like 251 but u=45
260 like 251 but u=46
261 like 251 but u=47
262 like 251 but u=48
263 27 2.4616E-05 13 -21 imp:n=1 u=37 $ annulus between compact and sleeve
264 like 263 but u=38
265 like 263 but u=39
266 like 263 but u=40
267 like 263 but u=41
268 like 263 but u=42

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

269 like 263 but u=43
270 like 263 but u=44
271 like 263 but u=45
272 like 263 but u=46
273 like 263 but u=47
274 like 263 but u=48
275 19 8.8747E-02 21 -22 imp:n=1 u=37 $ graphite sleeve
1275 like 275 but u=38
276 like 275 but u=39
277 like 275 but u=40
278 like 275 but u=41
279 like 275 but u=42
280 like 275 but u=43
281 like 275 but u=44
282 like 275 but u=45
283 like 275 but u=46
284 like 275 but u=47
285 like 275 but u=48
c c ----- Coolant Channels -----
286 27 2.4616E-05 22 -31 imp:n=1 u=37 $ annulus between sleeve and block
287 like 286 but u=38
288 like 286 but u=39
289 like 286 but u=40
290 like 286 but u=41
291 like 286 but u=42
292 like 286 but u=43
293 like 286 but u=44
294 like 286 but u=45
295 like 286 but u=46
296 like 286 but u=47
297 like 286 but u=48
298 25 8.7804E-02 31 imp:n=1 u=37 $ graphite block
299 like 298 but u=38
300 like 298 but u=39
301 like 298 but u=40
302 like 298 but u=41
303 like 298 but u=42
304 like 298 but u=43
305 like 298 but u=44
306 like 298 but u=45
307 like 298 but u=46
308 like 298 but u=47
309 like 298 but u=48
c ----- BP Pins -----
351 20 9.0451E-02 -41 imp:n=1 u=50 $ 2.0%
352 like 351 but mat=21 rho=9.0501E-02 u=51 $ 2.5%
353 22 8.8747E-02 -42 imp:n=1 u=50 $ graphite disks
354 like 353 but u=51
355 20 9.0451E-02 -43 imp:n=1 u=50 $ 2.0%
356 like 355 but mat=21 rho=9.0501E-02 u=51 $ 2.5%
357 27 2.4616E-05 41 42 43 -44 imp:n=1 u=50 $ pin gap
358 like 357 but u=51
359 27 2.4616E-05 -44 imp:n=1 u=52 $ empty pin position
360 25 8.7804E-02 44 imp:n=1 u=50 $ graphite block
361 like 360 but u=51
362 like 360 but u=52
c ----- Blocks -----
401 25 8.7804E-02 -501 imp:n=1 u=61 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 4/5
61 10r
61 10r
61 4r      50 37 37 37      61 1r
61 3r      37 37 37 37 37    61 1r
61 2r      37 37 37 37 37 37 61 1r
61 1r 37 37 37 37 61 37 37 52 61 1r
61 1r      37 37 37 37 37 37 61 2r
61 1r      37 37 37 37 37    61 3r
61 1r      50 37 37 37      61 4r
61 10r
61 10r
402 25 8.7804E-02 -501 imp:n=1 u=62 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 3
62 10r
62 10r
62 4r      51 39 39 39      62 1r
62 3r      39 39 39 39 39    62 1r
62 2r      39 39 39 39 39 39 62 1r
62 1r 39 39 39 62 39 39 52 62 1r
62 1r      39 39 39 39 39 39 62 2r

```



## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

62 1r      39 39 39 39 39      62 3r
62 1r      51 39 39 39      62 4r
62 10r
62 10r
403 25 8.7804E-02 -501 imp:n=1 u=63 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 2
63 10r
63 10r
63 4r      51 41 41 41      63 1r
63 3r      41 41 41 41 41      63 1r
63 2r 41 41 41 41 41 41      63 1r
63 1r 41 41 41 63 41 41 52 63 1r
63 1r 41 41 41 41 41 41      63 2r
63 1r      41 41 41 41 41      63 3r
63 1r      51 41 41 41      63 4r
63 10r
63 10r
404 25 8.7804E-02 -501 imp:n=1 u=64 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 1
64 10r
64 10r
64 4r      50 44 44 44      64 1r
64 3r      44 44 44 44 44      64 1r
64 2r 44 44 44 44 44 44      64 1r
64 1r 44 44 44 64 44 44 52 64 1r
64 1r 44 44 44 44 44 44      64 2r
64 1r      44 44 44 44 44      64 3r
64 1r      50 44 44 44      64 4r
64 10r
64 10r
405 25 8.7804E-02 -501 imp:n=1 u=65 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 4/5
65 10r
65 10r
65 4r      50 38 38 38      65 1r
65 3r      38 38 38 38 38      65 1r
65 2r 38 38 38 38 38 38      65 1r
65 1r 38 38 38 65 38 38 52 65 1r
65 1r 38 38 38 38 38 38      65 2r
65 1r      38 38 38 38 38      65 3r
65 1r      50 38 38 38      65 4r
65 10r
65 10r
406 25 8.7804E-02 -501 imp:n=1 u=66 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 3
66 10r
66 10r
66 4r      51 41 41 41      66 1r
66 3r      41 41 41 41 41      66 1r
66 2r 41 41 41 41 41 41      66 1r
66 1r 41 41 41 66 41 41 52 66 1r
66 1r 41 41 41 41 41 41      66 2r
66 1r      41 41 41 41 41      66 3r
66 1r      51 41 41 41      66 4r
66 10r
66 10r
407 25 8.7804E-02 -501 imp:n=1 u=67 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 2
67 10r
67 10r
67 4r      51 43 43 43      67 1r
67 3r      43 43 43 43 43      67 1r
67 2r 43 43 43 43 43 43      67 1r
67 1r 43 43 43 67 43 43 52 67 1r
67 1r 43 43 43 43 43 43      67 2r
67 1r      43 43 43 43 43      67 3r
67 1r      51 43 43 43      67 4r
67 10r
67 10r
408 25 8.7804E-02 -501 imp:n=1 u=68 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 1
68 10r
68 10r
68 4r      50 46 46 46      68 1r
68 3r      46 46 46 46 46      68 1r
68 2r 46 46 46 46 46 46      68 1r
68 1r 46 46 46 68 46 46 52 68 1r
68 1r 46 46 46 46 46 46      68 2r
68 1r      46 46 46 46 46      68 3r
68 1r      50 46 46 46      68 4r
68 10r
68 10r
409 25 8.7804E-02 -501 imp:n=1 u=69 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 4/5

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

69 10r
69 10r
69 4r      50 39 39 69      69 1r
69 3r      39 39 39 39 39  69 1r
69 2r      39 39 39 39 39  69 1r
69 1r 39 39 39 69 39 39 52 69 1r
69 1r 39 39 39 39 39 39  69 2r
69 1r      39 39 39 39 39  69 3r
69 1r      50 39 39 69      69 4r
69 10r
69 10r
410 25 8.7804E-02 -501 imp:n=1 u=70 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 3
70 10r
70 10r
70 4r      51 42 42 70      70 1r
70 3r      42 42 42 42 42  70 1r
70 2r      42 42 42 42 42  70 1r
70 1r 42 42 42 70 42 42 52 70 1r
70 1r      42 42 42 42 42  70 2r
70 1r      42 42 42 42 42  70 3r
70 1r      51 42 42 70      70 4r
70 10r
70 10r
411 25 8.7804E-02 -501 imp:n=1 u=71 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 2
71 10r
71 10r
71 4r      51 45 45 71      71 1r
71 3r      45 45 45 45 45  71 1r
71 2r      45 45 45 45 45  71 1r
71 1r 45 45 45 71 45 45 52 71 1r
71 1r      45 45 45 45 45  71 2r
71 1r      45 45 45 45 45  71 3r
71 1r      51 45 45 71      71 4r
71 10r
71 10r
412 25 8.7804E-02 -501 imp:n=1 u=72 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 1
72 10r
72 10r
72 4r      50 47 47 72      72 1r
72 3r      47 47 47 47 47  72 1r
72 2r      47 47 47 47 47  72 1r
72 1r 47 47 47 72 47 47 52 72 1r
72 1r      47 47 47 47 47  72 2r
72 1r      47 47 47 47 47  72 3r
72 1r      50 47 47 72      72 4r
72 10r
72 10r
413 25 8.7804E-02 -501 imp:n=1 u=73 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 4/5
73 10r
73 10r
73 4r      50 40 40 73      73 1r
73 3r      40 40 40 40 40  73 1r
73 2r      40 40 40 40 40  73 1r
73 1r 40 40 40 73 40 40 52 73 1r
73 1r      40 40 40 40 40  73 2r
73 1r      40 40 40 40 40  73 3r
73 1r      50 40 40 73      73 4r
73 10r
73 10r
414 25 8.7804E-02 -501 imp:n=1 u=74 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 3
74 10r
74 10r
74 4r      51 43 43 74      74 1r
74 3r      43 43 43 43 43  74 1r
74 2r      43 43 43 43 43  74 1r
74 1r 43 43 43 74 43 43 52 74 1r
74 1r      43 43 43 43 43  74 2r
74 1r      43 43 43 43 43  74 3r
74 1r      51 43 43 74      74 4r
74 10r
74 10r
415 25 8.7804E-02 -501 imp:n=1 u=75 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 2
75 10r
75 10r
75 4r      51 46 46 75      75 1r
75 3r      46 46 46 46 46  75 1r
75 2r      46 46 46 46 46  75 1r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

75 1r 46 46 46 75 46 46 52 75 1r
75 1r 46 46 46 46 46 46 75 2r
75 1r 46 46 46 46 46 75 3r
75 1r 51 46 46 75 75 4r
75 10r
75 10r
416 25 8.7804E-02 -501 imp:n=1 u=76 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 1
76 10r
76 10r
76 4r 50 48 48 76 76 1r
76 3r 48 48 48 48 48 76 1r
76 2r 48 48 48 48 48 48 76 1r
76 1r 48 48 48 76 48 48 52 76 1r
76 1r 48 48 48 48 48 48 76 2r
76 1r 48 48 48 48 48 76 3r
76 1r 50 48 48 76 76 4r
76 10r
76 10r
451 0 -502 imp:n=1 u=81 fill=61
452 like 451 but u=82 fill=62
453 like 451 but u=83 fill=63
454 like 451 but u=84 fill=64
455 like 451 but u=85 fill=65
456 like 451 but u=86 fill=66
457 like 451 but u=87 fill=67
458 like 451 but u=88 fill=68
459 like 451 but u=89 fill=69
460 like 451 but u=90 fill=70
461 like 451 but u=91 fill=71
462 like 451 but u=92 fill=72
463 like 451 but u=93 fill=73
464 like 451 but u=94 fill=74
465 like 451 but u=95 fill=75
466 like 451 but u=96 fill=76
467 27 2.4616E-05 502 imp:n=1 u=81
468 like 467 but u=82
469 like 467 but u=83
470 like 467 but u=84
471 like 467 but u=85
472 like 467 but u=86
473 like 467 but u=87
474 like 467 but u=88
475 like 467 but u=89
476 like 467 but u=90
477 like 467 but u=91
478 like 467 but u=92
479 like 467 but u=93
480 like 467 but u=94
481 like 467 but u=95
482 like 467 but u=96
c ----- Reflectors -----
483 27 2.4616E-05 -201 imp:n=1 u=97 $ coolant channels
484 25 8.7804E-02 201 imp:n=1 u=97 $ graphite block
485 25 8.7804E-02 -501 imp:n=1 u=98 lat=2 fill= -5:5 -5:5 0:0 $ 33-hole
98 10r
98 10r
98 4r 98 97 97 97 98 1r
98 3r 97 97 97 97 97 98 1r
98 2r 97 97 97 97 97 97 98 1r
98 1r 97 97 97 98 97 97 98 98 1r
98 1r 97 97 97 97 97 97 98 2r
98 1r 97 97 97 97 97 98 3r
98 1r 98 97 97 97 98 4r
98 10r
98 10r
486 25 8.7804E-02 -501 imp:n=1 u=99 lat=2 fill= -5:5 -5:5 0:0 $ 31-hole
99 10r
99 10r
99 4r 99 97 97 99 99 1r
99 3r 97 97 97 97 97 99 1r
99 2r 97 97 97 97 97 97 99 1r
99 1r 97 97 97 99 97 97 99 99 1r
99 1r 97 97 97 97 97 97 99 2r
99 1r 97 97 97 97 97 99 3r
99 1r 99 97 97 99 99 4r
99 10r
99 10r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

489 0 -502 imp:n=1 u=100 fill=98
490 0 -502 imp:n=1 u=101 fill=99
491 like 467 but u=100
492 like 467 but u=101
c ----- Columns -----
501 27 2.4616E-05 -503 imp:n=1 u=121 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 1
    121 121 121 121 121 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 81 121 121 121 121
    121 121 121 121 81 121 121 121 121
    121 121 121 121 82 121 121 121 121
    121 121 121 121 83 121 121 121 121
    121 121 121 121 84 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 121 121 121 121 121
502 27 2.4616E-05 -503 imp:n=1 u=122 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 2
    122 122 122 122 122 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 85 122 122 122 122
    122 122 122 122 85 122 122 122 122
    122 122 122 122 86 122 122 122 122
    122 122 122 122 87 122 122 122 122
    122 122 122 122 88 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 122 122 122 122 122
503 27 2.4616E-05 -503 imp:n=1 u=123 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 3
    123 123 123 123 123 123 123 123 123
    123 123 123 123 100 123 123 123 123
    123 123 123 123 100 123 123 123 123
    123 123 123 123 89 123 123 123 123
    123 123 123 123 89 123 123 123 123
    123 123 123 123 90 123 123 123 123
    123 123 123 123 91 123 123 123 123
    123 123 123 123 92 123 123 123 123
    123 123 123 123 101 123 123 123 123
    123 123 123 123 101 123 123 123 123
    123 123 123 123 123 123 123 123 123
504 27 2.4616E-05 -503 imp:n=1 u=124 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 4
    124 124 124 124 124 124 124 124 124
    124 124 124 124 100 124 124 124 124
    124 124 124 124 100 124 124 124 124
    124 124 124 124 93 124 124 124 124
    124 124 124 124 93 124 124 124 124
    124 124 124 124 94 124 124 124 124
    124 124 124 124 95 124 124 124 124
    124 124 124 124 96 124 124 124 124
    124 124 124 124 101 124 124 124 124
    124 124 124 124 101 124 124 124 124
    124 124 124 124 124 124 124 124 124
525 0 -950 imp:n=1 u=125 fill=121 $ B04
526 like 525 but u=126 fill=121 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ B01
527 like 525 but u=127 fill=121 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ B02
528 like 525 but u=128 fill=121 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ B03
529 like 525 but u=129 fill=121 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ B05
530 like 525 but u=130 fill=121 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ B06
531 0 -950 imp:n=1 u=131 fill=122 $ C02
532 like 531 but u=132 fill=122 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ C04
533 like 531 but u=133 fill=122 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ C06
534 like 531 but u=134 fill=122 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ C08
535 like 531 but u=135 fill=122 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ C10
536 like 531 but u=136 fill=122 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ C12
537 0 -950 imp:n=1 u=137 fill=123 $ D05/09
538 like 537 but u=138 fill=123 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ D02/06
539 like 537 but u=139 fill=123 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ D03/17
540 like 537 but u=140 fill=123 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ D08/12
541 like 537 but u=141 fill=123 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ D11/15
542 like 537 but u=142 fill=123 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ D14/18
543 0 -950 imp:n=1 u=143 fill=124 $ D07
544 like 543 but u=144 fill=124 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ D01
545 like 543 but u=145 fill=124 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ D04
546 like 543 but u=146 fill=124 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ D10
547 like 543 but u=147 fill=124 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ D13
548 like 543 but u=148 fill=124 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ D16

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

c
c --- Control Column -----
c ----- Control Rod Segments -----
601 24 8.7530E-02 -103 imp:n=1 u=150 $ spine
602 27 2.4616E-05 103 -104 imp:n=1 u=150 $ helium gap
603 24 8.7530E-02 104 -101 imp:n=1 u=150 $ inner clad
604 23 9.1660E-02 101 -102 imp:n=1 u=150 $ absorber
605 24 8.7530E-02 102 -105 imp:n=1 u=150 $ outer clad
606 27 2.4616E-05 105 imp:n=1 u=150 $ helium
607 0 -151 imp:n=1 u=151 lat=2 fill=150 $ rod segment
608 0 -152 imp:n=1 u=152 fill=151 $ control rod
609 27 2.4616E-05 152 imp:n=1 u=152 $ helium
c ----- Positions -----
610 0 -999 imp:n=1 u=153 trcl=(0 0 177.5) fill=152 $ C
611 0 -153 imp:n=1 u=154 trcl=(-5.4 -9.35307 0) fill=153
612 like 611 but trcl=(10.8 0 0)
613 27 2.4616E-05 -153 imp:n=1 u=154 trcl=(-5.4 9.35307 0) $ RSS
614 like 610 but u=155 trcl=(0 0 177.5) $ R1
615 like 611 but u=156 fill=155
616 like 612 but u=156 fill=155
617 like 613 but u=156
618 like 610 but u=157 trcl=(0 0 177.5) $ R2
619 like 611 but u=158 fill=157
620 like 612 but u=158 fill=157
621 like 613 but u=158
622 like 610 but u=159 trcl=(0 0 404.9) $ R3
623 like 611 but u=160 fill=159
624 like 612 but u=160 fill=159
625 like 613 but u=160
c ----- C Column -----
626 25 8.7804E-02 #611 #612 #613 -550 imp:n=1 u=154 $ graphite blocks
627 27 2.4616E-05 550 imp:n=1 u=154
628 0 -950 imp:n=1 u=161 fill=154 $ A01
c
c ----- R1 Columns -----
629 25 8.7804E-02 #615 #616 #617 -550 imp:n=1 u=156 $ graphite blocks
630 like 627 but u=156
631 like 628 but u=162 fill=156 $ C01
632 like 631 but u=163 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ C03
633 like 631 but u=164 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ C05
634 like 631 but u=165 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ C07
635 like 631 but u=166 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ C09
636 like 631 but u=167 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ C11
c ----- R2 Columns -----
637 25 8.7804E-02 #619 #620 #621 -550 imp:n=1 u=158 $ graphite blocks
638 like 627 but u=158
639 like 628 but u=168 fill=158 $ E23
640 like 639 but u=169 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ E03
641 like 639 but u=170 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ E07
642 like 639 but u=171 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ E11
643 like 639 but u=172 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ E15
644 like 639 but u=173 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ E19
c ----- R3 Columns -----
645 25 8.7804E-02 #623 #624 #625 -550 imp:n=1 u=160 $ graphite blocks
646 like 627 but u=160
647 like 628 but u=174 fill=160 $ E01
648 like 647 but u=175 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ E09
649 like 647 but u=176 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ E17
c
c --- Instrumentation Column -----
c ----- Positions -----
661 like 613 but u=181 trcl=(5.4 9.35307 0)
662 like 613 but u=181 trcl=(5.4 -9.35307 0)
663 like 613 but u=181 trcl=(-10.8 0 0)
c ----- Columns -----
664 25 8.7804E-02 #661 #662 #663 -550 imp:n=1 u=181 $ graphite blocks
665 like 627 but u=181
666 0 -950 imp:n=1 u=182 fill=181
c
c --- Reflector Column -----
c ----- Columns -----
671 25 8.7804E-02 -550 imp:n=1 u=183 $ graphite blocks
672 like 627 but u=183
673 0 -950 imp:n=1 u=184 fill=183
c
c --- HTTR Core -----
c ----- Core Map -----

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

701 26 8.6134E-02 -551 imp:n=1 lat=2 u=200 fill=-6:6 -6:6 0:0
200 12r
200 12r
200 5r      175 184 170 184 182      200 1r
200 4r      184 143 138 137 145 184      200 1r
200 3r      171 140 164 132 163 139 169      200 1r
200 2r      184 137 133 128 127 131 138 184      200 1r
200 1r      182 146 165 125 161 126 162 144 174      200 1r
200 1r      184 141 134 129 130 136 142 184      200 2r
200 1r      172 140 166 135 167 139 168      200 3r
200 1r      184 147 142 141 148 184      200 4r
200 1r      176 184 173 184 182      200 5r
200 12r
200 12r

```

```

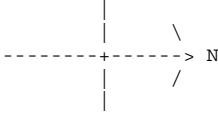
c
c ----- Core Map Legend ---

```

```

c   u 12r
c   u 12r
c   u 5r      Z G Y G I      u 1r
c   u 4r      G 4 3 3 4 G      u 1r
c   u 3r      Y 3 X 2 X 3 Y      u 1r
c   u 2r      G 3 2 1 1 2 3 G      u 1r
c   u 1r      I 4 X 1 C 1 X 4 Z      u 1r
c   u 1r      G 3 2 1 1 2 3 G      u 2r
c   u 1r      Y 3 X 2 X 3 Y      u 3r
c   u 1r      G 4 3 3 4 G      u 4r
c   u 1r      Z G Y G I      u 5r
c   u 12r
c   u 12r

```



```

c
c   1 = Fuel Columns #1
c   2 = Fuel Columns #2
c   3 = Fuel Columns #3
c   4 = Fuel Columns #4
c   C = Central Control Column
c   X = R1 Control Columns
c   Y = R2 Control Columns
c   Z = R3 Control Columns
c   I = Instrumentation Columns
c   G = Removable Reflector Columns

```

```

702 0 -602 imp:n=1 fill=200

```

```

c --- Permanent Reflector -----

```

```

711 26 8.6134E-02 602 -651 imp:n=1

```

```

c --- The Great Void -----

```

```

999 0 651 imp:n=0

```

```

c
c Surface Cards *****
c --- Fuel Blocks -----

```

```

c ----- TRISO Particles -----

```

```

1   so 0.03   $ UO2 kernal
2   so 0.036  $ buffer
3   so 0.039  $ IPyC
4   so 0.0415 $ SiC
5   so 0.046  $ OPyC
6   so 0.066  $ overcoat

```

```

901 px -0.125
902 px  0.125
903 py -0.125
904 py  0.125
905 pz -0.125
906 pz  0.125

```

```

c ----- Compacts -----

```

```

911 px -0.053
912 px  0.053
913 py -0.053
914 py  0.053
915 pz -0.05
916 pz  0.05
921 px -1.31
922 px  1.31
923 py -1.31
924 py  1.31
925 pz -0.05

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

926 pz 0.05
12  rcc 0 0 -27.3 0 0 54.6 0.5 $ inside
13  rcc 0 0 -27.3 0 0 54.6 1.3 $ outside
c ----- Fuel Pins -----
21  rcc 0 0 -27.45 0 0 54.9 1.325 $ inside
22  rcc 0 0 -28.85 0 0 57.7 1.7 $ outside
c ----- Coolant Channels -----
31  rcc 0 0 -31 0 0 62 2.05
c ----- BP Pins -----
41  rcc 0 0 -25 0 0 20 0.7 $ BP
42  rcc 0 0 -5 0 0 10 0.7 $ graphite
43  rcc 0 0 5 0 0 20 0.7 $ BP
44  rcc 0 0 -25 0 0 50 0.75 $ pin
c
c --- Control Blocks -----
c ----- Control Rod Segments -----
101 rcc 0 0 11 0 0 29 3.75 $ inside
102 rcc 0 0 11 0 0 29 5.25 $ outside
103 rcc 0 0 11 0 0 29 0.5 $ spine
104 rcc 0 0 11 0 0 29 3.25 $ inside clad
105 rcc 0 0 10 0 0 31 5.65 $ outside clad
c ----- Positions -----
151 hex 0 0 10 0 0 31 10 $ "box"
152 rcc 0 0 -145 0 0 310 6.30 $ control rod
153 rcc 0 0 -155 0 0 416 6.15 $ control rod hole
c
c --- Reflector Blocks -----
c ----- Coolant Channels -----
201 rcc 0 0 -31 0 0 62 1.15
c
c --- Blocks -----
501 hex 0 0 -30 0 0 60 2.575 0 0 $ pitch
502 hex 0 0 -29.5 0 0 59 0 18 0 $ graphite
503 hex 0 0 -29 0 0 58 0 18.2 0 $ helium
c
c --- Columns -----
550 hex 0 0 -261.5 0 0 523 0 18 0
551 hex 0 0 -261 0 0 522 0 18.1 0
c
c --- HTTR Core -----
602 hex 0 0 -261 0 0 522 -148 0 0
c
c --- Permanent Reflector -----
651 rcc 0 0 -261 0 0 522 212.5
c
950 rcc 0 0 -1000 0 0 2000 25 $ small cylinder
999 rcc 0 0 -2500 0 0 5000 2500 $ big cylinder
c

c Data Cards *****
c --- Material Cards -----
c ----- Kernel (3.4%) -----
m1  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 6.0946E-06
    92235.66c 7.9784E-04
    92238.66c 2.2376E-02
c    Total      6.9523E-02
c
c ----- Kernel (3.9%) -----
m2  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 6.9909E-06
    92235.66c 9.1517E-04
    92238.66c 2.2259E-02
c    Total      6.9524E-02
c
c ----- Kernel (4.3%) -----
m3  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 7.7079E-06
    92235.66c 1.0090E-03
    92238.66c 2.2165E-02
c    Total      6.9526E-02

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

c
c ----- Kernel (4.8%) -----
m4  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 8.6041E-06
      92235.66c 1.1264E-03
      92238.66c 2.2049E-02
c    Total      6.9527E-02
c
c ----- Kernel (5.2%) -----
m5  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 9.3211E-06
      92235.66c 1.2202E-03
      92238.66c 2.1955E-02
c    Total      6.9528E-02
c
c ----- Kernel (5.9%) -----
m6  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.0576E-05
      92235.66c 1.3845E-03
      92238.66c 2.1792E-02
c    Total      6.9530E-02
c
c ----- Kernel (6.3%) -----
m7  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.1293E-05
      92235.66c 1.4783E-03
      92238.66c 2.1699E-02
c    Total      6.9532E-02
c
c ----- Kernel (6.7%) -----
m8  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.2010E-05
      92235.66c 1.5722E-03
      92238.66c 2.1605E-02
c    Total      6.9533E-02
c
c ----- Kernel (7.2%) -----
m9  5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.2906E-05
      92235.66c 1.6895E-03
      92238.66c 2.1489E-02
c    Total      6.9534E-02
c
c ----- Kernel (7.9%) -----
m10 5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.4161E-05
      92235.66c 1.8538E-03
      92238.66c 2.1325E-02
c    Total      6.9536E-02
c
c ----- Kernel (9.4%) -----
m11 5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05
      92234.66c 1.6850E-05
      92235.66c 2.2058E-03
      92238.66c 2.0975E-02
c    Total      6.9541E-02
c
c ----- Kernel (9.9%) -----
m12 5010.66c 1.7276E-07
      8016.62c 4.6325E-02
      8017.66c 1.7610E-05

```



## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

92234.66c 1.7746E-05
92235.66c 2.3231E-03
92238.66c 2.0858E-02
c   Total      6.9542E-02
c
c ----- Buffer Layer -----
m13  5010.66c 1.8290E-08
      6000.66c 5.5153E-02
c   Total      5.5153E-02
mt13  grph.60t
c
c ----- IPyC Layer -----
m14  5010.66c 3.0761E-08
      6000.66c 9.2758E-02
c   Total      9.2758E-02
mt14  grph.60t
c
c ----- SiC Layer -----
m15  5010.66c 5.3208E-08
      6000.66c 8.0223E-02
      14028.62c 3.1641E-02
      14029.62c 1.6067E-03
      14030.62c 1.0591E-03
c   Total      1.1453E-01
c
c ----- OPyC Layer -----
m16  5010.66c 3.0761E-08
      6000.66c 9.2758E-02
c   Total      9.2758E-02
mt16  grph.60t
c
c ----- Graphite Overcoat -----
m17  5010.66c 2.8267E-08
      6000.66c 8.5237E-02
c   Total      8.5237E-02
mt17  grph.60t
c
c ----- Graphite Compact -----
m18  5010.66c 1.5452E-08
      6000.66c 8.5237E-02
c   Total      8.5237E-02
mt18  grph.60t
c
c ----- Graphite Sleeve -----
m19  5010.66c 7.2596E-09
      6000.66c 8.8747E-02
c   Total      8.8747E-02
mt19  grph.60t
c
c ----- Burnable Poison (2.0%) -----
m20  5010.66c 3.9906E-04
      5011.66c 1.6063E-03
      6000.66c 8.8446E-02
c   Total      9.0451E-02
c
c ----- Burnable Poison (2.5%) -----
m21  5010.66c 4.9882E-04
      5011.66c 2.0078E-03
      6000.66c 8.7995E-02
c   Total      9.0501E-02
c
c ----- Graphite Disks -----
m22  5010.66c 7.2596E-09
      6000.66c 8.8747E-02
c   Total      8.8747E-02
mt22  grph.60t
c
c ----- Neutron Absorber -----
m23  5010.66c 6.8220E-03
      5011.66c 2.4974E-02
      6000.66c 6.6685E-02
c   Total      9.1660E-02
c
c ----- Alloy 800H -----
m24  6000.66c 3.2210E-04
      13027.62c 6.7209E-04
      14028.62c 5.5580E-04

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
 CRIT-SUB-REAC-COEF-KIN-RRATE

```

14029.62c 2.8222E-05
14030.62c 1.8604E-05
15031.66c 3.1225E-05
16000.62c 1.5081E-05
22000.62c 3.7884E-04
24050.62c 8.4860E-04
24052.62c 1.6364E-02
24053.62c 1.8556E-03
24054.62c 4.6189E-04
25055.62c 8.8022E-04
26054.62c 2.2265E-03
26056.62c 3.4951E-02
26057.62c 8.0717E-04
26058.62c 1.0742E-04
28058.62c 1.8229E-02
28060.62c 7.0217E-03
28061.62c 3.0523E-04
28062.62c 9.7320E-04
28064.62c 2.4785E-04
29063.62c 1.5791E-04
29065.62c 7.0383E-05
c      Total      8.7530E-02
c
c ----- IG-110 Graphite -----
m25    5010.66c 7.7648E-09
        6000.66c 8.7804E-02
c      Total      8.7804E-02
mt25    grph.60t
c
c ----- PGX Graphite -----
m26    5010.66c 3.6372E-08
        6000.66c 8.6134E-02
c      Total      8.6134E-02
mt26    grph.60t
c
c ----- Helium Coolant -----
m27    2003.66c 3.3724E-11
        2004.66c 2.4616E-05
c      Total      2.4616E-05
c
c --- Control Cards -----
mode  n
kcode 10000 1 50 3550
ksrc   0 0 0
c print

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

*MCNP5 and MCNPX Input Deck for the fully-loaded 30-fuel-column core of the HTTR (uniformly-filled TRISO lattice):*

```

HTTR Start-Up Core Critical (30 fuel columns) --
c
c John Darrell Bess - Idaho National Laboratory
c Last Updated: August 8, 2008
c
c Cell Cards *****
c --- Fuel Column -----
c ----- TRISO Particles -----
1   1  6.9523E-02  -1 imp:n=1 u=1 $ 3.4% kernel
2   13 5.5153E-02  1 -2 imp:n=1 u=1 $ buffer
3   14 9.2758E-02  2 -3 imp:n=1 u=1 $ IPyC
4   15 1.1453E-01  3 -4 imp:n=1 u=1 $ SiC
5   16 9.2758E-02  4 -5 imp:n=1 u=1 $ OPyC
6   17 8.5237E-02  5 -6 imp:n=1 u=1 $ overcoat
7   18 8.5237E-02  6  imp:n=1 u=1 $ compact fill
11  like 1 but mat=2 u=2 rho=6.9524E-02 $ 3.9% kernel ---
12  like 2 but u=2 $ buffer
13  like 3 but u=2 $ IPyC
14  like 4 but u=2 $ SiC
15  like 5 but u=2 $ OPyC
16  like 6 but u=2 $ overcoat
17  like 7 but u=2 $ compact fill
21  like 1 but mat=3 u=3 rho=6.9526E-02 $ 4.3% kernel ---
22  like 2 but u=3 $ buffer
23  like 3 but u=3 $ IPyC
24  like 4 but u=3 $ SiC
25  like 5 but u=3 $ OPyC
26  like 6 but u=3 $ overcoat
27  like 7 but u=3 $ compact fill
31  like 1 but mat=4 u=4 rho=6.9527E-02 $ 4.8% kernel ---
32  like 2 but u=4 $ buffer
33  like 3 but u=4 $ IPyC
34  like 4 but u=4 $ SiC
35  like 5 but u=4 $ OPyC
36  like 6 but u=4 $ overcoat
37  like 7 but u=4 $ compact fill
41  like 1 but mat=5 u=5 rho=6.9528E-02 $ 5.2% kernel ---
42  like 2 but u=5 $ buffer
43  like 3 but u=5 $ IPyC
44  like 4 but u=5 $ SiC
45  like 5 but u=5 $ OPyC
46  like 6 but u=5 $ overcoat
47  like 7 but u=5 $ compact fill
51  like 1 but mat=6 u=6 rho=6.9530E-02 $ 5.9% kernel ---
52  like 2 but u=6 $ buffer
53  like 3 but u=6 $ IPyC
54  like 4 but u=6 $ SiC
55  like 5 but u=6 $ OPyC
56  like 6 but u=6 $ overcoat
57  like 7 but u=6 $ compact fill
61  like 1 but mat=7 u=7 rho=6.9532E-02 $ 6.3% kernel ---
62  like 2 but u=7 $ buffer
63  like 3 but u=7 $ IPyC
64  like 4 but u=7 $ SiC
65  like 5 but u=7 $ OPyC
66  like 6 but u=7 $ overcoat
67  like 7 but u=7 $ compact fill
71  like 1 but mat=8 u=8 rho=6.9533E-02 $ 6.7% kernel ---
72  like 2 but u=8 $ buffer
73  like 3 but u=8 $ IPyC
74  like 4 but u=8 $ SiC
75  like 5 but u=8 $ OPyC
76  like 6 but u=8 $ overcoat
77  like 7 but u=8 $ compact fill
81  like 1 but mat=9 u=9 rho=6.9534E-02 $ 7.2% kernel ---
82  like 2 but u=9 $ buffer
83  like 3 but u=9 $ IPyC
84  like 4 but u=9 $ SiC
85  like 5 but u=9 $ OPyC
86  like 6 but u=9 $ overcoat
87  like 7 but u=9 $ compact fill
91  like 1 but mat=10 u=10 rho=6.9536E-02 $ 7.9% kernel ---
92  like 2 but u=10 $ buffer

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

93  like 3 but u=10 $ IPyC
94  like 4 but u=10 $ SiC
95  like 5 but u=10 $ OPyC
96  like 6 but u=10 $ overcoat
97  like 7 but u=10 $ compact fill
101 like 1 but mat=11 u=11 rho=6.9541E-02 $ 9.4% kernel ---
102 like 2 but u=11 $ buffer
103 like 3 but u=11 $ IPyC
104 like 4 but u=11 $ SiC
105 like 5 but u=11 $ OPyC
106 like 6 but u=11 $ overcoat
107 like 7 but u=11 $ compact fill
111 like 1 but mat=12 u=12 rho=6.9542E-02 $ 9.9% kernel ---
112 like 2 but u=12 $ buffer
113 like 3 but u=12 $ IPyC
114 like 4 but u=12 $ SiC
115 like 5 but u=12 $ OPyC
116 like 6 but u=12 $ overcoat
117 like 7 but u=12 $ compact fill
c ----- Compacts -----
151 0 1001 -1002 1003 -1004 1005 -1006 imp:n=1 u=13 lat=1 fill=1
152 like 151 but u=14 fill=2
153 like 151 but u=15 fill=3
154 like 151 but u=16 fill=4
155 like 151 but u=17 fill=5
156 like 151 but u=18 fill=6
157 like 151 but u=19 fill=7
158 like 151 but u=20 fill=8
159 like 151 but u=21 fill=9
160 like 151 but u=22 fill=10
161 like 151 but u=23 fill=11
162 like 151 but u=24 fill=12
163 0 1011 -1012 1013 -1014 1015 -1016 imp:n=1 u=25 lat=1 fill=13
164 like 163 but u=26 fill=14
165 like 163 but u=27 fill=15
166 like 163 but u=28 fill=16
167 like 163 but u=29 fill=17
168 like 163 but u=30 fill=18
169 like 163 but u=31 fill=19
180 like 163 but u=32 fill=20
181 like 163 but u=33 fill=21
182 like 163 but u=34 fill=22
183 like 163 but u=35 fill=23
184 like 163 but u=36 fill=24
185 0 12 -13 imp:n=1 u=37 fill=25
186 like 185 but u=38 fill=26
187 like 185 but u=39 fill=27
188 like 185 but u=40 fill=28
189 like 185 but u=41 fill=29
190 like 185 but u=42 fill=30
191 like 185 but u=43 fill=31
192 like 185 but u=44 fill=32
193 like 185 but u=45 fill=33
194 like 185 but u=46 fill=34
195 like 185 but u=47 fill=35
196 like 185 but u=48 fill=36
c ----- Fuel Pins -----
251 27 2.4616E-05 -12 imp:n=1 u=37 $ central hole
252 like 251 but u=38
253 like 251 but u=39
254 like 251 but u=40
255 like 251 but u=41
256 like 251 but u=42
257 like 251 but u=43
258 like 251 but u=44
259 like 251 but u=45
260 like 251 but u=46
261 like 251 but u=47
262 like 251 but u=48
263 27 2.4616E-05 13 -21 imp:n=1 u=37 $ annulus between compact and sleeve
264 like 263 but u=38
265 like 263 but u=39
266 like 263 but u=40
267 like 263 but u=41
268 like 263 but u=42
269 like 263 but u=43
270 like 263 but u=44

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

271 like 263 but u=45
272 like 263 but u=46
273 like 263 but u=47
274 like 263 but u=48
275 19 8.8747E-02 21 -22 imp:n=1 u=37 $ graphite sleeve
1275 like 275 but u=38
276 like 275 but u=39
277 like 275 but u=40
278 like 275 but u=41
279 like 275 but u=42
280 like 275 but u=43
281 like 275 but u=44
282 like 275 but u=45
283 like 275 but u=46
284 like 275 but u=47
285 like 275 but u=48
c c ----- Coolant Channels -----
286 27 2.4616E-05 22 -31 imp:n=1 u=37 $ annulus between sleeve and block
287 like 286 but u=38
288 like 286 but u=39
289 like 286 but u=40
290 like 286 but u=41
291 like 286 but u=42
292 like 286 but u=43
293 like 286 but u=44
294 like 286 but u=45
295 like 286 but u=46
296 like 286 but u=47
297 like 286 but u=48
298 25 8.7804E-02 31 imp:n=1 u=37 $ graphite block
299 like 298 but u=38
300 like 298 but u=39
301 like 298 but u=40
302 like 298 but u=41
303 like 298 but u=42
304 like 298 but u=43
305 like 298 but u=44
306 like 298 but u=45
307 like 298 but u=46
308 like 298 but u=47
309 like 298 but u=48
c ----- BP Pins -----
351 20 9.0451E-02 -41 imp:n=1 u=50 $ 2.0%
352 like 351 but mat=21 rho=9.0501E-02 u=51 $ 2.5%
353 22 8.8747E-02 -42 imp:n=1 u=50 $ graphite disks
354 like 353 but u=51
355 20 9.0451E-02 -43 imp:n=1 u=50 $ 2.0%
356 like 355 but mat=21 rho=9.0501E-02 u=51 $ 2.5%
357 27 2.4616E-05 41 42 43 -44 imp:n=1 u=50 $ pin gap
358 like 357 but u=51
359 27 2.4616E-05 -44 imp:n=1 u=52 $ empty pin position
360 25 8.7804E-02 44 imp:n=1 u=50 $ graphite block
361 like 360 but u=51
362 like 360 but u=52
c ----- Blocks -----
401 25 8.7804E-02 -501 imp:n=1 u=61 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 4/5
61 10r
61 10r
61 4r      50 37 37 37      61 1r
61 3r      37 37 37 37 37      61 1r
61 2r      37 37 37 37 37 37      61 1r
61 1r 37 37 37 61 37 37 52 61 1r
61 1r      37 37 37 37 37 37      61 2r
61 1r      37 37 37 37 37      61 3r
61 1r      50 37 37 37      61 4r
61 10r
61 10r
402 25 8.7804E-02 -501 imp:n=1 u=62 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 3
62 10r
62 10r
62 4r      51 39 39 39      62 1r
62 3r      39 39 39 39 39      62 1r
62 2r      39 39 39 39 39 39      62 1r
62 1r 39 39 39 62 39 39 52 62 1r
62 1r      39 39 39 39 39 39      62 2r
62 1r      39 39 39 39 39      62 3r
62 1r      51 39 39 39      62 4r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

62 10r
62 10r
403 25 8.7804E-02 -501 imp:n=1 u=63 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 2
63 10r
63 10r
63 4r      51 41 41 41      63 1r
63 3r      41 41 41 41 41    63 1r
63 2r      41 41 41 41 41 41 63 1r
63 1r 41 41 41 41 63 41 41 52 63 1r
63 1r      41 41 41 41 41 41 63 2r
63 1r      41 41 41 41 41    63 3r
63 1r      51 41 41 41      63 4r
63 10r
63 10r
404 25 8.7804E-02 -501 imp:n=1 u=64 lat=2 fill= -5:5 -5:5 0:0 $ Zone 1 Lvl 1
64 10r
64 10r
64 4r      50 44 44 44      64 1r
64 3r      44 44 44 44 44    64 1r
64 2r      44 44 44 44 44 44 64 1r
64 1r 44 44 44 64 44 44 52 64 1r
64 1r      44 44 44 44 44 44 64 2r
64 1r      44 44 44 44 44    64 3r
64 1r      50 44 44 44      64 4r
64 10r
64 10r
405 25 8.7804E-02 -501 imp:n=1 u=65 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 4/5
65 10r
65 10r
65 4r      50 38 38 38      65 1r
65 3r      38 38 38 38 38    65 1r
65 2r      38 38 38 38 38 38 65 1r
65 1r 38 38 38 65 38 38 52 65 1r
65 1r      38 38 38 38 38 38 65 2r
65 1r      38 38 38 38 38    65 3r
65 1r      50 38 38 38      65 4r
65 10r
65 10r
406 25 8.7804E-02 -501 imp:n=1 u=66 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 3
66 10r
66 10r
66 4r      51 41 41 41      66 1r
66 3r      41 41 41 41 41    66 1r
66 2r      41 41 41 41 41 41 66 1r
66 1r 41 41 41 66 41 41 52 66 1r
66 1r      41 41 41 41 41 41 66 2r
66 1r      41 41 41 41 41    66 3r
66 1r      51 41 41 41      66 4r
66 10r
66 10r
407 25 8.7804E-02 -501 imp:n=1 u=67 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 2
67 10r
67 10r
67 4r      51 43 43 43      67 1r
67 3r      43 43 43 43 43    67 1r
67 2r      43 43 43 43 43 43 67 1r
67 1r 43 43 43 67 43 43 52 67 1r
67 1r      43 43 43 43 43 43 67 2r
67 1r      43 43 43 43 43    67 3r
67 1r      51 43 43 43      67 4r
67 10r
67 10r
408 25 8.7804E-02 -501 imp:n=1 u=68 lat=2 fill= -5:5 -5:5 0:0 $ Zone 2 Lvl 1
68 10r
68 10r
68 4r      50 46 46 46      68 1r
68 3r      46 46 46 46 46    68 1r
68 2r      46 46 46 46 46 46 68 1r
68 1r 46 46 46 68 46 46 52 68 1r
68 1r      46 46 46 46 46 46 68 2r
68 1r      46 46 46 46 46    68 3r
68 1r      50 46 46 46      68 4r
68 10r
68 10r
409 25 8.7804E-02 -501 imp:n=1 u=69 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 4/5
69 10r
69 10r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

69 4r      50 39 39 69      69 1r
69 3r      39 39 39 39 39  69 1r
69 2r      39 39 39 39 39  69 1r
69 1r 39 39 39 69 39 39 52 69 1r
69 1r      39 39 39 39 39  69 2r
69 1r      39 39 39 39 39  69 3r
69 1r      50 39 39 69      69 4r
69 10r
69 10r
410 25 8.7804E-02 -501 imp:n=1 u=70 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 3
70 10r
70 10r
70 4r      51 42 42 70      70 1r
70 3r      42 42 42 42 42  70 1r
70 2r      42 42 42 42 42  70 1r
70 1r 42 42 42 70 42 42 52 70 1r
70 1r      42 42 42 42 42  70 2r
70 1r      42 42 42 42 42  70 3r
70 1r      51 42 42 70      70 4r
70 10r
70 10r
411 25 8.7804E-02 -501 imp:n=1 u=71 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 2
71 10r
71 10r
71 4r      51 45 45 71      71 1r
71 3r      45 45 45 45 45  71 1r
71 2r      45 45 45 45 45  71 1r
71 1r 45 45 45 71 45 45 52 71 1r
71 1r      45 45 45 45 45  71 2r
71 1r      45 45 45 45 45  71 3r
71 1r      51 45 45 71      71 4r
71 10r
71 10r
412 25 8.7804E-02 -501 imp:n=1 u=72 lat=2 fill= -5:5 -5:5 0:0 $ Zone 3 Lvl 1
72 10r
72 10r
72 4r      50 47 47 72      72 1r
72 3r      47 47 47 47 47  72 1r
72 2r      47 47 47 47 47  72 1r
72 1r 47 47 47 72 47 47 52 72 1r
72 1r      47 47 47 47 47  72 2r
72 1r      47 47 47 47 47  72 3r
72 1r      50 47 47 72      72 4r
72 10r
72 10r
413 25 8.7804E-02 -501 imp:n=1 u=73 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 4/5
73 10r
73 10r
73 4r      50 40 40 73      73 1r
73 3r      40 40 40 40 40  73 1r
73 2r      40 40 40 40 40  73 1r
73 1r 40 40 40 73 40 40 52 73 1r
73 1r      40 40 40 40 40  73 2r
73 1r      40 40 40 40 40  73 3r
73 1r      50 40 40 73      73 4r
73 10r
73 10r
414 25 8.7804E-02 -501 imp:n=1 u=74 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 3
74 10r
74 10r
74 4r      51 43 43 74      74 1r
74 3r      43 43 43 43 43  74 1r
74 2r      43 43 43 43 43  74 1r
74 1r 43 43 43 74 43 43 52 74 1r
74 1r      43 43 43 43 43  74 2r
74 1r      43 43 43 43 43  74 3r
74 1r      51 43 43 74      74 4r
74 10r
74 10r
415 25 8.7804E-02 -501 imp:n=1 u=75 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 2
75 10r
75 10r
75 4r      51 46 46 75      75 1r
75 3r      46 46 46 46 46  75 1r
75 2r      46 46 46 46 46  75 1r
75 1r 46 46 46 75 46 46 52 75 1r
75 1r      46 46 46 46 46  75 2r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

75 1r      46 46 46 46 46      75 3r
75 1r      51 46 46 75      75 4r
75 10r
75 10r
416 25 8.7804E-02 -501 imp:n=1 u=76 lat=2 fill= -5:5 -5:5 0:0 $ Zone 4 Lvl 1
76 10r
76 10r
76 4r      50 48 48 76      76 1r
76 3r      48 48 48 48 48      76 1r
76 2r 48 48 48 48 48 48 76 1r
76 1r 48 48 48 76 48 48 52 76 1r
76 1r 48 48 48 48 48 48 76 2r
76 1r      48 48 48 48 48      76 3r
76 1r      50 48 48 76      76 4r
76 10r
76 10r
451 0 -502 imp:n=1 u=81 fill=61
452 like 451 but u=82 fill=62
453 like 451 but u=83 fill=63
454 like 451 but u=84 fill=64
455 like 451 but u=85 fill=65
456 like 451 but u=86 fill=66
457 like 451 but u=87 fill=67
458 like 451 but u=88 fill=68
459 like 451 but u=89 fill=69
460 like 451 but u=90 fill=70
461 like 451 but u=91 fill=71
462 like 451 but u=92 fill=72
463 like 451 but u=93 fill=73
464 like 451 but u=94 fill=74
465 like 451 but u=95 fill=75
466 like 451 but u=96 fill=76
467 27 2.4616E-05 502 imp:n=1 u=81
468 like 467 but u=82
469 like 467 but u=83
470 like 467 but u=84
471 like 467 but u=85
472 like 467 but u=86
473 like 467 but u=87
474 like 467 but u=88
475 like 467 but u=89
476 like 467 but u=90
477 like 467 but u=91
478 like 467 but u=92
479 like 467 but u=93
480 like 467 but u=94
481 like 467 but u=95
482 like 467 but u=96
c ----- Reflectors -----
483 27 2.4616E-05 -201 imp:n=1 u=97 $ coolant channels
484 25 8.7804E-02 201 imp:n=1 u=97 $ graphite block
485 25 8.7804E-02 -501 imp:n=1 u=98 lat=2 fill= -5:5 -5:5 0:0 $ 33-hole
98 10r
98 10r
98 4r      98 97 97 97      98 1r
98 3r      97 97 97 97 97      98 1r
98 2r 97 97 97 97 97 97 98 1r
98 1r 97 97 97 98 97 97 98 98 1r
98 1r 97 97 97 97 97 97 98 2r
98 1r      97 97 97 97 97      98 3r
98 1r      98 97 97 97      98 4r
98 10r
98 10r
486 25 8.7804E-02 -501 imp:n=1 u=99 lat=2 fill= -5:5 -5:5 0:0 $ 31-hole
99 10r
99 10r
99 4r      99 97 97 99      99 1r
99 3r      97 97 97 97 97      99 1r
99 2r 97 97 97 97 97 97 99 1r
99 1r 97 97 97 99 97 97 99 99 1r
99 1r 97 97 97 97 97 97 99 2r
99 1r      97 97 97 97 97      99 3r
99 1r      99 97 97 99      99 4r
99 10r
99 10r
489 0 -502 imp:n=1 u=100 fill=98
490 0 -502 imp:n=1 u=101 fill=99

```



## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

491 like 467 but u=100
492 like 467 but u=101
c ----- Columns -----
501 27 2.4616E-05 -503 imp:n=1 u=121 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 1
    121 121 121 121 121 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 81 121 121 121 121
    121 121 121 121 81 121 121 121 121
    121 121 121 121 82 121 121 121 121
    121 121 121 121 83 121 121 121 121
    121 121 121 121 84 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 100 121 121 121 121
    121 121 121 121 121 121 121 121 121
502 27 2.4616E-05 -503 imp:n=1 u=122 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 2
    122 122 122 122 122 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 85 122 122 122 122
    122 122 122 122 85 122 122 122 122
    122 122 122 122 86 122 122 122 122
    122 122 122 122 87 122 122 122 122
    122 122 122 122 88 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 100 122 122 122 122
    122 122 122 122 122 122 122 122 122
503 27 2.4616E-05 -503 imp:n=1 u=123 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 3
    123 123 123 123 123 123 123 123 123
    123 123 123 123 100 123 123 123 123
    123 123 123 123 100 123 123 123 123
    123 123 123 123 89 123 123 123 123
    123 123 123 123 89 123 123 123 123
    123 123 123 123 90 123 123 123 123
    123 123 123 123 91 123 123 123 123
    123 123 123 123 92 123 123 123 123
    123 123 123 123 101 123 123 123 123
    123 123 123 123 101 123 123 123 123
    123 123 123 123 123 123 123 123 123
504 27 2.4616E-05 -503 imp:n=1 u=124 lat=2 fill= -1:1 -1:1 -5:5 $ Zone 4
    124 124 124 124 124 124 124 124 124
    124 124 124 124 100 124 124 124 124
    124 124 124 124 100 124 124 124 124
    124 124 124 124 93 124 124 124 124
    124 124 124 124 93 124 124 124 124
    124 124 124 124 94 124 124 124 124
    124 124 124 124 95 124 124 124 124
    124 124 124 124 96 124 124 124 124
    124 124 124 124 101 124 124 124 124
    124 124 124 124 101 124 124 124 124
    124 124 124 124 124 124 124 124 124
525 0 -950 imp:n=1 u=125 fill=121 $ B04
526 like 525 but u=126 fill=121 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ B01
527 like 525 but u=127 fill=121 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ B02
528 like 525 but u=128 fill=121 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ B03
529 like 525 but u=129 fill=121 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ B05
530 like 525 but u=130 fill=121 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ B06
531 0 -950 imp:n=1 u=131 fill=122 $ C02
532 like 531 but u=132 fill=122 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ C04
533 like 531 but u=133 fill=122 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ C06
534 like 531 but u=134 fill=122 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ C08
535 like 531 but u=135 fill=122 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ C10
536 like 531 but u=136 fill=122 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ C12
537 0 -950 imp:n=1 u=137 fill=123 $ D05/09
538 like 537 but u=138 fill=123 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ D02/06
539 like 537 but u=139 fill=123 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ D03/17
540 like 537 but u=140 fill=123 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ D08/12
541 like 537 but u=141 fill=123 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ D11/15
542 like 537 but u=142 fill=123 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ D14/18
543 0 -950 imp:n=1 u=143 fill=124 $ D07
544 like 543 but u=144 fill=124 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ D01
545 like 543 but u=145 fill=124 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ D04
546 like 543 but u=146 fill=124 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ D10
547 like 543 but u=147 fill=124 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ D13
548 like 543 but u=148 fill=124 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ D16
c
c --- Control Column -----

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

c ----- Control Rod Segments -----
601 24 8.7530E-02 -103 imp:n=1 u=150 $ spine
602 27 2.4616E-05 103 -104 imp:n=1 u=150 $ helium gap
603 24 8.7530E-02 104 -101 imp:n=1 u=150 $ inner clad
604 23 9.1660E-02 101 -102 imp:n=1 u=150 $ absorber
605 24 8.7530E-02 102 -105 imp:n=1 u=150 $ outer clad
606 27 2.4616E-05 105 imp:n=1 u=150 $ helium
607 0 -151 imp:n=1 u=151 lat=2 fill=150 $ rod segment
608 0 -152 imp:n=1 u=152 fill=151 $ control rod
609 27 2.4616E-05 152 imp:n=1 u=152 $ helium
c ----- Positions -----
610 0 -999 imp:n=1 u=153 trcl=(0 0 177.5) fill=152 $ C
611 0 -153 imp:n=1 u=154 trcl=(-5.4 -9.35307 0) fill=153
612 like 611 but trcl=(10.8 0 0)
613 27 2.4616E-05 -153 imp:n=1 u=154 trcl=(-5.4 9.35307 0) $ RSS
614 like 610 but u=155 trcl=(0 0 177.5) $ R1
615 like 611 but u=156 fill=155
616 like 612 but u=156 fill=155
617 like 613 but u=156
618 like 610 but u=157 trcl=(0 0 177.5) $ R2
619 like 611 but u=158 fill=157
620 like 612 but u=158 fill=157
621 like 613 but u=158
622 like 610 but u=159 trcl=(0 0 404.9) $ R3
623 like 611 but u=160 fill=159
624 like 612 but u=160 fill=159
625 like 613 but u=160
c ----- C Column -----
626 25 8.7804E-02 #611 #612 #613 -550 imp:n=1 u=154 $ graphite blocks
627 27 2.4616E-05 550 imp:n=1 u=154
628 0 -950 imp:n=1 u=161 fill=154 $ A01
c
c ----- R1 Columns -----
629 25 8.7804E-02 #615 #616 #617 -550 imp:n=1 u=156 $ graphite blocks
630 like 627 but u=156
631 like 628 but u=162 fill=156 $ C01
632 like 631 but u=163 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ C03
633 like 631 but u=164 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ C05
634 like 631 but u=165 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ C07
635 like 631 but u=166 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ C09
636 like 631 but u=167 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ C11
c ----- R2 Columns -----
637 25 8.7804E-02 #619 #620 #621 -550 imp:n=1 u=158 $ graphite blocks
638 like 627 but u=158
639 like 628 but u=168 fill=158 $ E23
640 like 639 but u=169 *trcl=(0 0 0 300 210 90 30 300 90 90 90 0) $ E03
641 like 639 but u=170 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ E07
642 like 639 but u=171 *trcl=(0 0 0 180 90 90 270 180 90 90 90 0) $ E11
643 like 639 but u=172 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ E15
644 like 639 but u=173 *trcl=(0 0 0 60 330 90 150 60 90 90 90 0) $ E19
c ----- R3 Columns -----
645 25 8.7804E-02 #623 #624 #625 -550 imp:n=1 u=160 $ graphite blocks
646 like 627 but u=160
647 like 628 but u=174 fill=160 $ E01
648 like 647 but u=175 *trcl=(0 0 0 240 150 90 330 240 90 90 90 0) $ E09
649 like 647 but u=176 *trcl=(0 0 0 120 30 90 210 120 90 90 90 0) $ E17
c
c --- Instrumentation Column -----
c ----- Positions -----
661 like 613 but u=181 trcl=(5.4 9.35307 0)
662 like 613 but u=181 trcl=(5.4 -9.35307 0)
663 like 613 but u=181 trcl=(-10.8 0 0)
c ----- Columns -----
664 25 8.7804E-02 #661 #662 #663 -550 imp:n=1 u=181 $ graphite blocks
665 like 627 but u=181
666 0 -950 imp:n=1 u=182 fill=181
c
c --- Reflector Column -----
c ----- Columns -----
671 25 8.7804E-02 -550 imp:n=1 u=183 $ graphite blocks
672 like 627 but u=183
673 0 -950 imp:n=1 u=184 fill=183
c
c --- HTTR Core -----
c ----- Core Map -----
701 26 8.6134E-02 -551 imp:n=1 lat=2 u=200 fill=-6:6 -6:6 0:0
200 12r

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE

```

200 12r
200 5r      175 184 170 184 182      200 1r
200 4r      184 143 138 137 145 184      200 1r
200 3r      171 140 164 132 163 139 169      200 1r
200 2r      184 137 133 128 127 131 138 184      200 1r
200 1r 182 146 165 125 161 126 162 144 174      200 1r
200 1r      184 141 134 129 130 136 142 184      200 2r
200 1r      172 140 166 135 167 139 168      200 3r
200 1r      184 147 142 141 148 184      200 4r
200 1r      176 184 173 184 182      200 5r
200 12r
200 12r

c
c ----- Core Map Legend ---
c   u 12r
c   u 12r
c   u 5r      Z G Y G I      u 1r
c   u 4r      G 4 3 3 4 G      u 1r
c   u 3r      Y 3 X 2 X 3 Y      u 1r
c   u 2r      G 3 2 1 1 2 3 G      u 1r
c   u 1r      I 4 X 1 C 1 X 4 Z      u 1r
c   u 1r      G 3 2 1 1 2 3 G      u 2r
c   u 1r      Y 3 X 2 X 3 Y      u 3r
c   u 1r      G 4 3 3 4 G      u 4r
c   u 1r      Z G Y G I      u 5r
c   u 12r
c   u 12r
c
c   1 = Fuel Columns #1
c   2 = Fuel Columns #2
c   3 = Fuel Columns #3
c   4 = Fuel Columns #4
c   C = Central Control Column
c   X = R1 Control Columns
c   Y = R2 Control Columns
c   Z = R3 Control Columns
c   I = Instrumentation Columns
c   G = Removable Reflector Columns
c
702  0  -602 imp:n=1 fill=200
c
c --- Permanent Reflector -----
711  26 8.6134E-02 602 -651 imp:n=1
c
c --- The Great Void -----
999  0 651 imp:n=0
c

c Surface Cards *****
c --- Fuel Blocks -----
c ----- TRISO Particles -----
1   so 0.03  $ UO2 kernal
2   so 0.036 $ buffer
3   so 0.039 $ IPyC
4   so 0.0415 $ SiC
5   so 0.046  $ OPyC
6   so 0.066  $ overcoat
1001 px -0.0553839
1002 px 0.0553839
1003 py -0.0553839
1004 py 0.0553839
1005 pz -0.05455402
1006 pz 0.05455402
c ----- Compacts -----
1011 px -1.4
1012 px 1.4
1013 py -1.4
1014 py 1.4
1015 pz -0.05455402
1016 pz 0.05455402
12  rcc 0 0 -27.3 0 0 54.6 0.5 $ inside
13  rcc 0 0 -27.3 0 0 54.6 1.3 $ outside
c ----- Fuel Pins -----
21  rcc 0 0 -27.45 0 0 54.9 1.325 $ inside
22  rcc 0 0 -28.85 0 0 57.7 1.7  $ outside
c ----- Coolant Channels -----
31  rcc 0 0 -31 0 0 62 2.05

```

## Gas Cooled (Thermal) Reactor - GCR

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CRIT-SUB-REAC-COEF-KIN-RRATE

```

c ----- BP Pins -----
41  rcc 0 0 -25 0 0 20 0.7  $ BP
42  rcc 0 0 -5 0 0 10 0.7  $ graphite
43  rcc 0 0 5 0 0 20 0.7  $ BP
44  rcc 0 0 -25 0 0 50 0.75 $ pin
c
c --- Control Blocks -----
c ----- Control Rod Segments -----
101 rcc 0 0 11 0 0 29 3.75 $ inside
102 rcc 0 0 11 0 0 29 5.25 $ outside
103 rcc 0 0 11 0 0 29 0.5 $ spine
104 rcc 0 0 11 0 0 29 3.25 $ inside clad
105 rcc 0 0 10 0 0 31 5.65 $ outside clad
c ----- Positions -----
151 hex 0 0 10 0 0 31 10 $ "box"
152 rcc 0 0 -145 0 0 310 6.30 $ control rod
153 rcc 0 0 -155 0 0 416 6.15 $ control rod hole
c
c --- Reflector Blocks -----
c ----- Coolant Channels -----
201 rcc 0 0 -31 0 0 62 1.15
c
c --- Blocks -----
501 hex 0 0 -30 0 0 60 2.575 0 0 $ pitch
502 hex 0 0 -29.5 0 0 59 0 18 0 $ graphite
503 hex 0 0 -29 0 0 58 0 18.2 0 $ helium
c
c --- Columns -----
550 hex 0 0 -261.5 0 0 523 0 18 0
551 hex 0 0 -261 0 0 522 0 18.1 0
c
c --- HTTR Core -----
602 hex 0 0 -261 0 0 522 -148 0 0
c
c --- Permanent Reflector -----
651 rcc 0 0 -261 0 0 522 212.5
c
950 rcc 0 0 -1000 0 0 2000 25 $ small cylinder
999 rcc 0 0 -2500 0 0 5000 2500 $ big cylinder
c

c Data Cards *****
c --- Material Cards -----
c ----- Kernel (3.4%) -----
m1  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 6.9139E-06
    92235.66c 7.9784E-04
    92238.66c 2.2375E-02
c    Total      6.9523E-02
c
c ----- Kernel (3.9%) -----
m2  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 7.9307E-06
    92235.66c 9.1517E-04
    92238.66c 2.2258E-02
c    Total      6.9524E-02
c
c ----- Kernel (4.3%) -----
m3  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 8.7441E-06
    92235.66c 1.0090E-03
    92238.66c 2.2165E-02
c    Total      6.9526E-02
c
c ----- Kernel (4.8%) -----
m4  5010.66c 1.7276E-07
    8016.62c 4.6325E-02
    8017.66c 1.7610E-05
    92234.66c 9.7608E-06
    92235.66c 1.1264E-03
    92238.66c 2.2048E-02

```

## Gas Cooled (Thermal) Reactor - GCR

HTTR-GCR-RESR-001  
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```

c      Total      6.9527E-02
c
c ----- Kernel (5.2%) -----
m5      5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.0574E-05
          92235.66c 1.2202E-03
          92238.66c 2.1954E-02
c      Total      6.9528E-02
c
c ----- Kernel (5.9%) -----
m6      5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.1998E-05
          92235.66c 1.3845E-03
          92238.66c 2.1791E-02
c      Total      6.9530E-02
c
c ----- Kernel (6.3%) -----
m7      5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.2811E-05
          92235.66c 1.4783E-03
          92238.66c 2.1697E-02
c      Total      6.9532E-02
c
c ----- Kernel (6.7%) -----
m8      5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.3624E-05
          92235.66c 1.5722E-03
          92238.66c 2.1604E-02
c      Total      6.9533E-02
c
c ----- Kernel (7.2%) -----
m9      5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.4641E-05
          92235.66c 1.6895E-03
          92238.66c 2.1487E-02
c      Total      6.9534E-02
c
c ----- Kernel (7.9%) -----
m10     5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.6065E-05
          92235.66c 1.8538E-03
          92238.66c 2.1323E-02
c      Total      6.9536E-02
c
c ----- Kernel (9.4%) -----
m11     5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 1.9115E-05
          92235.66c 2.2058E-03
          92238.66c 2.0973E-02
c      Total      6.9541E-02
c
c ----- Kernel (9.9%) -----
m12     5010.66c 1.7276E-07
          8016.62c 4.6325E-02
          8017.66c 1.7610E-05
          92234.66c 2.0132E-05
          92235.66c 2.3231E-03
          92238.66c 2.0856E-02
c      Total      6.9542E-02
c
c ----- Buffer Layer -----
m13     5010.66c 1.8290E-08
          6000.66c 5.5153E-02

```

## Gas Cooled (Thermal) Reactor - GCR

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```

c      Total      5.5153E-02
mt13  grph.60t
c
c ----- IPyC Layer -----
m14   5010.66c 3.0761E-08
      6000.66c 9.2758E-02
c      Total      9.2758E-02
mt14  grph.60t
c
c ----- SiC Layer -----
m15   5010.66c 5.3208E-08
      6000.66c 8.0223E-02
      14028.62c 3.1641E-02
      14029.62c 1.6067E-03
      14030.62c 1.0591E-03
c      Total      1.1453E-01
c
c ----- OPyC Layer -----
m16   5010.66c 3.0761E-08
      6000.66c 9.2758E-02
c      Total      9.2758E-02
mt16  grph.60t
c
c ----- Graphite Overcoat -----
m17   5010.66c 2.8267E-08
      6000.66c 8.5237E-02
c      Total      8.5237E-02
mt17  grph.60t
c
c ----- Graphite Compact -----
m18   5010.66c 1.5452E-08
      6000.66c 8.5237E-02
c      Total      8.5237E-02
mt18  grph.60t
c
c ----- Graphite Sleeve -----
m19   5010.66c 7.2596E-09
      6000.66c 8.8747E-02
c      Total      8.8747E-02
mt19  grph.60t
c
c ----- Burnable Poison (2.0%) -----
m20   5010.66c 3.9906E-04
      5011.66c 1.6063E-03
      6000.66c 8.8446E-02
c      Total      9.0451E-02
c
c ----- Burnable Poison (2.5%) -----
m21   5010.66c 4.9882E-04
      5011.66c 2.0078E-03
      6000.66c 8.7995E-02
c      Total      9.0501E-02
c
c ----- Graphite Disks -----
m22   5010.66c 7.2596E-09
      6000.66c 8.8747E-02
c      Total      8.8747E-02
mt22  grph.60t
c
c ----- Neutron Absorber -----
m23   5010.66c 6.8220E-03
      5011.66c 2.4974E-02
      6000.66c 6.6685E-02
c      Total      9.1660E-02
c
c ----- Alloy 800H -----
m24   6000.66c 3.2210E-04
      13027.62c 6.7209E-04
      14028.62c 5.5580E-04
      14029.62c 2.8222E-05
      14030.62c 1.8604E-05
      15031.66c 3.1225E-05
      16000.62c 1.5081E-05
      22000.62c 3.7884E-04
      24050.62c 8.4860E-04
      24052.62c 1.6364E-02
      24053.62c 1.8556E-03

```

## Gas Cooled (Thermal) Reactor - GCR

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```

24054.62c 4.6189E-04
25055.62c 8.8022E-04
26054.62c 2.2265E-03
26056.62c 3.4951E-02
26057.62c 8.0717E-04
26058.62c 1.0742E-04
28058.62c 1.8229E-02
28060.62c 7.0217E-03
28061.62c 3.0523E-04
28062.62c 9.7320E-04
28064.62c 2.4785E-04
29063.62c 1.5791E-04
29065.62c 7.0383E-05
c   Total      8.7530E-02
c
c ----- IG-110 Graphite -----
m25   5010.66c 7.7648E-09
      6000.66c 8.7804E-02
c   Total      8.7804E-02
mt25  grph.60t
c
c ----- PGX Graphite -----
m26   5010.66c 3.6372E-08
      6000.66c 8.6134E-02
c   Total      8.6134E-02
mt26  grph.60t
c
c ----- Helium Coolant -----
m27   2003.66c 3.3724E-11
      2004.66c 2.4616E-05
c   Total      2.4616E-05
c
c --- Control Cards -----
mode  n
kcode 10000 1 50 3550
ksrc  0 0 0
c print

```

**APPENDIX B: CALCULATED SPECTRAL DATA****B.1 Spectral Data for the Critical and Subcritical Configurations**

Data generated in the MCNP5 output files include information regarding the energy of the average lethargy causing fission (EALF) and the percentages of fission caused by thermal, intermediate, and fast energy neutrons for each case shown in Tables 4.1 and 4.2; results are shown in Tables B.1 and B.2, respectively. There was no significant difference between libraries in the spectra. It should be noted, that cases without cross section data to account for the upscatter of thermal neutrons have EALF values approximately 20% smaller. The MCNP5 calculations were performed with 3,550 generations (skipping the first 50) and 10,000 neutrons per generation. There is no significant difference between the results shown in Tables B.1 and B.2.

Table B.1. Spectral Data for the HTTR Benchmark Model Evaluation using an Ordered Lattice.

Neutron Cross-Section Library	EALF (eV)	Percentage of Neutrons Causing Fission		
		<0.625 eV	0.625 eV – 100 keV	>100 keV
ENDF/B-V.2	0.0793	92.23 %	6.86 %	0.91 %
ENDF/B-VI.8	0.0775	92.41 %	6.69 %	0.90 %
ENDF/B-VI.8 <sup>(a)</sup> [no S( $\alpha,\beta$ )]	0.0616	92.64 %	6.48 %	0.88 %
END/B-VII.0	0.0775	92.42 %	6.68 %	0.90 %
JEFF-3.1	0.0772	92.41 %	6.69 %	0.90 %
JENDL-3.3	0.0619	92.61 %	6.50 %	0.89 %
JENDL-3.3 with ENDF/B-VI.8 S( $\alpha,\beta$ ) <sup>(a)</sup>	0.0779	92.38 %	6.71 %	0.91 %

(a) The JENDL-3.3 library did not include additional cross-section data for thermal scattering treatment. Cases were performed for ENDF/B-VI.8 without thermal neutron scattering correction to compare with the JENDL-3.3 results, and JENDL-3.3 was also analyzed using thermal neutron scattering correction from the ENDF/B-VI.8 library.



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Table B.2. Spectral Data for the HTTR Benchmark Model Evaluation using a Uniformly-Filled Lattice.

Neutron Cross-Section Library	EALF (eV)	Percentage of Neutrons Causing Fission		
		<0.625 eV	0.625 eV – 100 keV	>100 keV
ENDF/B-V.2	0.0785	92.22 %	6.88 %	0.90 %
ENDF/B-VI.8	0.0777	92.37 %	6.74 %	0.89 %
ENDF/B-VI.8 <sup>(a)</sup> [no S( $\alpha,\beta$ )]	0.0617	92.60 %	6.52 %	0.88 %
END/B-VII.0	0.0778	92.37 %	6.73 %	0.90 %
JEFF-3.1	0.0774	92.38 %	6.73 %	0.89 %
JENDL-3.3	0.0620	92.57 %	6.54 %	0.89 %
JENDL-3.3 with ENDF/B-VI.8 S( $\alpha,\beta$ ) <sup>(a)</sup>	0.0781	92.33 %	6.76 %	0.90 %

(a) The JENDL-3.3 library did not include additional cross-section data for thermal scattering treatment. Cases were performed for ENDF/B-VI.8 without thermal neutron scattering correction to compare with the JENDL-3.3 results, and JENDL-3.3 was also analyzed using thermal neutron scattering correction from the ENDF/B-VI.8 library.

**APPENDIX C: DATA FROM THE 16<sup>TH</sup> EDITION CHART OF THE NUCLIDES<sup>a</sup>****C.1 Isotopic Abundances and Atomic Weights**

This evaluation incorporated atomic weights and isotopic abundances found in the 16<sup>th</sup> edition of the Chart of the Nuclides. A list of the values used in the benchmark model or in the generation of the MCNP input deck is compiled in Table C.1.

Table C.1. Summary of Data Employed from the 16<sup>th</sup> Ed. of the Chart of the Nuclides.

Isotope or Element	Atomic Weight	Isotopic Abundance
He	4.002602	--
<sup>3</sup> He	--	0.000137
<sup>4</sup> He	--	99.999863
<sup>10</sup> B	10.0129370	19.9
<sup>11</sup> B	11.0093055	80.1
C	12.0107	--
O	15.9994	--
<sup>16</sup> O	--	99.757
<sup>17</sup> O	--	0.038
<sup>18</sup> O <sup>(a)</sup>	--	0.205
Al	26.981538	--
Si	28.0855	--
<sup>28</sup> Si	--	92.2297
<sup>29</sup> Si	--	4.6832
<sup>30</sup> Si	--	3.0872
P	30.973761	--
S	32.065	--
Ti	47.867	--

(a) Neutronically, <sup>18</sup>O is treated as <sup>16</sup>O.

<sup>a</sup> Nuclides and Isotopes: Chart of the Nuclides, 16<sup>th</sup> edition, (2002).

Table C.1 (cont.). Summary of Data Employed from the 16<sup>th</sup> Ed. of the Chart of the Nuclides.

Isotope or Element	Atomic Weight	Isotopic Abundance
Cr	51.9961	--
<sup>50</sup> Cr	--	4.345
<sup>52</sup> Cr	--	83.789
<sup>53</sup> Cr	--	9.501
<sup>54</sup> Cr	--	2.365
Mn	54.938049	--
Fe	55.845	--
<sup>54</sup> Fe	--	5.845
<sup>56</sup> Fe	--	91.754
<sup>57</sup> Fe	--	2.119
<sup>58</sup> Fe	--	0.282
Ni	58.6934	--
<sup>58</sup> Ni	--	68.0769
<sup>60</sup> Ni	--	26.2231
<sup>61</sup> Ni	--	1.1399
<sup>62</sup> Ni	--	3.6345
<sup>64</sup> Ni	--	0.9256
Cu	63.546	--
<sup>63</sup> Cu	--	69.17
<sup>65</sup> Cu	--	30.83
<sup>234</sup> U	234.040946	0.0055 <sup>(b)</sup>
<sup>235</sup> U	235.043923	0.7200 <sup>(b)</sup>
<sup>238</sup> U	238.050783	99.2745 <sup>(b)</sup>

(b) Natural isotopic abundance of U.

## Summary Information

- 1.0 Experiment identification number: HTTR-GCR-RESR-001  
CRIT-SUB-REAC-COEF-KIN-RRATE
- 2.0 Date: July 1998 – January 1999
- 3.0 Name of experiment: Evaluation of the Start-Up Core Physics Tests at Japan's High Temperature Engineering Test Reactor (Fully-Loaded Core)
  - 3.1. Purpose of experiment: Perform start-up core physics tests of a HTGR prototype reactor
  - 3.2. Phenomena measured and scope: Criticality, Reaction Rates, Reactivity Worth (Excess Reactivity, Shutdown Margin, Control Rods), Temperature Coefficient
- 4.0 Name or designation of experimental programme: HTTR Start-Up Core Physics Tests
- 5.0 Description of facility: High Temperature Engineering Test Reactor (HTTR), at JAERI Oarai Research Establishment
- 6.0 Description of test or experiment:
  - 6.1 Experimental configuration:
    - 6.1.1 Type of assemblies: Clean, pin-in-block, prismatic TRISO LEU, engineering benchmark facility, zero power
    - 6.1.2 Assembly details: (See detailed description.)
      - 6.1.2.1 Type: Hexagonal blocks in columns installed vertically in a triangular-pitched lattice
      - 6.1.2.2 Fuel: LEU-TRISO sintered in graphite compacts, enclosed within a graphite pin that is inserted into graphite blocks to form assemblies
      - 6.1.2.3 Moderator: IG-110 graphite
      - 6.1.2.4 Absorbers: Sintered boron carbide ( $B_4C$ ) annular pellets, enclosed with alloy 800H control rods, motor-driven vertically
      - 6.1.2.5 Critical mass: The first criticality was achieved with 19 core fuel subassemblies, containing approximately 35.3 kg of  $^{235}U$  (557.3 kg U). Full core criticality with 30 fuel subassemblies contained approximately 52.4 kg  $^{235}U$  (899.5 kg U).
      - 6.1.2.6 Core volume: Approximately 12.05 m<sup>3</sup> (volume of the active core region)
      - 6.1.2.7 Blanket: None
      - 6.1.2.8 Reflectors: IG-110 and PGX graphite
      - 6.1.2.9 Reactivity adjustment: 16 pairs – 7 in the fueled region  
9 in the reflector region
      - 6.1.2.10 Other: Coolant: Forced helium downward through core components
    - 6.1.3 Assembly variants: During fuel loading, critical annular cores were tested
  - 6.2 Core life cycle: The first critical core layouts during initial fuel loading.
  - 6.3 Experimental limitations or shortcomings: Unknown

## 7.0 Phenomena measured

### 7.1 Description of results and analysis:

- 7.1.1 Reactivity control for criticality: (§1.1.2.4)
- 7.1.2 Reaction rates / ratios: axial neutron flux (not evaluated)
- 7.1.3 Reactivity worth: excess reactivity & shutdown margin (not evaluated)
- 7.1.4 Sample Doppler reactivity: N/A
- 7.1.5 Temperature coefficients: (not evaluated)
- 7.1.6 Control rod or rod-bank worths: (not evaluated)
- 7.1.7 Soluble boron worth: N/A
- 7.1.8 Gamma heating distributions: N/A
- 7.1.9 Neutron spectrum: N/A
- 7.1.10 Kinetic parameters: N/A
- 7.1.11 Reactor power distributions: N/A
- 7.1.12 Isotopic measurements: N/A

### 7.2 Special features and characteristics of experiment:

- 7.2.1 Moderator/fuel ratio: N/A
- 7.2.2 Measured spectral indices: N/A

### 7.3 Measurement systems (methods) and uncertainties:

- 7.3.1 Rod heights ( $\pm 5$  mm) for critical configurations
- 7.3.2 Fission counters for  $^{235}\text{U}$  reaction rate (proportional to neutron flux)
- 7.3.3 Inverse kinetics, rod drop, or fuel addition for reactivity worths and control rod worths
- 7.3.4 Rod heights ( $\pm 5$  mm) for critical conditions at different temperatures for temperature coefficient

## 8.0 Duplicate or complementary experiments / other related experiments:

- 8.1 Very high temperature reactor critical assembly (VHTRC) reactor physics experiments
- 8.2 HTTR rise-to-power tests
- 8.3 HTTR safety demonstration tests
- 8.4 HTTR irradiation tests
- 8.5 HTTR demonstration tests of hydrogen production

## 9.0 Status of completion of the evaluation: Completed

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- 11.0 Authors/ organizers:
  - 11.1 Establishment: JAERI Oarai Research Establishment, HTTR Facility
  - 11.2 Staff involved in experiment: Unknown
  - 11.3 Contact information: <http://httr.jaea.go.jp/>
  - 11.4 Reviewers of compiled data: John Darrell Bess (Idaho National Laboratory)  
Nozomu Fujimoto (Japan Atomic Energy Agency)
    - 11.4.1 Internal reviewers: Barbara Dolphin (Idaho National Laboratory)
    - 11.4.2 Independent reviewers: Luka Snoj (Jožef Stefan Institute)  
Atsushi Zukeran (Senior Reactor Physics Consultant)
- 12.0 Material available:
  - 12.1 Data and Format: NA