PRISMATIC COUPLED NEUTRONICS/THERMAL FLUIDS TRANSIENT BENCHMARK OF THE MHTGR-350 MW CORE DESIGN

BENCHMARK DEFINITION

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
EOEC	End of Equilibrium Cycle
HTR	High Temperature Reactor
LBP	Lumped Burnable Poison
MCSS	Metallic Core Support Structure
NSS	Nuclear Steam Supply
PMR	Prismatic Modular Reactor
RCCS	Reactor Cavity Cooling System
RSC	Reserve Shutdown Control
UPTPS	Upper Plenum Thermal Protection Structure

1. INTRODUCTION

The Prismatic Modular Reactor (PMR) is one of the High Temperature Reactor (HTR) design concepts that have existed for some time. Several prismatic units have operated in the world (DRAGON, Fort St. Vrain, Peach Bottom) and one unit is still in operation (HTTR). The deterministic neutronics, thermal-fluids and transient analysis tools and methods available to design and analyse PMRs have, in many cases, lagged behind the state of the art compared to other reactor technologies. This has motivated the testing of existing methods for HTGRs but also the development of more accurate and efficient tools to analyze the neutronics and thermal-hydraulic behaviour for the design and safety evaluations of the PMR. In addition to the development of new methods, this includes defining appropriate benchmarks to verify and validate the new methods in computer codes.

Benchmark exercises provide some of the best avenues for the verification of current analysis tools. A very good example was the PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark for the PBMR-400 Core Design [1], which served as the foundation for this document.

2. GOVERNANCE

2.1 SPONSORSHIP

The first workshop for {*Prismatic Benchmark Exercise (PCCTBWG)*} the will be held on {*dates*} at the OECD headquarters in Paris, France with the support of the Nuclear Science Committee (NSC) of the NEA of OECD and the supervision of the Working Party on Scientific Issues in Reactor Systems (WPRS).

2.2 PARTICIPATION IN THE BENCHMARK AND WORKSHOPS

Participation in the Benchmark Workshops is sponsored by the Nuclear Science Committee (NSC), and is restricted, for efficiency, to experts (research laboratories, safety authorities, regulatory agencies, utilities, owners' groups, vendors, etc.) from OECD Member countries nominated by delegates to the Committees in consultation with official authorities concerned and with the assistance of members of the Nuclear Science Committee (information about members are provided as Annex II) and in particular to participants in this study.

The meeting is open also to experts from IAEA member countries, which are in a position to provide a substantive contribution to this study.

2.3 ORGANIZATION AND PROGRAM COMMITTEE OF THE BENCHMARK WORKSHOPS

An Organization and Programme Committee has been proposed to make the necessary arrangements for the First Benchmark Workshop and to organize the Sessions, draw up the final programme, appoint Session Chairmen, etc. Its proposed members are:

3. SCOPE AND TECHNICAL CONTENT OF THE BENCHMARK

The scope of the benchmark is twofold: 1) to establish a well-defined problem, based on a common given set of cross sections, to compare methods and tools in core simulation and thermal hydraulics analysis with a specific focus on transient events through a set of multi-dimensional computational test problems, 2) to tests the cross section preparation capabilities available at this time.

In addition the benchmark exercise has the following objectives:

• Establish a standard benchmark for coupled codes (neutronics/thermal-hydraulics) for PMR design.

• Code to code comparison using a common cross section library – important for verification.

• Obtain a detailed understanding of the events and the processes.

• Benefit from different approaches, understanding limitations and approximations.

• Organize a special session at conference/special issue of publication (good exposure)

The technical topics to be presented in the final documentation of the benchmark exercise are shown below.

• The PMR benchmark definition

• Steady state test case definitions

• Transient test case definitions

• Cross section preparation case definitions

• Specific technical issues of the benchmark such as cross sections, correlations and formats of results

• Information of codes and methods used by participants

• Results and discussions of results

• Conclusions and recommendations

4. THE MHTGR-350 NUCLEAR POWER PLANT

The MHTGR 350 is a General Atomics (GA) design that has existed since the 1980's. The Nuclear Steam Supply (NSS) module arrangement is shown in Figure 1. The reactor vessel contains the reactor core, reflectors and associated neutron control systems, core support structures, and shutdown cooling heat exchanger and motor-driven circulator. The steam generator vessel houses a helically coiled steam generator bundle as well as the motor-driven main circulator [2]. The pressure-retaining components are constructed of steel and designed using existing technology.

The reactor vessel is uninsulated to provide for decay heat removal under loss-of-forced-circulation conditions. In such events, heat is transported to the passive Reactor Cavity Cooling System (RCCS), which circulates outside air by natural circulation within enclosed panels surrounding the reactor vessel. No valves, fans, or other active components or operator actions are needed to remove heat using the RCCS.

The reactor core and the surrounding graphite neutron reflectors are supported within a steel reactor vessel. The restraining structures within the reactor vessel are a steel and graphite core support structure at the bottom and a metallic core barrel around the periphery of the side reflectors.

Table 1: Major Design and Operating Characteristics of the MHTRG-350

MHTGR Characteristic	Value
Installed thermal capacity	350 MW(t)
Installed electric capacity	165 MW(e)
Core configuration	Annular
Fuel	Prismatic Hex-Block fuelled with Uranium Oxycarbide fuel compact of 15.5 wt% enriched U-235 (average)
Primary coolant	Helium
Primary coolant pressure	6.39 MPa
Moderator	Graphite
Core outlet temperature	687°C.
Core inlet temperature	259°C.
Mass Flow Rate	157.1 kg/s
Reactor Vessel Height	22 m
Reactor Vessel Outside Diameter	6.8 m

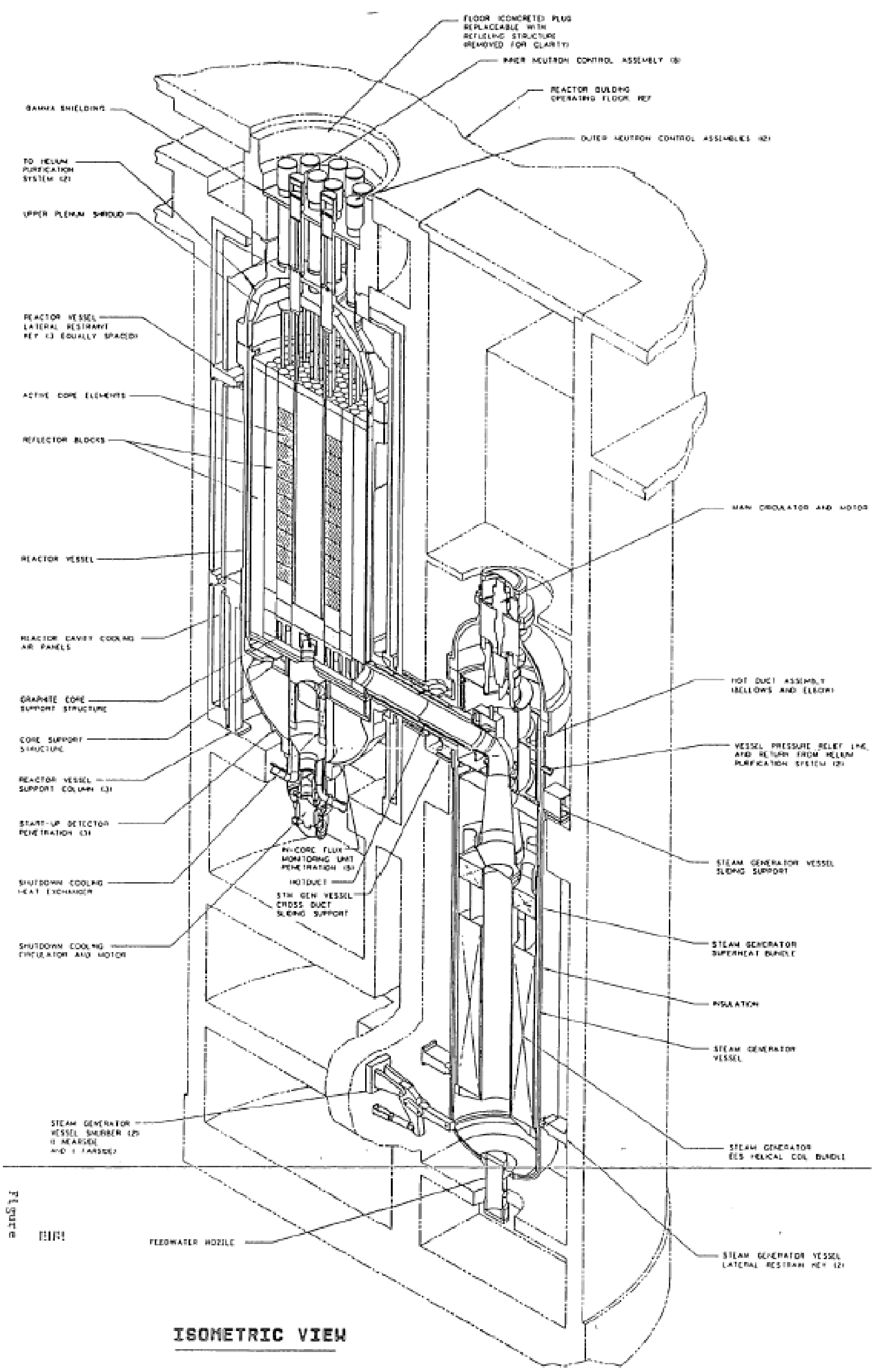

Figure 1: Layout of the MHTGR Reactor Module

5. THE MHTGR-350 REACTOR UNIT SPECIFICATION

5.1 THE REFERENCE CORE DESIGN DESCRIPTION

The design of the core consists of an array of hexagonal fuel elements in a cylindrical arrangement surrounded by a single ring of identically sized solid graphite replaceable reflector elements, followed by a region of permanent reflector elements all located within a reactor pressure vessel. The permanent reflector elements contain a 10 cm (3.94 in.) thick borated region at the outer boundary, adjacent to the core barrel. The borated region contains B_4C particles of the same design as in the FBP (see lower half of Table 5), but dispersed throughout the entire borated region with a volume fraction of 61%.

The core is designed to provide 350 MWt at a power density of 5.9 MW/m^3. A core elevation view is shown in Figure 2 and a plane view is shown in Figure 3. The active core consists of hexagonal graphite fuel elements containing blind holes for fuel compacts and full length channels for helium coolant flow. The fuel elements are stacked to form columns (10 fuel elements per column) that rest on support structures. The active core columns form a three row annulus with columns of hexagonal graphite reflector elements in the inner and outer regions. Thirty reflector columns contain channels for control rods. Twelve columns in the core also contain channels for reserve shutdown material.

The annular core configuration was selected, along with the power density of 5.9 MW/ m^3, to achieve maximum power rating and still permit passive core heat removal while maintaining the fuel temperature at ~ 1600°C (2912°F) during a conduction cooldown event. The active core effective outer diameter of 3.5 m is sized to maintain a minimum reflector thickness of 1 m within the 6.55 m inner diameter reactor vessel. The radial thickness of the active core annulus was specified on the basis of ensuring that the control rod worths of the reflector located rods would meet all shutdown and operating control worth requirements. The choice of reflector control rods was made to ensure that the control rod integrity is maintained during passive decay heat removal events. These radial dimensions also allow for a lateral restraint structure between the reflector and vessel. The height of the core with ten elements in each column is 7.9 m, which allows maximum power rating and axial power stability over the cycle.

The core reactivity is controlled by a combination of lumped burnable poison (LBP), movable poison, and a negative temperature coefficient. This fixed poison is in the form of lumped burnable poison compacts; the movable poison is in the form of metal clad control rods. Should the control rods become inoperable, a backup reserve shutdown control (RSC) is provided in the form of boronated pellets that may be released into channels in the active core.

The control rods are fabricated from natural boron in annular graphite compacts with metal cladding for structural support. The control rods are located in the outer ring of the inner reflector and the inner ring of the outer reflector (Figure 3). These control rods enter the reflector through the top reactor vessel penetrations in which the control rod drives are housed. The 24 control rods located in the outer reflector are the operating control rods, and are used for control during power operation, and for reactor trip. These operating rods can maintain the required 1% $\Delta\rho$ shutdown margin indefinitely under hot conditions, or for at least one day under cold conditions. Locating the operating rods in the outer reflector prevents damage during depressurized or pressurized passive heat removal. The six control rods in the inner reflector are the startup control rods, which are withdrawn before the reactor reaches criticality. With the startup and operating rods inserted, a 1% $\Delta\rho$ shutdown margin can be indefinitely maintained under cold conditions.

The RSC consists of boronated graphite pellets, housed in hoppers above the core. When the RSC is actuated, these pellets drop into channels in 12 columns of the active core. The RSC is used to institute reactor shutdown if the control rods become inoperable, or if necessary, to provide additional negative reactivity beyond that available in the inserted control rods.

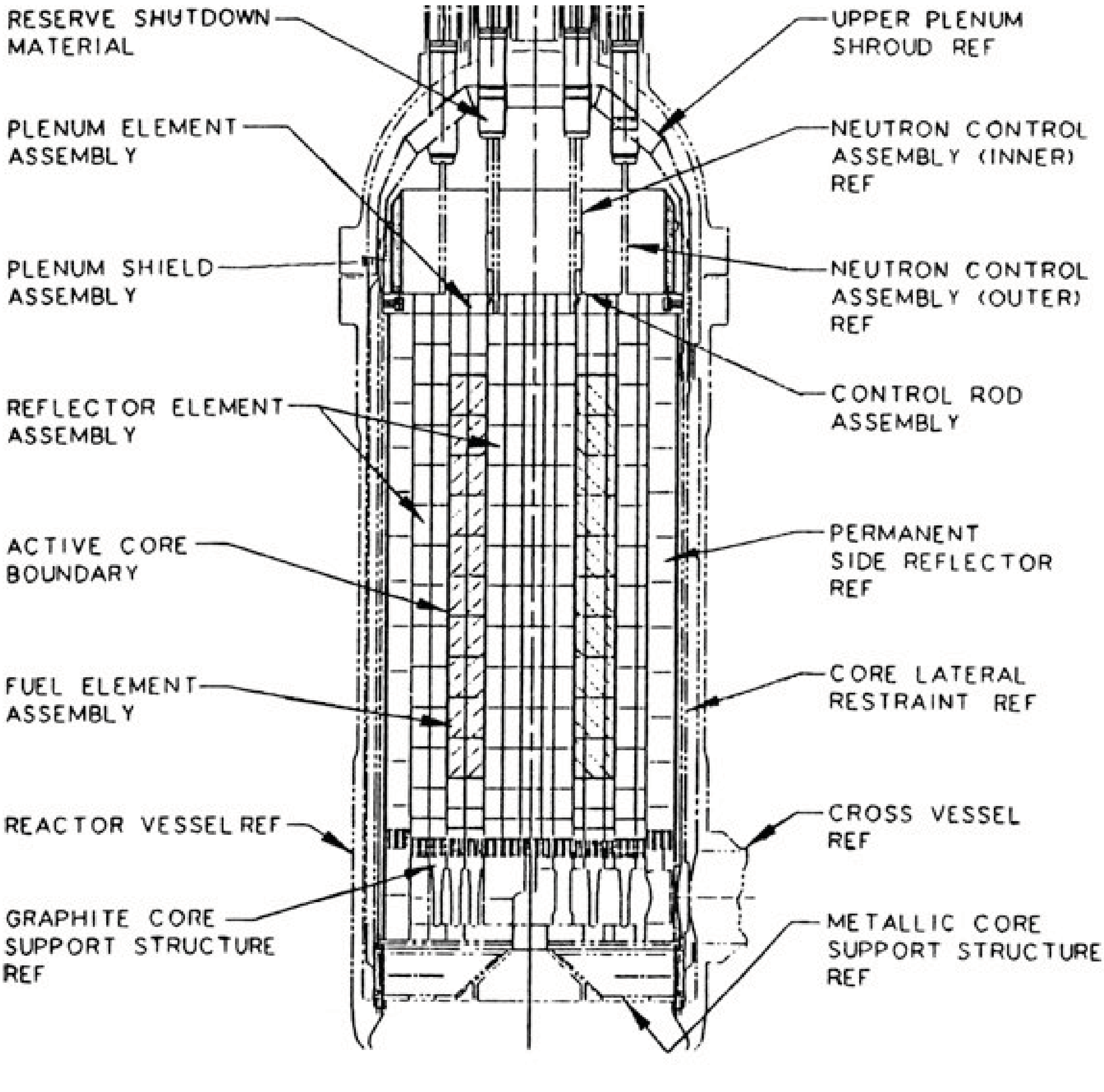

Figure 2: MHTGR Reactor Unit layout – Axial

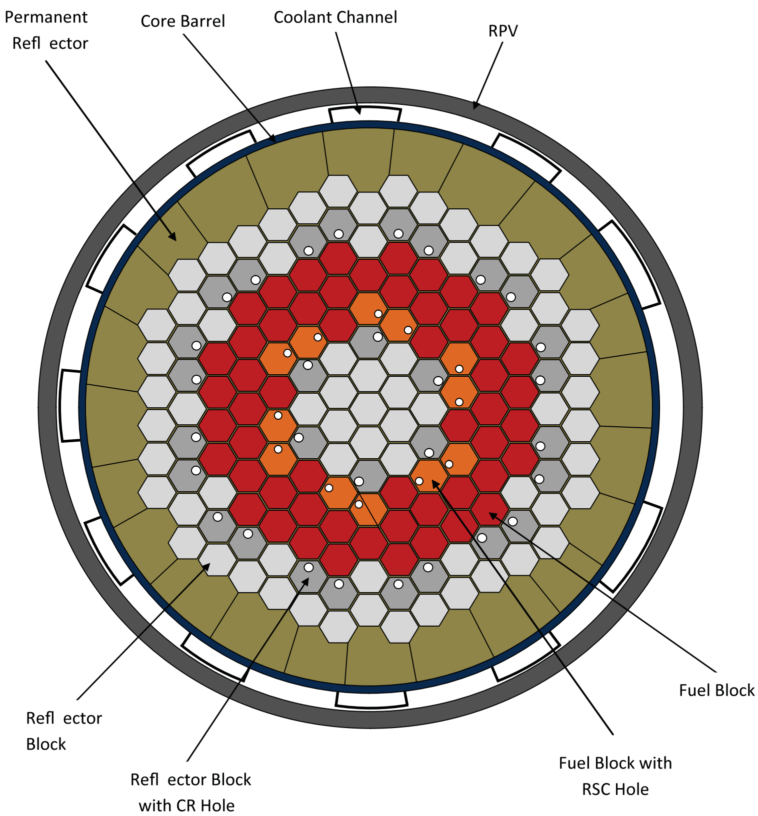

Figure 3: MHTGR Reactor Unit Layout – Plane

Table 2: Core Design Parameters

Core Parameter	**Value**	**Unit**
Thermal Power	350	MW(t)
Core power density	5.9	MW/m^3
Number of fuel columns	66	
Effective inner diameter of active core	1.65	m
Effective outer diameter of active core	3.5	m
Active core height	7.93	m
Number of fuel elements		
Standard elements	540	10/column
RSC elements	120	
Number of control rods		
Inner reflector	6	
Outer reflector	24	
Number of RSC channels in core	12	
Compacts per core	2025000	
Particles per core	1.21E+10	

5.1.1 Fuel Element Design

There are two types of fuel elements, a standard element, and a reserve shutdown that contains a channel for reserve shutdown control (RSC). The fuel elements are right hexagonal prisms of the same size and shape as the Fort St. Vrain HTGR elements. The fuel element components are shown in Figure 4 and Table 3.

The fuel and coolant holes are located in parallel through the length of the element. The standard fuel element contains a continuous array of fuel and coolant holes in a regular triangular array of two fuel holes per one coolant hole. The six corner holes contain lumped burnable poison compacts.

At each element-to-element interface in a column, there are four dowel/socket connections which provide alignment of coolant channels. A 3.5-cm diameter fuel handling hole, located at the center of the element, extends down about one-third of the height, with a ledge where the grapple of a fuel handling machine engages.

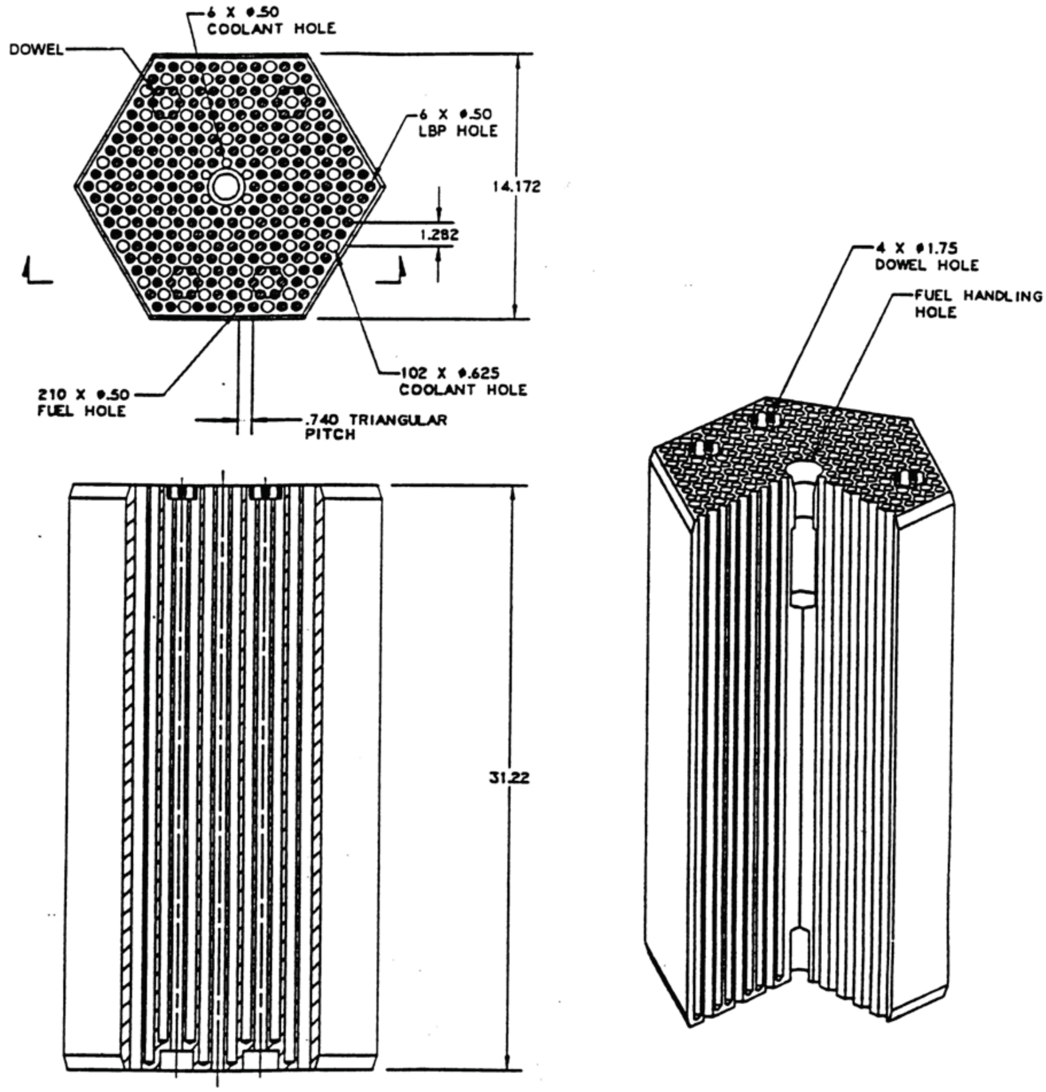

Figure 4: Hexagonal fuel element

Table 3: Fuel Element Description

Fuel Element Geometry	Value	Units
Block graphite density	1.85	g/cm^3
Fuel holes per element		
Standard element	210	
RSC element	186	
Fuel hole radius	0.635	cm
Compacts per hole	15	
Compacts per element		
Standard element	3150	
RSC element	2790	
Coolant holes per element (large/small)		
Standard element	102/6	
RSC element	88/7	
Large coolant hole radius	0.794	cm
Small coolant hole radius	0.635	cm
Fuel/coolant pitch	1.8796	cm
Block pitch	36	cm
Element length	79.3	cm
Fuel handling diameter	3.5	cm
Fuel handling length	26.4	cm
RSC hole diameter	9.525	cm
LBP holes per element	6	
LBP radius	0.5715	cm
LBP gap radius	0.635	cm
LBP length	72.14	cm

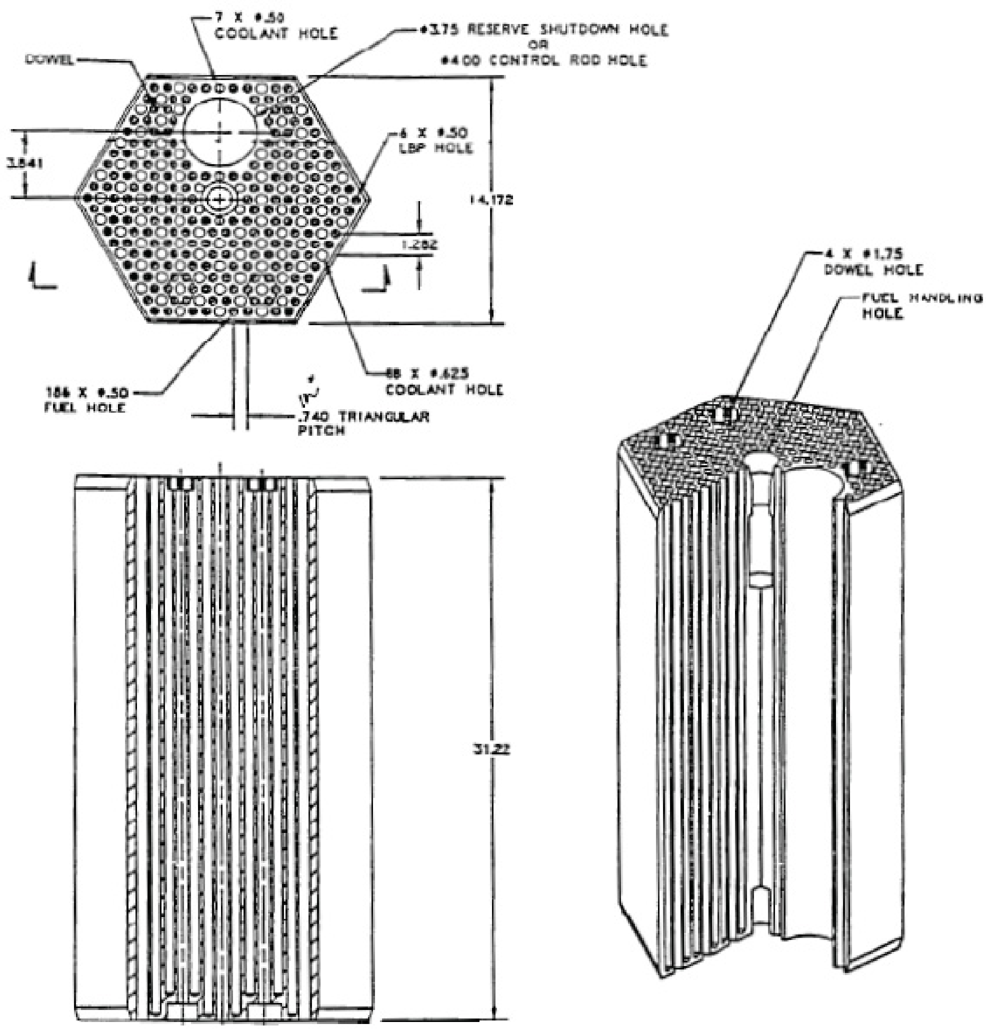

Figure 5: Hexagonal fuel element with RSC hole

5.1.2 Fuel particle and compact Design

The fuel is comprised of Tristructural-Isotropic (TRISO) fuel particles bonded in a cylindrical graphite matrix to form a compact. The compacts are then inserted into hexagonal graphite blocks to construct a fuel element. TRISO particles consist of various layers acting in concert to provide a containment structure that prevents radioactive product release. They include a fuel kernel, porous carbon layer, inner pyrolitic carbon (IPyC), SiC, and outer pyrolitic carbon (OPyC). The buffer layer allows for limited kernel migration and provides some retention of gas particles. The silicon carbide layer ensures the structural integrity of the particle under constant pressure and also helps retain metallic fission products. Details of the TRISO particle and compact designs are given Table 4.

Table 4: TRISO/Fuel Compact Description

TRISO Fuel Element	Value	Unit
Fissile material	$UC_{0.5}O_{1.5}$	
Enrichment (U-235 average)	15.5	wt%
Radii (cm)		
Kernel	0.02125	cm
Buffer	0.03125	cm
IPyC	0.03475	cm
SiC	0.03825	cm
OPyC	0.04225	cm
Densities (g/cm^3)		
Kernel	10.5	g/cm^3
Buffer	1	g/cm^3
IPyC	1.9	g/cm^3
SiC	3.2	g/cm^3
OPyC	1.9	g/cm^3
Packing Fraction (average)	0.350	
Number of Particles/compact (average)	5986 ??	
Compact Radius	0.6223	cm
Compact Gap Radius	0.635	cm
Compact Length	4.928	cm

5.1.3 Lumped Burnable Poison Design

The LBP consists of boron carbide (B_4C) granules dispersed in graphite compacts. The B_4C granules are pyrocarbon (PyC) coated to limit oxidation and loss from the system. The amount of burnable poison is determined by reactivity control requirements, which may vary with each reload cycle. The diameters of the FBP rods are specified according to requirements for self-shielding of the absorber material to control its burnout rate relative to the fissile fuel burnout rate. The goals are to achieve near complete burnout of the material when the element is replaced, and to minimize the hot excess reactivity swing over the cycle. The current design uses six LBP rods per element in all core layers, while axial zoning is performed through having relatively less LBP mass in the top and bottom layers compared to the middle layers of the core. Axial LBP zoning will be used to maintain the axial power shape during burnup and to prevent xenon induced axial power oscillations. The current design also uses a constant FBP compact diameter of 1.143 cm for all cycles. Details of the FBP design are given in Table 5, assuming that each FBP rod contains 14 compacts.

Table 5: Lumped Burnable Poison Description

LBP holes per element	6			
LBP compacts per LBP rod	14			
Compact diameter [cm]	1.143			
Compact length [cm]	5.156			
Rod length [cm]	72.187			
Volume fraction of B_4C particles				
FBP Component	Composition	Diameter [μm]	Thickness [μm]	Density [g/cm^3]
B_4C Particle				
Kernel	B_4C	200	-	2.47
Buffer coating	Graphite	-	18	1.0
Pyrolitic coating	Graphite	-	23	1.87
Kernel	B_4C	200	-	2.47
Matrix	Graphite	-	-	0.94

5.1.4 Replaceable Reflector Design

The replaceable reflector elements are graphite blocks of the same shape, size, and material as the fuel elements. The top and bottom reflector elements contain coolant holes to match those in the active core. All reflector elements have dowel connection for alignment.

The reflector above the active core is composed of two layers: one layer of full-height elements above a layer of half-height elements, for total reflector height of 1.2 m. The top reflector elements channel coolant flow to the active core and provide for the insertion of reserve shutdown material into the active core. They have the same array of coolant holes as the fuel element and the same holes for the insertion of reactivity control devices.

The reflector below the active core has a total height of 1.6 m. It consists of two layers: one layer of two half-height reflector elements above a layer of two half-height flow distribution and support elements. The bottom two elements provide for the passage of coolant from the active core into the core support area. This is accomplished by directing the coolant channel flow to the outside of the core support pedestal. The channels for the control rods and reserve shutdown material (RSS) stop at the top of the lower reflector so that neither the rods nor the RSS material can exit the core at the bottom. However, small holes are drilled through the reflector below the control rod channels so that adequate cooling is provided for the rods when they are inserted in the core or side reflectors without excessive coolant flow through these channels when the rods are withdrawn from the core.

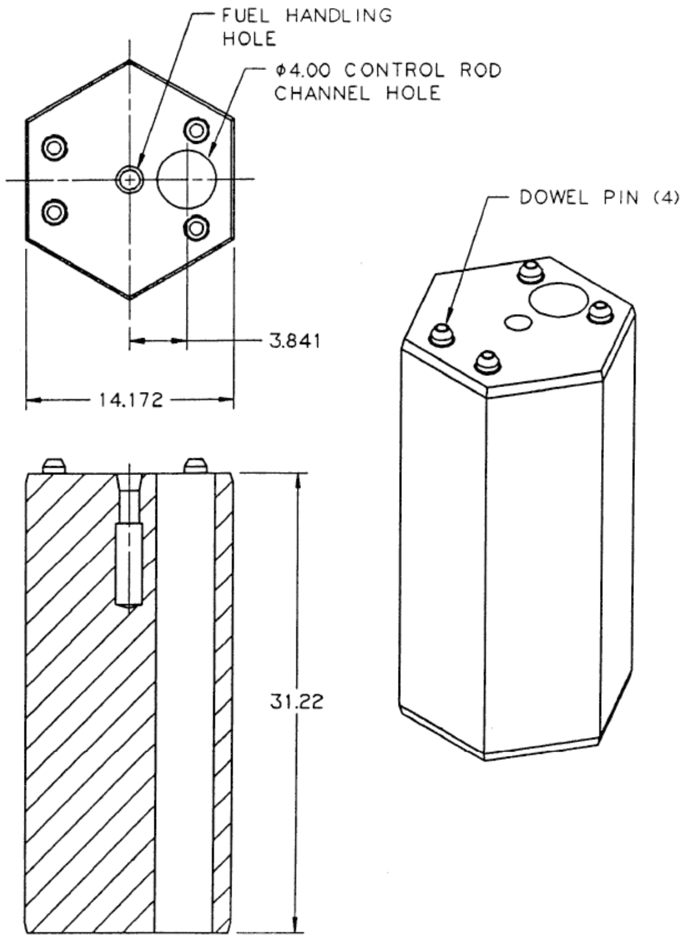

Figure 6: Hexagonal Reflector element with CR hole

The outer side reflector includes one full row and a partial second row of hexagonal reflector columns. The outer row of hexagonal elements is solid, with the exception of the handling holes. Twenty-four of the elements in the inner row of the outer side reflector also have a control rod channel as shown in Figure 3. The control rod channel has a diameter of 10.2 cm and stops at an elevation just below the active core. Crushable graphite matrix at the lower end of each control rod channel will limit the load between the control rod assembly and reflector element in the event that the neutron control assembly support fails. The control rod channel is centered on the flat nearest the active core 10.2 cm from the center of the reflector element. The distance from the flat of the reflector block to the edge of the control rod channel is 2.7 cm.

The inner (central) reflector includes 19 columns of hexagonal elements. The central and side reflector columns consist of, from top down, one three-quarter height element, eleven full-height elements, one three-quarter height element, and two half-height elements, above the core support pedestal. The total reflector height for the equivalent 13.5 elements above the top of the core support pedestal is 10.7 m. The dowel/socket connection at each axial element-to-element interface provides alignment for refuelling and control rod channels, and transfers seismic loads from reflector elements. There are six control blocks in the inner reflector.

5.1.5 Control Rods and Reserve Shutdown Control

The control rod design use din the MHTGR is shown in Figure 7. The neutron absorber material consists of B_4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The boron is enriched to 90 weight percent B-10 and the compacts contain 40 weight percent B_4C. The compacts have an inner diameter of 52.8 mm, an outer diameter of 82.6 mm, and are enclosed in Incoloy 800H canisters for structural support. Alternatively, carbon-fiber reinforced carbon (CC) composite canisters, or SiC, may be used for structural support. The control rod consists of a string of 18 canisters with sufficient mechanical flexibility to accommodate any postulated offset between elements, even during a seismic event.

The reserve shutdown control material consists of 40 weight percent natural boron in B_4C granules dispersed in a graphite matrix and formed into pellets. The B_4C granules are coated with PyC to limit oxidation and loss from the system during high temperature, high moisture events. When released into the reserve shutdown channel in the fuel element, the pellets have a PF of ≥ 0.55.

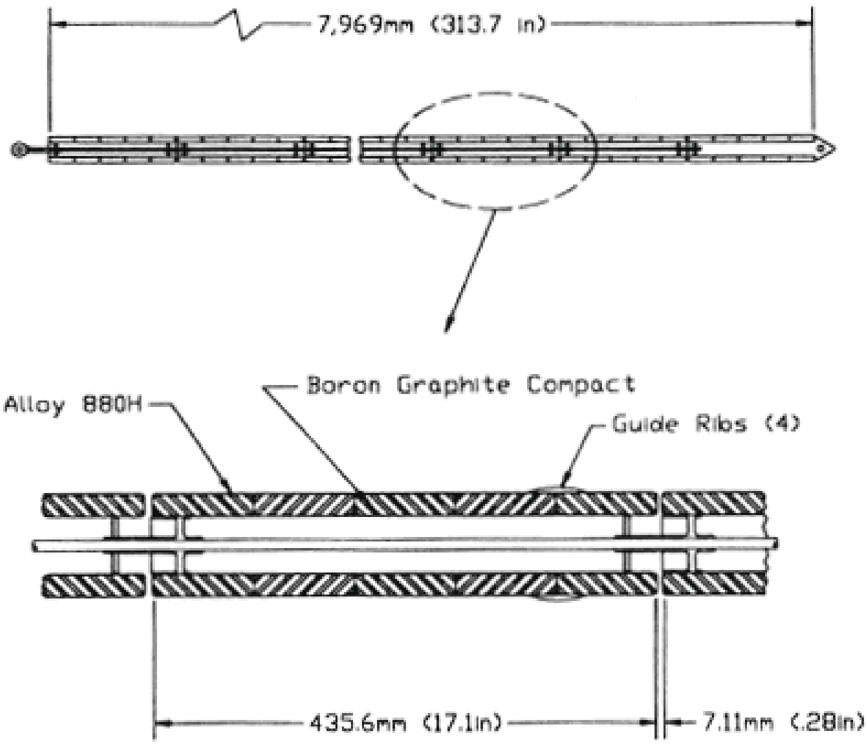

Figure 7: Control Rod Design

The control rods are withdrawn in groups with three control rods in each group. The three control rods in each group are symmetrically located around the core, so that one rod is located in each 120° sector of the core. During normal power operation, control is accomplished with only the operating control rods (the startup control rods are in the fully withdrawn position.) These rods are operated automatically on the demand signal from the Plant Control Data Instrumentation System (PCDIS) in symmetric groups, with three control rods per group. The neutron flux level is continuously monitored by the ex-vessel detectors that supply signals to the PCDIS, the Investment Protection System (IPS) and the Reactor Protection System (RPS).

5.1.6 Permanent Reflector Design

There are 60 permanent side reflector columns (Figure 3). The current design uses 60 cold helium return holes equally spaced around the reactor, each 20.32 cm in diameter. The volume fraction occupied by these holes (in the non-borated region of the permanent reflector) is 0.29.

Neutron shielding of the reactor structural equipment consists of graphite permanent reflector elements containing a 10 cm thick borated region at the outer boundary, adjacent to the core barrel. The borated region contains B_4C particles of the same design as the LBP. As opposed to containing the particles in compacts, the current design assumes B4C particles are dispersed throughout the entire borated region, and the volume fraction the particles occupy within the borated region is 0.61.

5.2 SIMPLIFICATIONS INTRODUCED IN THE BENCHMARK MODEL

5.2.1 Neutronic Simplifications

The following simplifications are assumed for the neutronic definition:

- The core is 1/3 symmetric as far as cross section specification. Asymmetries in the thermal-fluids might make the full core coupled problem asymmetric, depending on the modelling.
- Neutron streaming in the gaps, coolant holes, and control holes is ignored.
- Element bowing due to temperature gradients is ignored.
- Axial dimensions of the fuel rod are simplified: the length of the fuel rods and FBP are assumed to be the full height of the block, the fuel handling holes are replaced with lower density graphite, the axial details of the control rods are ignored.
- The borated region by the core barrel is not modelled.

5.2.2 Thermal Fluids Simplifications

The following simplifications are assumed for the thermal fluids definition:

- The complexity of the flow path in the upper plenum is simplified with the use of a flow area and equivalent hydraulic diameter.
- The effect of excluding specific coolant flows is to some extend balanced by the assumption that all heat sources (from fission) will be deposited locally, i.e. in the fuel and that no other heat sources exist outside the core (for example neutron absorption in the control rods). Simplifications are also made in the material thermal properties in as far as constant values are employed or specific correlations are employed. These assumptions are clearly listed in the sections that follow.
- No cross flow is modelled.

6. CORE LAYOUT

6.1 REACTOR AND CORE STRUCTURE GEOMETRY AND DIMENSIONS

The benchmark reactor unit geometry definition is given in this section. Figure 8 and Figure 9 show the general layout of the reactor. The dimensions of the key components are included in Figure 10 and Figure 11.

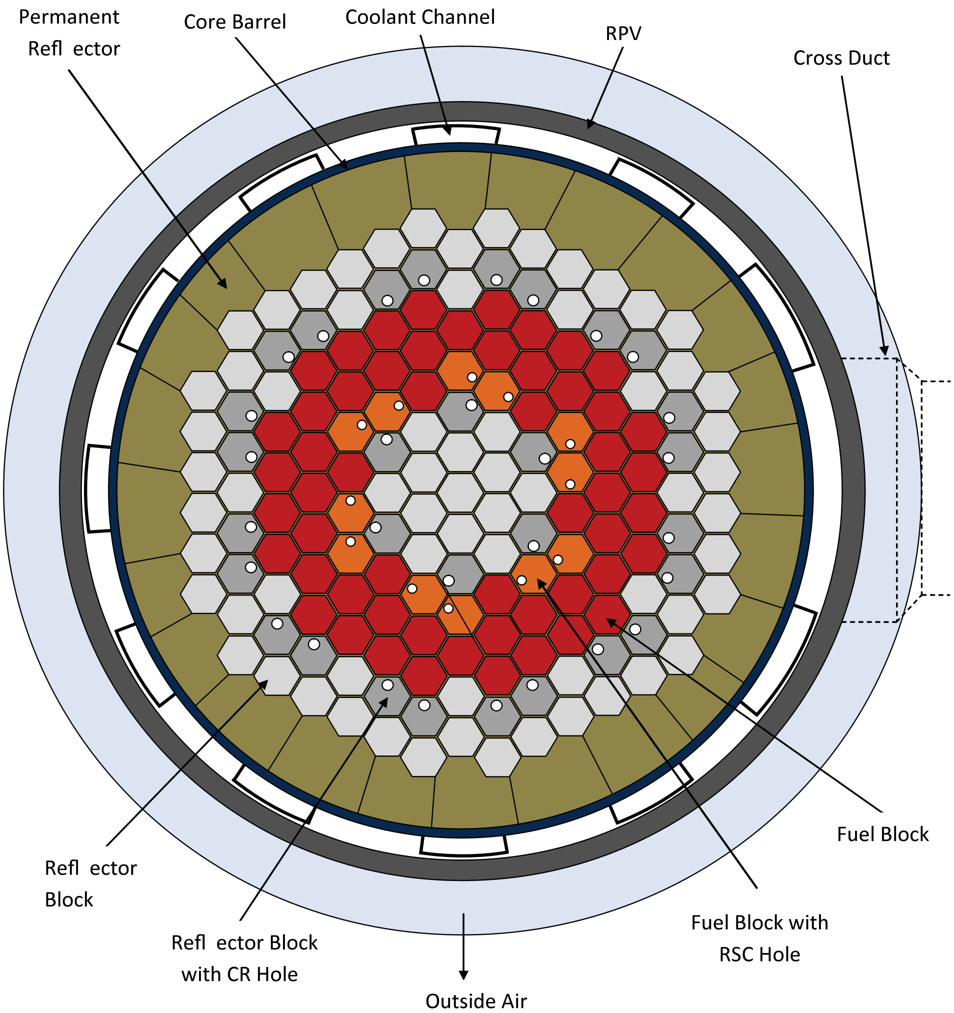

Figure 8: Core Radial Layout

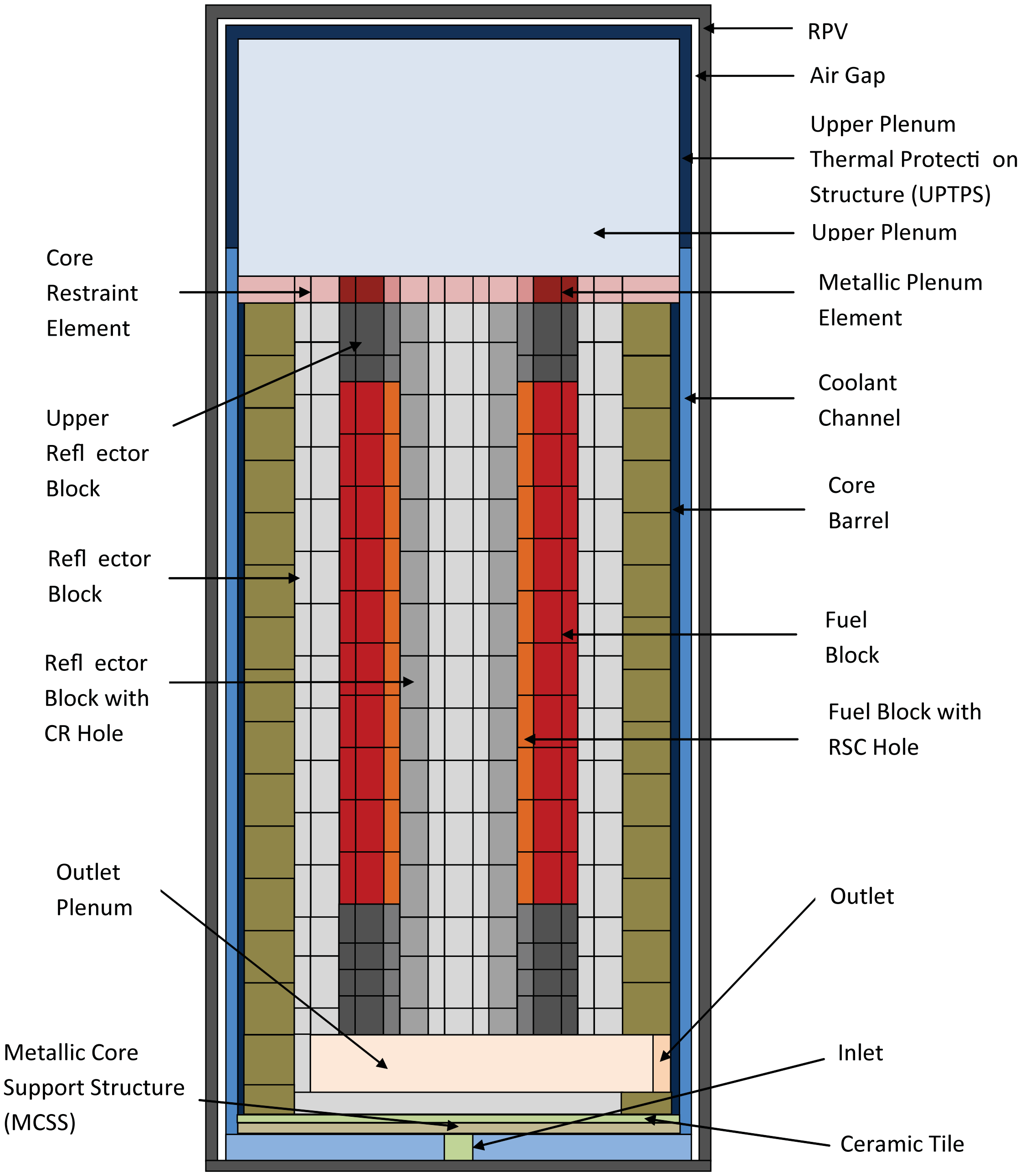

Figure 9: Core Axial Layout

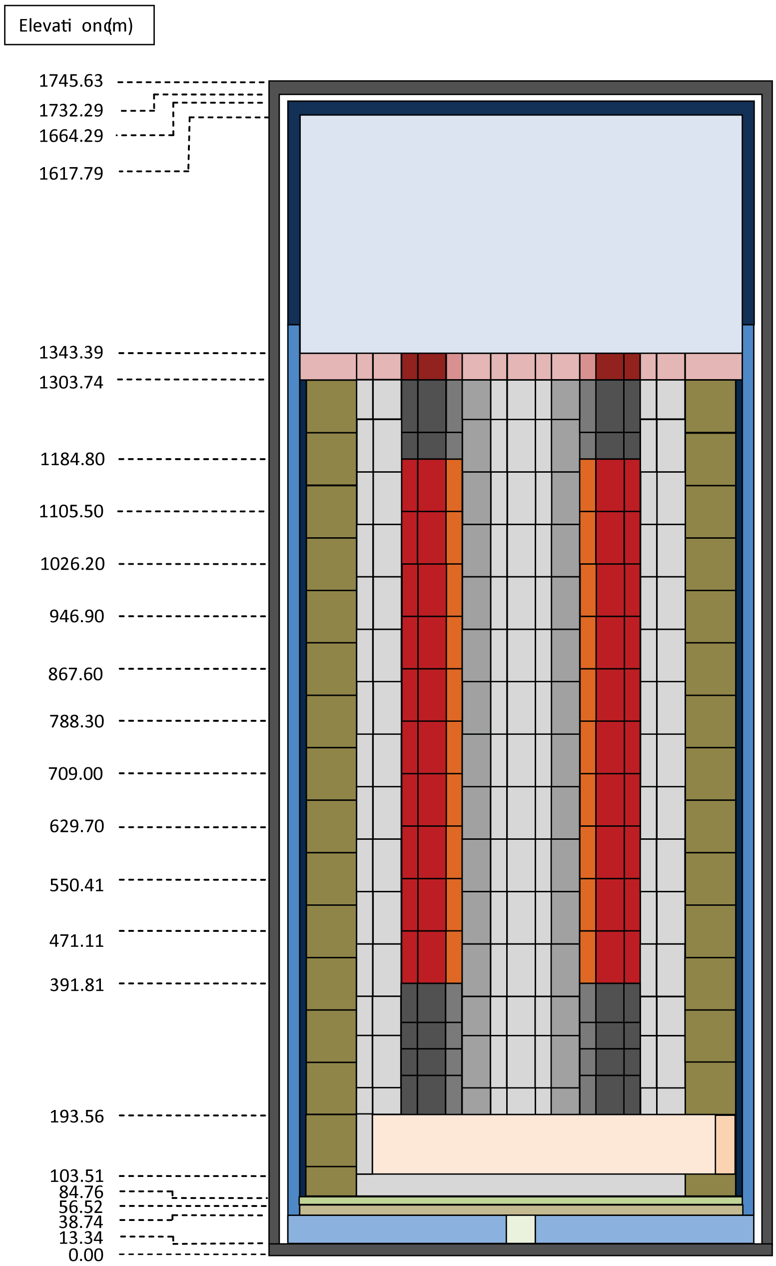

Figure 10: Core Axial Layout (dimensions)

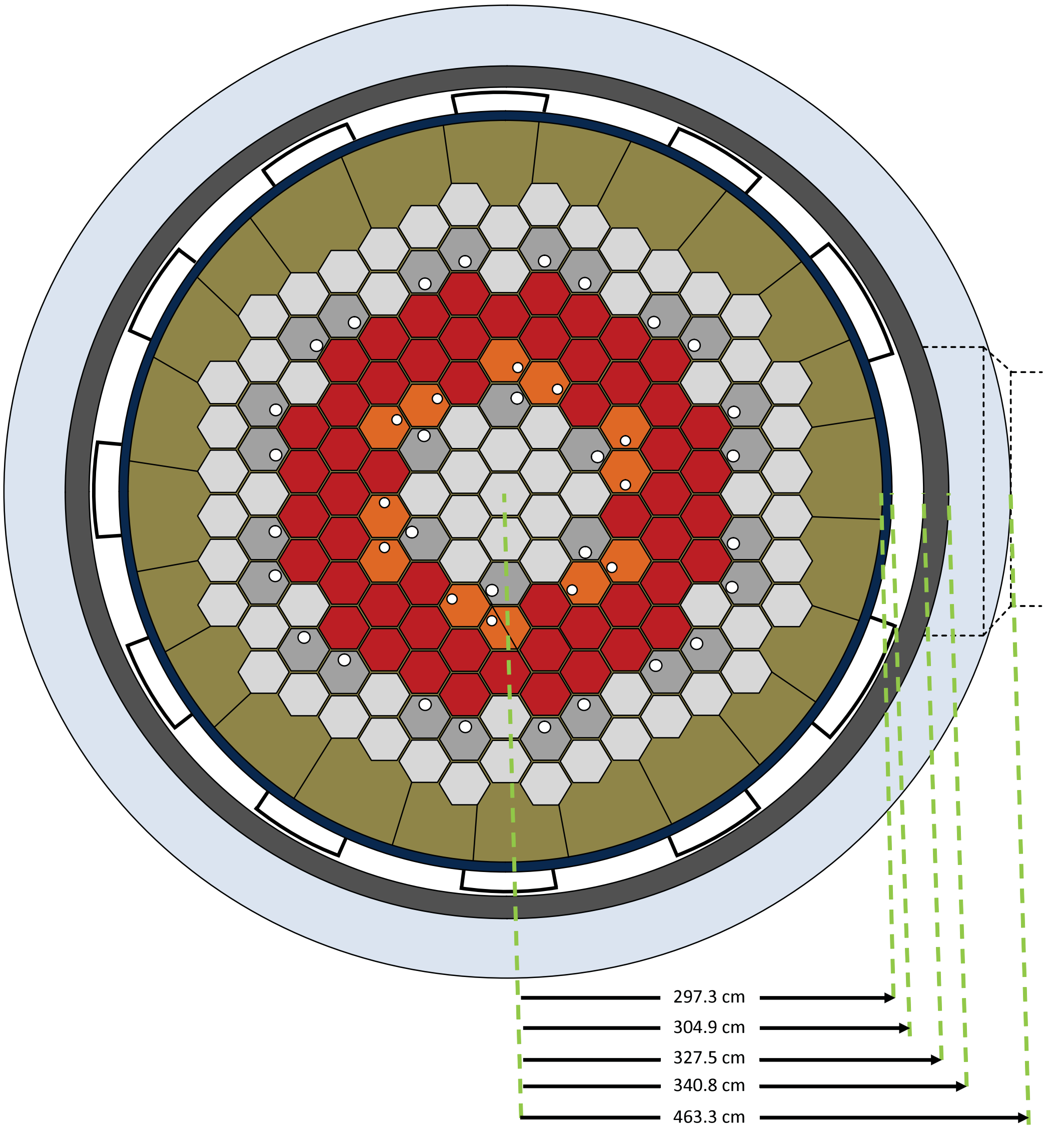

Figure 11: Core Planar Layout (dimensions)

6.2 NEUTRONIC DEFINITION

The neutronic solution of the benchmark problem is only required on a geometrical subset or a smaller part of the reactor. All neutronically important regions are included but regions far from the core, where flux solutions may be problematic, were excluded.

The axial neutronic mesh extends from the top reflector and core restraint element interface to the bottom reflector (just above the inlet plenum). Radially the core barrel forms the outer boundary, where the borated region before the barrel creates this neutronic boundary.

The material numbering will be organized in layers and columns. Figure 12 shows the numbering for the 1/3rd core. Note that for this Figure each block is subdivided into 7 sub-hex domains, but this is just for illustration purposes. All of the data and reporting requirements for the benchmark are on a block basis. The bottom reflector is defined as layer 0, whereas the upper reflector is layer 11. Radially the central column is column 0, the rest of the numbering follows the various radial rings as shown in Figure 12.

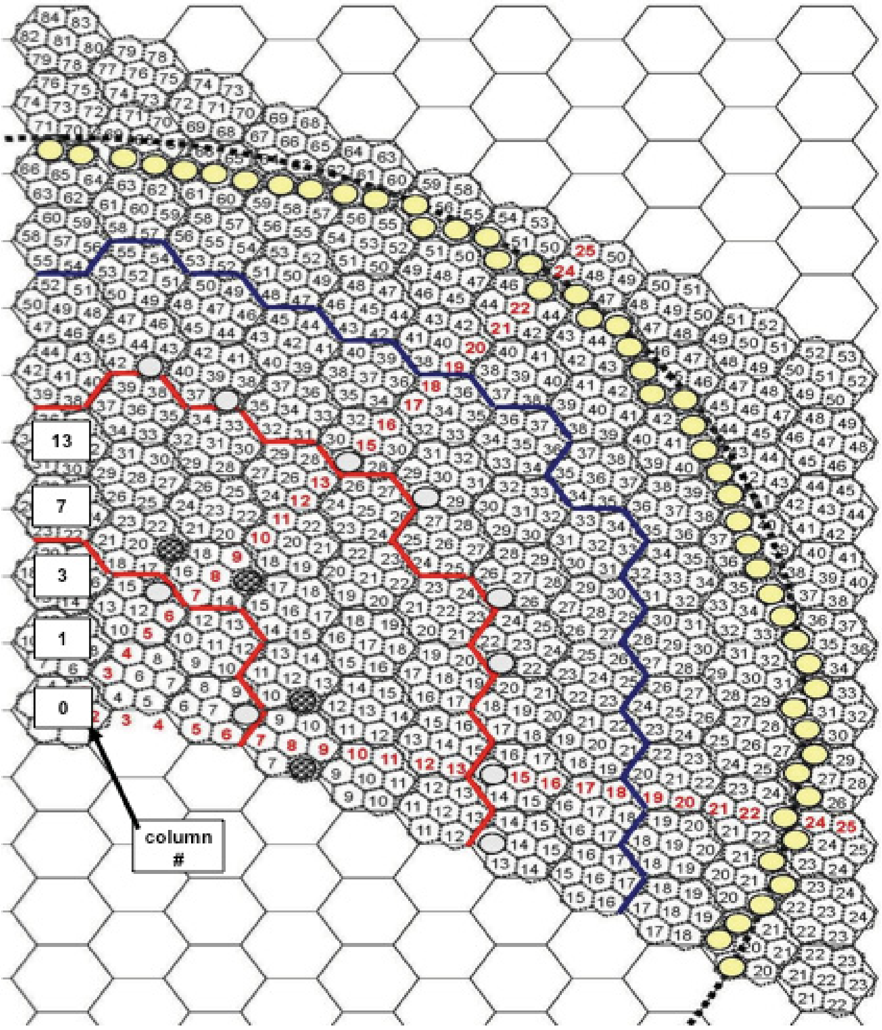

Figure 12: Core Position Numbering Layout

The active core region includes layers 1 -10 and columns 7-28. The fist layer is located at the bottom of the active core near the bottom reflector. The top layer, 10, is near the top reflector. Therefore, the active core includes 22 material regions per layer, as shown in Figure 13, with a total of 220 material regions. Note that there is only one control rod group inserted in this EOEC core. The insertion depth is 1 block height, so the CR is only inserted 1 layer into the active core from the top reflector.

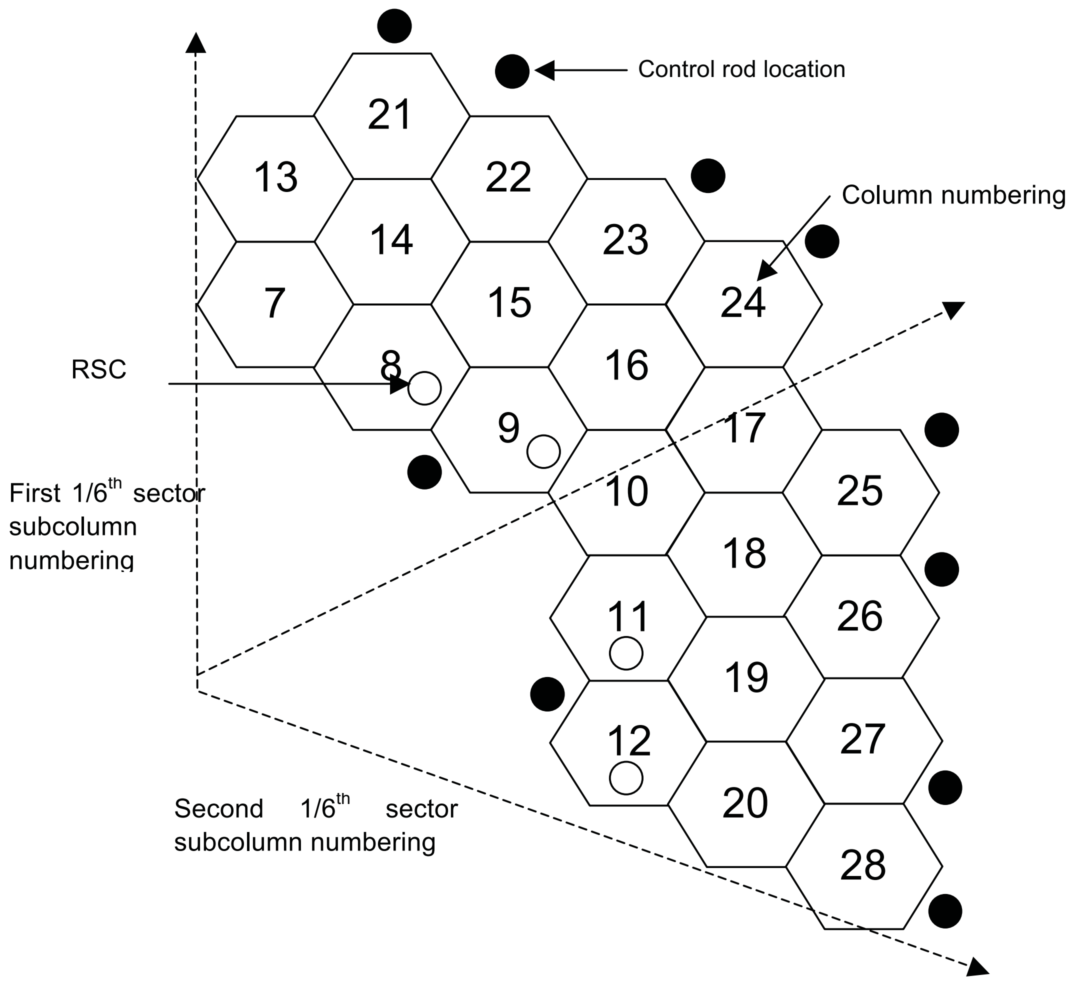

Figure 13: Active Core Composition Layout

6.3 NEUTRONIC BOUNDARY CONDITIONS

Table 6: Neutronic Boundary Conditions

	Description	Position [cm]	B.C. Type
1	Outer boundary (core barrel)	297.3	Non-re-entrant current
2	Above upper reflector	1303.74	Non-re-entrant current
3	Below lower reflector	193.56	Non-re-entrant current
4	Core segment sides	(1/3 core segment)	Periodic

7. DESCRIPTION OF THE CROSS SECTION TABLES

This section outlines the format and origin of the cross section tables, which were generated for the MHTGR benchmark. The exposure and burnup history for the equilibrium cycle is taken into account implicitly into the cross-section libraries by defining the different fuel mixtures. The average isotopic composition of the different regions of the core was determined as discussed in section 7.1.

Two sets of cross sections are provided. The first set is referred to as the simplified set since the macroscopic cross sections provided is constant and thus contains no dependence on changing core conditions or state parameters. This set is therefore only useful to test the implementation of the cross section interpolation routines into the codes.

The second set contains cross sections as a function of a number of state parameters and is therefore used for the actual calculations (steady state starting condition and transient analysis – see section 7.3).

7.1 NUMBER DENSITIES USED TO GENERATE CROSS SECTIONS

The number densities were generated by the DIFF-3D code. In accordance with the specifications, the core region was represented. Therefore, there are 220 number density sets for the fuel regions. Additionally, the number densities for the central column / graphite reflector, control rods (homogeneous absorber material) and Core Barrel are also supplied.

7.2 SIMPLIFIED CROSS SECTIONS SETS

The purpose of the simplified cross section set is to provide a reference that can be used to test and compare stand-alone neutronic predictions by using the same cross sections with no thermal-hydraulic feedback. This will help to understand and quantify the differences introduced by the different neutronics models used and assist in the process to narrow them down. The set is generated from DRAGON 4 lattice physics calculations and based on the DIFF3D end of equilibrium core (EOEC) number densities. The calculation was performed in twenty six energy groups. An example of the simplified cross section set is included in Appendix III.

MATERIAL #

Group #	Scalar Flux	Σ_T	D	$\nu\Sigma_f$	χ

Scattering Profile (g = 1 – ngroup columns)

Start Group ID End Group ID

P0 Scattering Matrix (g = 1 – ngroup columns)

$\Sigma_{g\text{->}g'}$ (g' = Starting Group ID, End Group ID

/MATERIAL #

7.3 INITIAL STEADY STATE AND TRANSIENT CROSS SECTION TABLES

Three-dimensional tables are used to represent the instantaneous variation in cross-section due to changes in the reactor. The cross section models are designed to cover the initial steady state conditions and the expected ranges of change of the 3 selected instantaneous feedback parameters in the transients to be simulated in the benchmark. The set is generated from the DRAGON 4 code using the end of equilibrium core number densities with 26 energy groups.

Cross sections were generated for all the combinations of the given state parameters. The four state parameters are:

- Fuel temperature
- Moderator temperature
- Xenon concentration

In all of the fuel material cross section tables, there were four fuel temperatures, seven moderator temperatures, and three Xenon number densities while for all the non-fuel materials no fuel temperature or xenon variations were included.

The cross section file name: "OECD-MHTGR350.XS", an ASCII data file should be used for all exercises except for the moisture ingress. A separate cross section file: "OECD-MHTGR350_H2O.XS", should we use for the moisture ingress analysis.

7.3.1 State parameters

The ranges chosen for each parameter were selected based on the reactor conditions for normal operation as well as for accident conditions. The following values for the 3 state parameters were selected:

Fuel temperature (Doppler temperature): 300K, 800K, 1400K, 2400K

Moderator temperature: 300K, 600K, 800K, 1100K, 1400K, 1800K, 2400K

Xenon concentrations expressed as homogenised concentrations: 0.0 (or very small 1.0E-15), 2.0E-11, 8.0E-10 [#/barn.cm]

Primary system water inventory: 0, 600, 1200, 1800 kg.

7.3.2 Layout of Data in Cross Section Tables

- homogenized flux
- total macro x-sec
- isotropic diffusion coefficient
- nu sigma fission macro x-sec
- sigma fission macro x-sec
- prompt and delayed fission spectrum
- scattering matrix (provided with a scattering matrix profile from g->g' for each gth row)
- Inverse neutron speed
- Beta 1-6 (delayed neutron fraction)

- Kappa (energy released per fission)
- Microscopic absorption for Xe
- I and Xe yields

MATERIAL #

Group #	Scalar Flux	Σ_T	D	$\nu\Sigma_f$	χ

Scattering Profile (g = 1 – ngroup columns)

Start Group ID   End Group ID

P0 Scattering Matrix (g = 1 – ngroup columns)

$\Sigma_{g\text{->}g'}$ (g' = Starting Group ID, End Group ID

/MATERIAL #

7.3.3 Interpolation an Extrapolation

TBD

7.3.4 Additional notes on the "OECD-MHTGR350.XS" cross section tables

- All macroscopic cross sections in units cm-1
- The supplied cross sections are macroscopic cross sections, except for Xenon absorption cross sections which are the microscopic absorption cross sections (barns).
- The first records of the data in each table are the points (or values) of all the state parameters specified for the material. The order is thus very important. The first four values are fuel temperatures, the next seven values are moderator temperatures, and the last three are the Xenon number densities.
- The total macroscopic absorption cross-sections provided exclude the absorption effect of xenon that should be treated explicitly by the participants' codes. During the cross section generation process the xenon absorption was subtracted from the macroscopic cross section using the xenon microscopic cross sections and the input xenon number densities.
- The diffusion coefficients were calculated from the transport cross sections in the DRAGON 4 code.

- The B (delayed neutron fraction) and κ (energy release per fission) values are given per material per energy group.
- The delayed neutron λ values are given in Table 7, are not included in the cross section library, and should be treated as user input. Note that a single set of decay constants was derived using the U_{235} data as the basis.

Table 7: Delayed Neutron Decay Constants

Group	λ [sec^{-1}]
1	3.870E+00
2	1.400E+00
3	3.110E-01
4	1.160E-01
5	3.174E-02
6	1.272E-02

8. THERMO-FLUIDS DATA

The thermo-fluids model of the benchmark is designed to preserve all the characteristics and phenomena related to the heat transfer and fluid flow in the MHTGR-350 design. In order to reduce the complications in the computational modelling of the benchmark core, some minor simplifications were made to the geometry.

In the original MHTGR design, the helium coolant enters the core from cross duct, flows down to the bottom of the metallic core support structure (MCSS) with making a u-turn cools down the MCSS. To reduce the complexity of the flow at the entrance the flow inlet is placed to the center bottom. The coolant flows towards the outer direction cooling down the MCSS and keeps the same flow path as the original design.

Another simplification is done at the upper plenum region by removing the structure between the core restraint elements and the upper plenum. The helium from the coolant channels flows into the upper plenum and then is directed to the fuel elements.

8.1 REACTOR THERMAL-FLUIDS LAYOUT

The radial and axial layouts of the core are shown in **Error! Reference source not found.** and Figure 9. The radial geometry for the thermo fluids model extends to the stagnant air environment outside the Reactor Pressure Vessel (RPV). The geometry in the axial direction extends to the outer surface of the RPV.

8.2 REACTOR MAIN COOLANT FLOW SPECIFICATIONS

The helium coolant enters the core from the bottom, cools down the bottom plate and flows up in the 11 coolant channels located between the core barrel and RPV. When it reaches the upper plenum, it is directed down through the core coolant channel to the outlet plenum and leaves the reactor. The main coolant flow map is shown in Figure 14.

Table 8: Main flow parameters

#	Description	Unit	Value
1	He inlet temperature	ºC	259
2	He outlet temperature	ºC	~687
3	Total inlet mass flow rate	kg/s	157.1
4	Inlet Pressure	kPa	6390

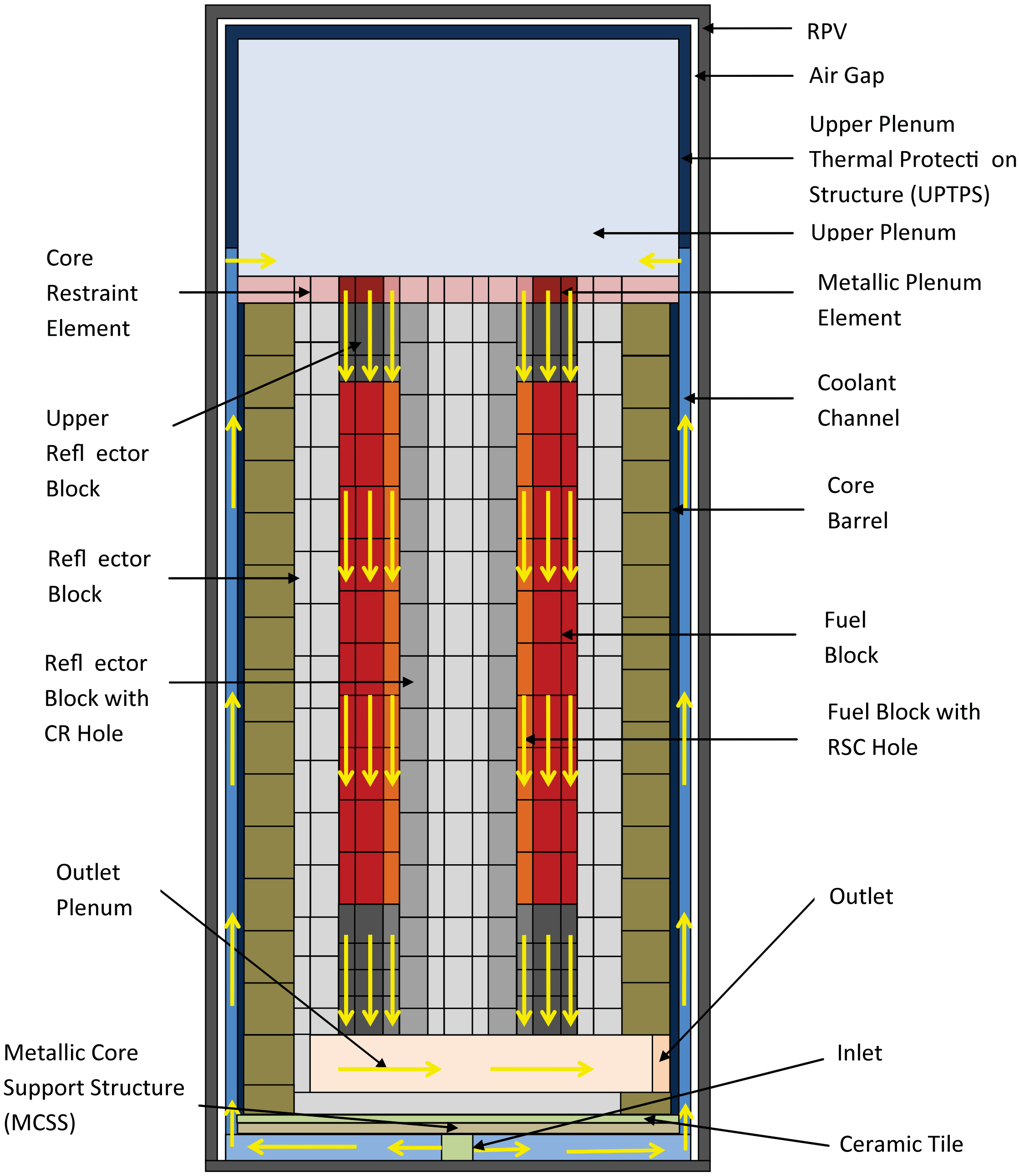

Figure 14: Main Core Flow Path

8.2.1 BYPASS FLOW SPECIFICATION

The bypass flows considered in this benchmark are divided into four categories. These are in-core gaps, ex-core gaps, reflector coolant channels and control rod channels. Figure 15 shows the bypass flow paths.

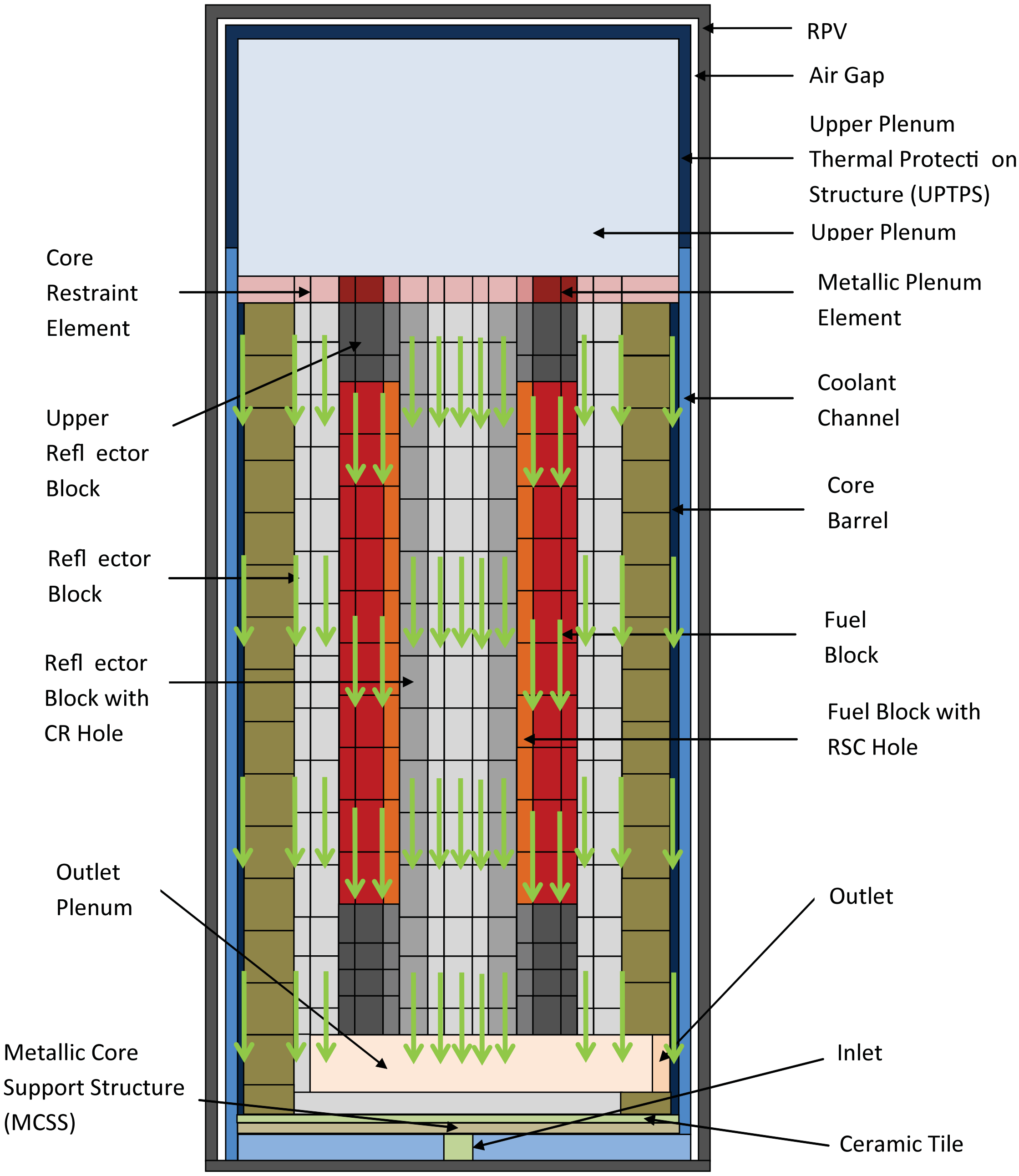

Figure 15: Core Bypass Flow Paths

8.2.1.1 TYPE-I

The bypass flow is specified and azimuthally uniform for each radial ring of the core.

Table 9: Bypass flow distribution

	Component	%
1	Incore	1.5
2	Inner Reflector	0.5
3	Control Rod Cooling (Inner)	1.2
4	Control Rod Cooling(Outer)	1.8
5	Outer Reflector (First Ring)	1.38
6	Outer Reflector (Second Ring)	1.62
7	Permanent Side Reflector	3.0
	Total	11%

8.2.1.2 TYPE-II

Bypass flow path specifications are given. The bypass flow is calculated implicitly.

Table 10: Bypass flow gap sizes

	Flow Path	**Width (mm)**
1	Gaps between Fuel Blocks	2.0
2	Gaps between Permanent Side Reflector and Core Barrel	3.5
3	Control Rod Hole Diameter	102.0

8.3 MATERIAL PROPERTIES

ALL MATERIAL PROPERTIES ARE PENDING VERIFICATION FROM GA

The radial effective conductivity of the fuel block can be determined from the AMEC model included in APPENDIX II and the data from the various constituents in this section. The axial effective thermal conductivity will be taken as the volume ratio of the solid region multiplied by the graphite conductivity. Temperature and fluence dependent thermo-physical properties are provided where relevant. The file “OECD-MHTGR350-fluence.inp” includes the fluence distribution in all regions of the core including reflector regions. This file is included in APPENDIX IV.

8.3.1 Grade H-451 Graphite

Grade H-451 graphite is a near-isotopic, petroleum-coke-based, artificial graphite developed specifically for HTGR fuel element and reflector application. Grade H-451 graphite is used in standard fuel, RSC fuel, and reflector blocks. The geometric descriptions of the fuel block are given in Figure 4, Figure 5, and Figure 6.

Temperature and fluence dependent conductivities are included in Table 11. Conductivities for fluence levels that lie between the points provided can be linearly interpolated. Any value outside the range of validity [500K, 1800K] and [0,8 $x10^{25}$ n/m^2] will retain a constant value based on the closest value in the range of validity.

Other H-451 graphite thermal properties are shown in Table 12.

Table 11: Thermal Conductivity of Grade H-451 Graphite

Fluence [$x10^{25}$ n/m^2]	Value* [W/m/K]
Un-irradiated	$k=3.28248E-05T^2 - 1.24890E-01T + 1.69245E+02$
0.2	$k= 4.56817E-09T^3 - 3.42932E-06T^2 - 3.64930E-02T + 9.01445E+01$
0.5	$k= 3.33540E-09T^3 - 7.83929E-06T^2 - 6.75616E-03T + 4.66649E+01$
1	$k=2.03348E-09T^3 - 5.51300E-06T^2 - 1.55010E-03T + 3.05337E+01$
3-8	$k= 1.20901E-06T^2 - 7.56914E-03T + 2.98193E+01$

- Range of validity [500K – 1800K], T in K

Table 12: Grade H-451 Graphite Thermo-Physical Properties

Property	Value*
Density (kg/m^3)	1740
Specific Heat (J/kg/K)	$C_p=(0.54212 - 2.42667E-06T - 9.02725E+01T^{-1} - 4.34493E+04T^{-2} + 1.59309E+07T^{-3} - 1.43688E+09T^{-4}) \times 4184$
Emissivity	0.85

* T in K

Graphical representations of both the thermal conductivity and the specific heat capacity are included as reference material to the participant in Figure 16 and Figure 17 respectively.

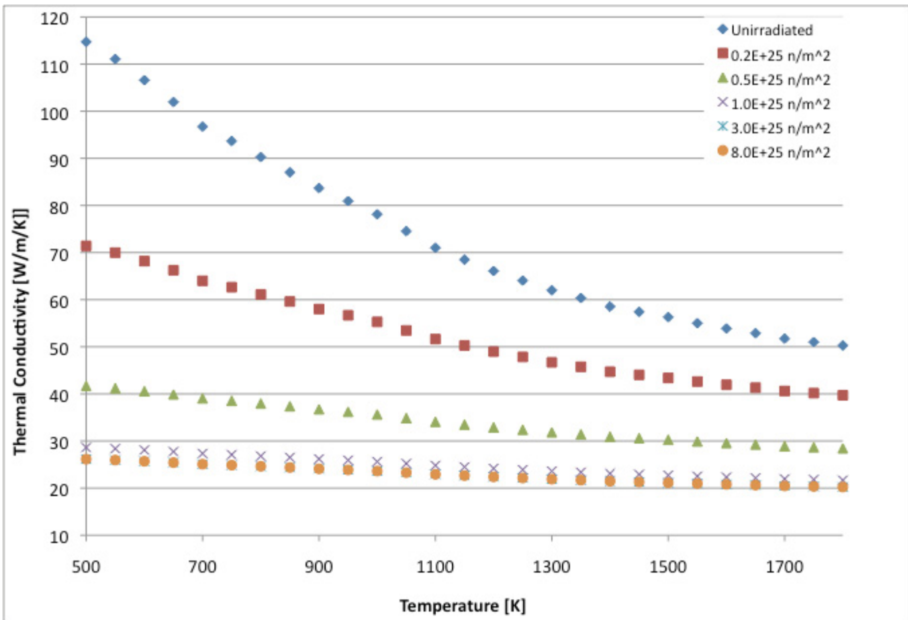

Figure 16: Thermal Conductivity of Grade H-451 Graphite

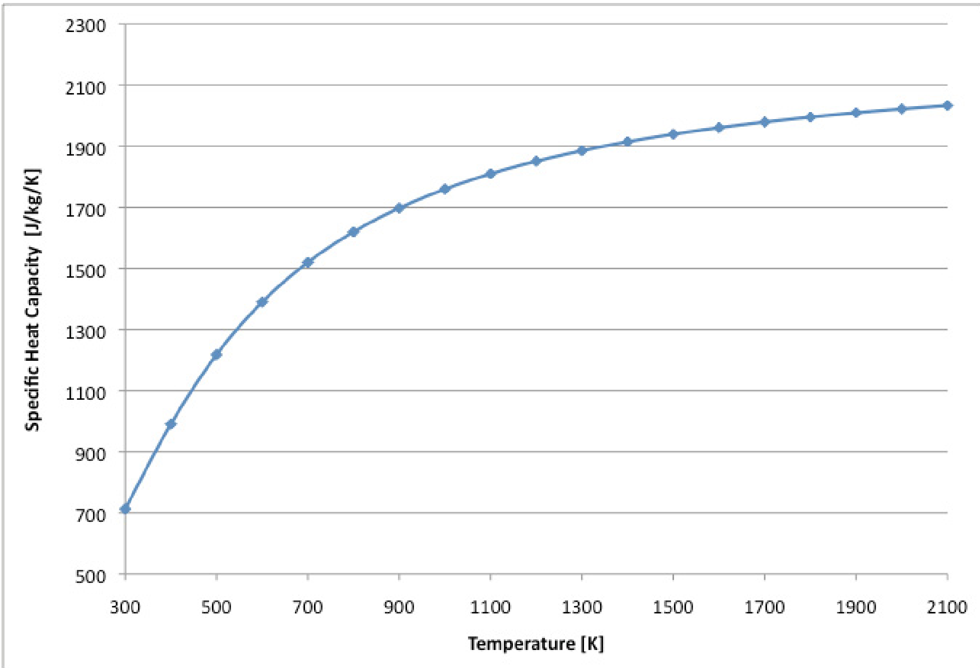

Figure 17: Specific Heat Capacity of Grade H-451 Graphite

8.3.2 Grade 2020 Graphite

The large rectangular grade 2020 graphite is a fine-grained, molded artificial graphite produced in large rectangular blocks. It is the reference material for permanent side reflectors and central reflector column support blocks.

Temperature and fluence dependent conductivities are included in Table 13. Conductivities for fluence levels that lie between the points provided can be linearly interpolated. Any value outside the range of validity will retain a constant value based on the closest value in the range of validity. The magnitude of the fluence is different than that provided for the H-451 graphite, 10^{22} versus 10^{25} n/m^2. A graphical representation of the thermal conductivity of grade 2020 graphite is shown in Figure 18. Other grade 2020 graphite thermal properties are shown in Table 14. The specific heat capacity of grade 2020 graphite is very similar to that of grade H-451 with the exception of the porosity correction factor.

Table 13: Thermal Conductivity of Grade 2020 Graphite

Fluence (10^{22} n/m^2)	Value (W/m/K)
Un-irradiated	k*= 1.71039E-07T^3-3.73458E-04T^2+2.18725E-01T+2.65411E+01
0.4	k**= 5.89227E-05T^2-1.28522E-01T+1.11808E+02
1	k**= 7.53255E-06T^2-3.46161E-02T+6.98153E+01
4	k**= -1.26995E-05T^2+1.08450E-02T+4.32150E+01
10	k**= -2.87164E-05T^2+4.83551E-02T+2.02541E+01
20	k**= -4.29785E-05T^2+8.18658E-02T-7.13659E-01

* Range of validity [295K – 1073K]
** Range of validity [673K – 1073K]

Table 14: Grade 2020 Graphite Thermo-Physical Properties

Property	Value
Density (kg/m^3)	1780
Specific Heat (J/kg/K)	see Table 16
Emissivity	0.85

A graphical representation of the thermal conductivity of grade 2020 is included as reference material to the participant in Figure 18.

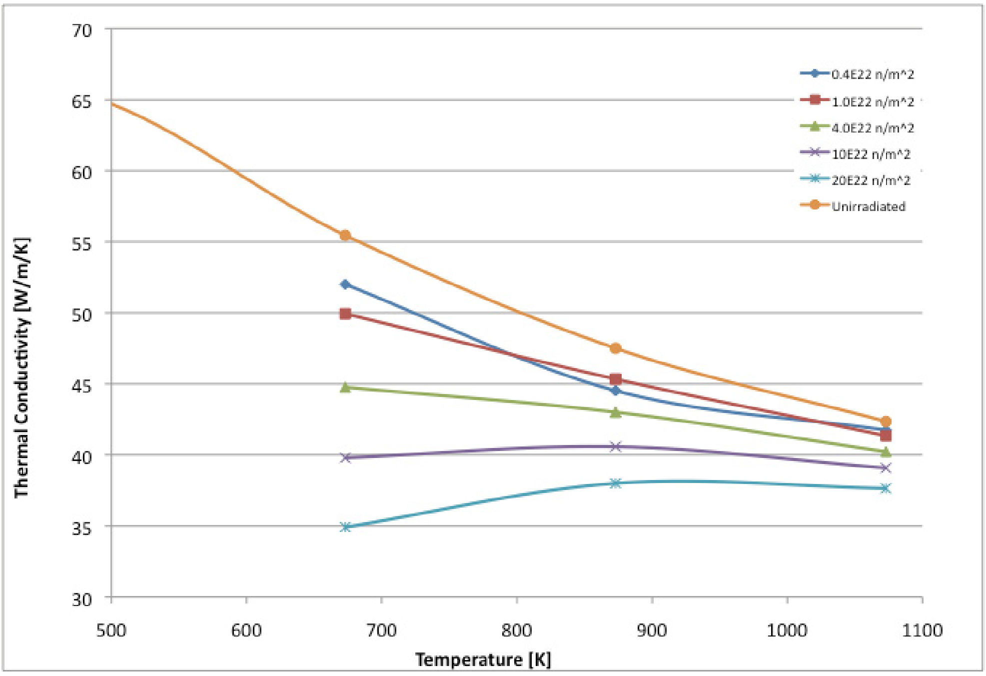

Figure 18: Thermal Conductivity of Grade 2020 Graphite

8.3.3 Fuel Compact Graphite Matrix

The fuel compact consists of a large number of coated particles imbedded in a graphite matrix. The graphite matrix thermo-physical properties are shown in Table 15.

Table 15: Compact Matrix Graphite Thermo-Physical Properties

Property	Value
Thermal Conductivity(W/m/K)	?
Density Matrix(kg/m^3)	1740
Specific Heat (J/kg/K)	see Table 16

8.3.4 Pyrolitic and Porous Graphite Layer

The TRISO particles include an outer and inner pyrolitic carbon layer as well as a porous graphite layer. The thermo-physical properties are included in Table 16.

Table 16: Pyrolitic and Porous Graphite Thermo-Physical Properties

Property	Value
Thermal Conductivity(W/m/K)	$k = 244.3T^{-0.574}\left[1-0.3662(1-e^{-1.1028\Gamma})-0.03554\Gamma\right]\left[\frac{\rho}{2.2(1930-\rho)+\rho}\right]$
Density PyC(kg/m^3)	1900
Density Porous(kg/m^3)	970
Specific Heat (J/kg/K)	$C_p = (0.54212 - 2.42667\mathrm{E}\text{-}6\mathrm{T}\ \text{-}9.02725\mathrm{E}1\ \mathrm{T}^{-1}\ \text{-}4.34493\mathrm{E}4\mathrm{T}^{-2}$ $+1.59309\mathrm{E}7\mathrm{T}^{-3}\ \text{-}1.43688\mathrm{E}9\mathrm{T}^{-4})\cdot 4184\cdot \left[\frac{\rho}{2.2(1740-\rho)+\rho}\right]$

Graphical representations of the thermal conductivity in the pryrolitic and porous carbon layers are included as reference material to the participant in Figure 19 and Figure 20, respectively.

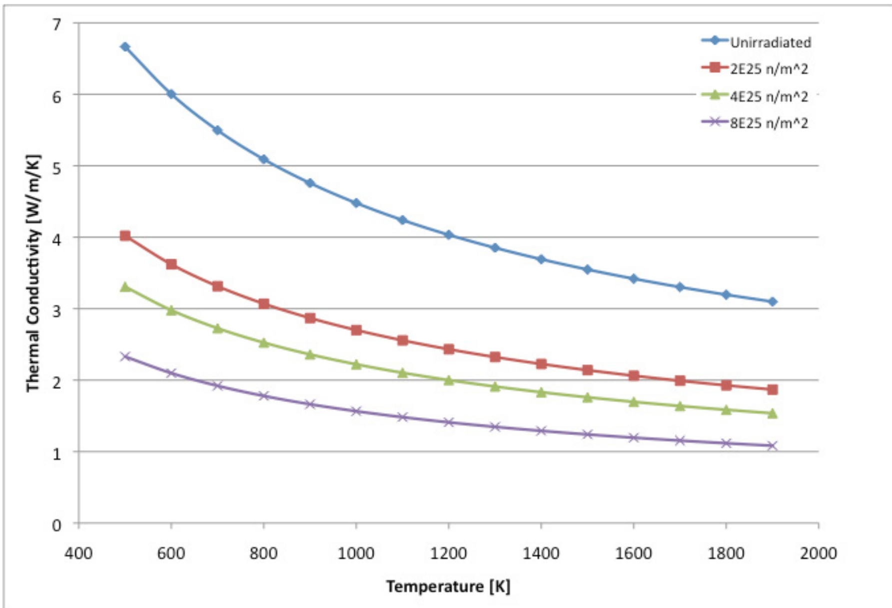

Figure 19: Thermal Conductivity of Pyrolitic Carbon

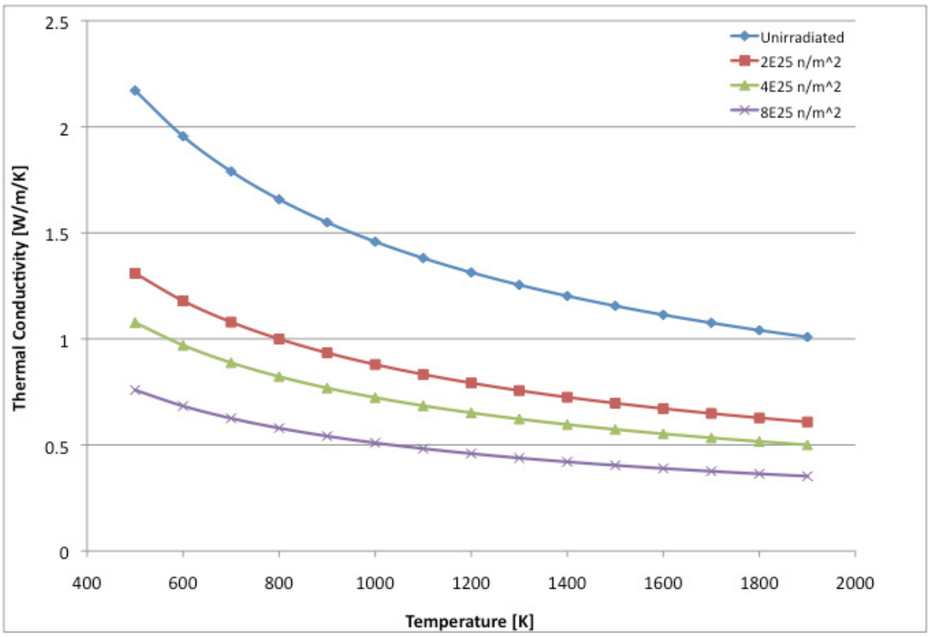

Figure 20: Thermal Conductivity of the Porous Graphite Layer

8.3.5 SiC Layer

The SiC layer is and essential constituent of the TRISO particle. It provides the fission product barrier against release. The thermo-physical properties are shown in Table 17.

Table 17: SiC Thermo-Physical Properties

Property	Value
Thermal Conductivity(W/m/K)	$k_{unirr} = \left(\frac{17885}{T} + 2\right)e^{-0.1277\Gamma}$
Density (kg/m^3)	4210
Specific Heat (J/kg/K)	

Γ = neutron fluence in 10^{25} n/m^2 DNE units, T is in Kelvin.

Graphical representations of both the thermal conductivity and the specific heat capacity are included as reference material to the participant Figure 21 and Figure 22, respectively.

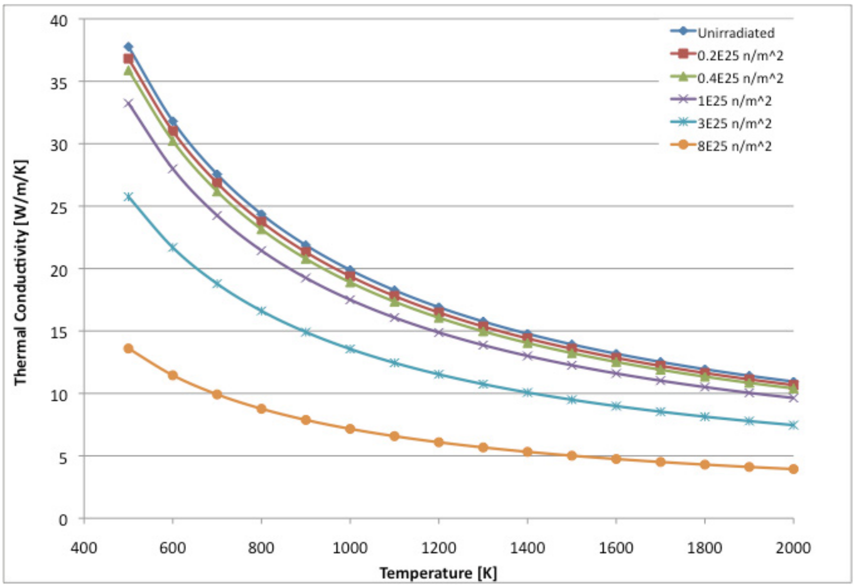

Figure 21: Thermal Conductivity of the SiC Layer

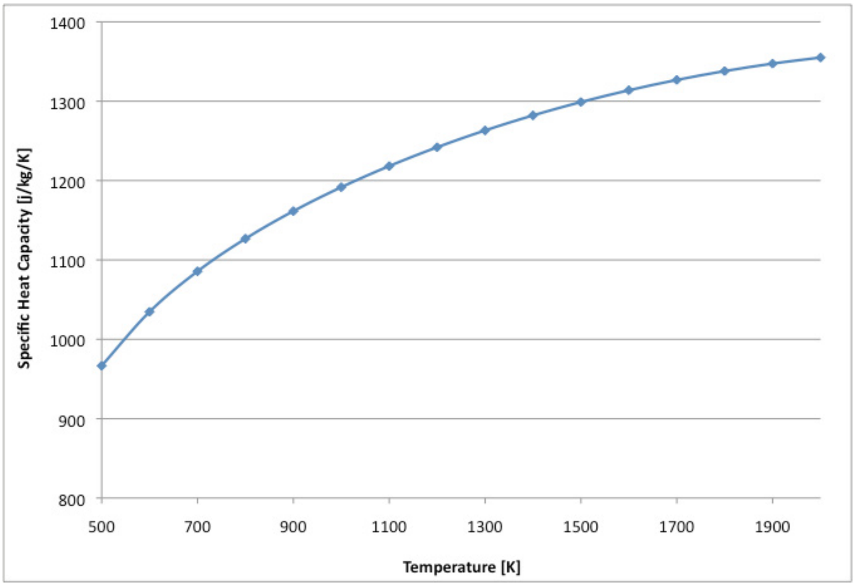

Figure 22: Specific Heat Capacity of the SiC Layer

8.3.6 $UC_{0.5}O_{1.5}$ Kernel

The thermo-physical properties for the uranium oxycarbide kernel are included in Table 18.

PENDING INPUT FROM GA

Table 18: $UC_{0.5}O_{1.5}$ Thermo-Physical Properties

Property	**Value**
Thermal Conductivity(W/m/K)	
Density (kg/m^3)	
Specific Heat (J/kg/K)	

8.3.7 Core Barrel

The thermo-physical properties for the core barrel are shown in Table 19.

PENDING INPUT FROM GA

Table 19: Core Barrel Thermo-Physical Properties

Property	**Value**
Thermal Conductivity(W/m/K)	15.6

Density (kg/m^3)	7800
Specific Heat (J/kg/K)	525
Emissivity	0.85

8.3.8 Pressure Vessel

The thermo-physical properties for the core barrel are shown in Table 20.

PENDING INPUT FROM GA

Table 20: Pressure Vessel Thermo-Physical Properties

Property	**Value**
Thermal Conductivity(W/m/K)	33.0
Density (kg/m^3)	7800
Specific Heat (J/kg/K)	525
Emissivity	0.85

8.3.9 Core Restraint Element / Metallic Core Support Structure / Upper Plenum Thermal Protection Structure

The thermo-physical properties for the various core support and thermal protection structures are shown in Table 21.

PENDING INPUT FROM GA

Table 21: CRE / MCSS / UPTPS Thermo-Physical Properties

Property	**Value**
Thermal Conductivity(W/m/K)	15.6
Density (kg/m^3)	7800
Specific Heat (J/kg/K)	525
Emissivity	0.85

8.3.10 Coolant (Helium)

The thermo-physical properties of the helium coolant are included in Table 22.

Table 22: Helium Coolant Thermo-Physical Properties

Property	**Value**
Thermal Conductivity(W/m/K)	$k = 2.682e^{-4} \times \left(1 + 1.123e^{-3} \times {}^{P}\!/_{100}\right) \times T^{C}$ where $C = 0.71 \times \left(1 - 2.0e^{-4}\right) \times {}^{P}\!/_{100}$
Density (kg/m^3)	ideal gas
Specific Heat (J/kg/K)	5195

8.4 BOUNDARY CONDITIONS

Table 23: Boundary Conditions

	Thermal Hydraulic model boundaries		
1	Radial (cm)		463.3
2	Top		1745.63
3	Bottom		0
	Thermal Hydraulic boundary conditions		
4	Radial	Constant Temperature	30°C
		RCCS heat removal rate	0 - 1.74 MW ??
5	Top		Adiabatic
6	Bottom		Adiabatic

8.5 DECAY HEAT SOURCES

For the benchmark problem the decay heat source is only of importance in certain transient cases, typically where the fission power is reduced to zero during the event. For the steady-state cases the decay heat is assumed to be part of the energy released per fission, which are assumed in this specification to be all deposited locally, i.e. where the fission took place.

The decay heat value for each material mesh in the core must be derived making use of the relative core average decay heat behaviour (values provided as determined from the DIN 25485 standard) and the material mesh power. This implies that the decay heat is directly related to the steady-state power produced in the mesh prior to the start of the transient. No history effects or power excursions after the start of the transient (t=0) should be taken into account and the time when decay start is indicated for the appropriate transient cases. Note that the decay heat contribution is ignored in certain transient cases.

The decay heat as calculated for the EOEC is shown in.

9. PHASE I: STEADY STATE BENCHMARK CALCULATIONAL CASES

9.1 CASE DEFINITIONS

9.1.1 Exercise 1: Neutronics Solution with Fixed Cross Sections

Make use of model description and the following conditions:

• Use the simplified cross section set with no state parameter dependence

• No thermal-hydraulic solution required

• Report keff, power profile,.

• calculational mesh to be determined by each participant

9.1.2 Exercise 2: Thermal Fluids solution with given power / heat sources

Make use of the thermal hydraulic properties and model description and the following conditions:

• The provided power / heat source density given in through **Error! Reference source not found.**should be used. The values correspond to the various axial layers 1 – 10, with 1 .

• Calculate the temperatures distribution, outlet temperature, pressure drop over the core and heat loss to the constant temperature boundary.

• calculational mesh to be determined by each participant.

9.1.3 Exercise 3: Coupled Neutronics - Thermal Fluids Steady State Solution

9.2 CONVERGENCE CRITERIA FOR PHASE I

General convergence criteria guidelines for Phase I are provided in . Each participant should ensure that a well-converged result is obtained by performing a sensitivity study on the code-specific input parameters, mesh sizes and acceleration parameters.

Table 24: Suggested convergence criteria – PHASE I

Parameter	Unit	Convergence criteria
k-eff		1.0E-6
Local Fluxes		1.0E-4
Local Temperatures	°C	0.1
Local Flows	kg/s	0.1

10. PHASE II: TRANSIENT BENCHMARK

10.1 EXERCISE DEFINITIONS

The sequence of events, boundary conditions and requested output parameters are specified for four transient scenarios in the subsections below. The four exercises consist of two Loss of Cooling Accidents (LOCA) at pressurised and depressurised conditions, a moisture ingress event with a reactor trip, and an operational power load follow transient from 100% power to 80% and back. All transients start at t = 0 seconds, with a data output required at this point to capture the starting steady-state situation for each of the transient cases. MS Excel templates will be provided to ensure that participants supply the correct output data in the same format and at the same time points.

For this benchmark, some modifications were made to the MHTGR Safety Report [6] or PRA [7] event sequences to enable the completion and/or isolation of some phenomena. An example of this is the delay in the reactor trip during the moisture ingress event (exercise 3) to allow sufficient time for a full analysis of the reactivity insertion progression. The event sequences specified here should therefore not be seen as representative of the MHTGR's safety case in any way, but purely as the basis for code-to-code comparisons.

10.1.1 Exercise 1: Depressurized Conduction Cooldown (DCC) without reactor trip

The sequence of events for a typical Depressurized Conduction Cooldown (DCC) transient is specified in Table 25. This event is the equivalent of a Loss of Cooling Accident (LOCA) in LWRs, and is also referred to as a Depressurised Loss of Forced Cooling (DLOFC). It represents the bounding case for the fuel temperatures, and is usually initiated by a large break in the system pressure boundary. The following aspects should be noted:

- Since a reactor trip is not performed, re-criticality will occur. The control rod position remains constant at the steady state depth.
- The provided decay heat fit data should be used for this exercise, i.e. decay heat should *not* be calculated using the participants' own routines.
- For simplicity and decreased calculation times, the effects of natural convection are excluded for this exercise. It can be assumed that all convective heat transport terminates after 20 s.
- The system pressure equalises at atmospheric (1 MPa), and the inlet mass flow rate decreases to 0 kg/s.
- It is assumed that the decrease in the pressure and mass flow rates are linear between 0 and 20 s. No flow reversal occurs during the decrease of the mass flow.
- The Shutdown Cooling System (SCS) is not operational (i.e. the helium flow rate remains at 0.0 kg/s through the entire transient), but the Reactor Cavity Cooling System (RCCS) is in active mode. The RCCS inlet water temperature will not be ramped up with time, i.e. the radial temperature boundary should remain at the RCCS normal operation temperature of 30^{o}C. (A second case is possible where the 800? kW [TDB] removal capability of the RCCS is specified, instead of the constant temperature boundary).

- The transient is calculated for a total of 150 hours. To enable consistent output data comparisons, output data (see Table 43 for definitions) must be provided in the Excel templates at the requested time points.

Table 25: Transient exercise 1 - sequence of events and required data edits

Time (seconds)	Description	Required spatial output data	Required global output data
0	Equilibrium steady state completed for the End-of-Life core state.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature and power density. Single parameter value for axial power offset.	Maximum and average fuel temperature, total and decay power and total heat loss rate through the radial boundary. Required at every second for the first 60 seconds, then every 5 minutes for the first hour, then every 10 minutes up to the end of the transient. (See Excel template).
0 – 20	Reduce the reactor inlet coolant mass flow from nominal to 0.0 kg/s over 20 seconds. The mass flow ramp is assumed linear.	None	
0 – 20	Reduce the reactor outlet pressure from nominal to 1 MPa (atmospheric) over 20 seconds. The pressure ramp is assumed linear. (Note that all pressures defined in this benchmark are absolute pressure values, and not gauge values).	None	
20	Depressurisation phase completed. All convective heat transport can be disabled (i.e. heat removal by natural convection is not calculated).	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature and power density. Single parameter value for axial power offset.	
Re-critical phase (time will vary between participants)	The core should attain re-criticality after some time, with sufficient cool down and xenon decay.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature and power density. Single parameter value for axial power offset.	
~ 540000	Transient case completed at 150 hours, or at least 10 hours after re-criticality if it occurs later than 150 hours.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature and power density. Single parameter value for axial power offset.	

10.1.2 Exercise 2: Pressurized Conduction Cooldown (PCC) with reactor trip

The sequence of events for the Pressurized Conduction Cooldown (PCC) transient, as shown in Table 26, is identical to the DCC transient except for the higher system pressure and the addition of a reactor trip. This event is also referred to as a Pressurised Loss of Forced Cooling (PLOFC), and can be initiated by a turbine trip. No pressure boundary break occurs during this transient. The following aspects should be noted:

- The turbine spin down is assumed to be completed at 30 s.
- The control rods are inserted (reactor trip) to the shutdown position 300 s into the transient. An insertion time of 3 seconds can be assumed.
- The provided decay heat fit data should be used for this exercise, i.e. decay heat should *not* be calculated using the participants' own routines.
- The effects of natural convection must be calculated for this exercise, since the full helium coolant inventory is still present in the primary system.
- The system pressure equalises at the stagnant system pressure of 5 MPa, and the inlet mass flow rate decreases to 0 kg/s. (Under review)
- It is assumed that the decrease in the pressure and mass flow rates are linear between 0 and 20 s. No flow reversal occurs during the decrease of the mass flow.
- The Shutdown Cooling System (SCS) is not operational (i.e. the helium flow rate remains at 0.0 kg/s through the entire transient), but the Reactor Cavity Cooling System (RCCS) is in active mode. The RCCS inlet water temperature will not be ramped up with time, i.e. the radial temperature boundary should remain at the RCCS normal operation temperature of 30^{o}C.
- The transient is calculated for a total of 150 hours. To enable consistent output data comparisons, output data (see Table 43 for definitions) must be provided in the Excel templates at the requested time points.

Table 26: Transient exercise 2 - sequence of events and required data edits

Time (seconds)	Description	Required spatial output data	Required global output data
0	Equilibrium steady state completed for the End-of-Life core state.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, mass flow rate and power density.	Maximum and average fuel temperature and total heat loss rate through the radial boundary. Required at every second for the first 60 seconds, then every 5 minutes for the first hour, then every 10 minutes up to the end of the transient. (See Excel template).
0 – 30	Reduce the reactor inlet coolant mass flow from nominal to 0.0 kg/s over 30 seconds. The mass flow ramp is assumed linear.	None	
0 – 30	Reduce the reactor outlet pressure from nominal to 5 MPa over 30 seconds. The pressure ramp is assumed linear.	None	
30	Turbine trip phase completed.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, mass flow rate and power density. Single parameter value for axial power offset.	

Time (seconds)	Description	Required spatial output data	Required global output data
300	Initiate reactor trip – insert control rods to shutdown position over 3 seconds.		
~ 540000	Transient case completed at 150 hours.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, mass flow rate and power density. Single parameter value for axial power offset.	

10.1.3 Exercise 3: Moisture (or Water) Ingress with reactor trip

The sequence of events for the moisture ingress transient is shown in Table 27. This event can be initiated by leaks or breaks in the Steam Generator (SG) tubes, and results in a reactivity increase in the primary system. The following aspects should be noted:

- The interaction of steam and the graphite core structures (fuel, reflectors) does not need to be calculated, i.e. the benchmark scenario assumes that no graphite corrosion will take place.
- The steam ingress must be treated homogenously over the entire core volume as a porous medium mixture of steam and helium gas. An explicit "steam plume" tracking is therefore not required, but if participants codes' already have this capability, a second set of results would be encouraged to compare the two approaches.
- Typical SG leak rates into the core vary from 0.0023 kg/s (small leaks), 0.28 kg/s for a single tube rupture and up to 6.7 kg/s for a multiple tube rupture event [6]. The bounding multiple SG tube rupture scenario (6.7 kg/s) is selected for the benchmark to obtain the maximum reactivity response from the End of Life core, although it has a low probability to occur.
- For this scenario, a reactor trip signal is usually initiated after 30 s of steam ingress on a high system pressure signal. The SG isolation valve is also closed at the same time, effectively terminating further steam ingress into the core region. For this benchmark case, the reactor trip signal is delayed until 80 s in order to observe the full core reactivity behaviour, and the SG isolation valve closure is only specified at 90s. (This checks if the rod insertion actually counters the reactivity rise successfully). The total mass of steam reaching the core region is therefore 603 kg, which corresponds with the 600kg Design Basis accident considered for the HTR Module [8].
- The control rods are inserted (reactor trip) to the shutdown position 80 s into the transient. No other control rod adjustments should be made before or after this time point. An insertion time of 3 seconds can be assumed.
- The main cooling system stays operational throughout the transient, i.e. a turbine trip does not occur. All other boundary conditions also remain at their nominal normal operation values. In order to isolate the hydrogen moderation effect, the incoming steam temperature is assumed to be identical to the inlet helium gas temperature (i.e. the mixture temperature is 250°C). The rise in the system pressure (due to the

mixture of heavier steam and helium in the system) is likewise not included in the benchmark scenario.

- The transient is calculated for a total of 1000 seconds. To enable consistent output data comparisons, output data (see Table 43 for definitions) must be provided in the Excel templates at the requested time points.

Table 27: Transient exercise 3 - sequence of events and required data edits

Time (seconds)	Description	Required spatial output data	Required global output data
0	Equilibrium steady state completed for the End-of-Life core state.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, power density.	Maximum and average fuel temperature, total power, reactivity, axial offset. Required at every second for the first 100 seconds, then every 5 seconds up to the end of the transient. (See Excel template).
0 – 90	Initiate and continue steam ingress into the primary system at a constant rate of 6.7 kg/s.	None	
80	Initiate reactor trip – insert control rods to shutdown position over 3 seconds.		
90	Terminate steam ingress.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, power density.	
90-1000	No changes in input parameters.	None	
1000	Transient case completed at 1000 s.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, power density.	

10.1.4 Exercise 4: Power 100-80-100 load follow

Although the End of Life core reactivity is limited, it will assumed for this exercise that a sufficient control rod margin exist to overcome the build-up in xenon for the decrease to 80% of full power. The ramp back to full power is specified close to the xenon peak around 7 hours after the initial decrease. The sequence of events for the 100-80-100% load follow transient is shown in Table 28. (Note that this event is a normal operation transient, as opposed to the accidents specified in exercises 1-3). The following aspects should be noted:

- It is not required to model the control rod movements that are usually performed to keep the core critical. The control rods should remain in the normal operation inserted location for the duration of the exercise. Since the total power, inlet mass flow rate and pressure are defined as input, the reactivity required to keep the reactor critical needs to be calculated during the load follow. (As an example: with the decrease in power the core average xenon levels will increase, say to around 102% of the nominal equilibrium xenon levels. In an operating HTR, this increase in xenon poison will be offset by withdrawing the control rods, to add the required 2% reactivity and still remain at constant power (in LWRs the xenon increase can also be balanced by global soluble boron dilution). If the rods are static, an artificial reactivity "requirement" can be calculated by simply tracking k-eff – and therefore reactivity - as a function of time).

- It is assumed that the decrease in the pressure and mass flow rates are linear over the 60 seconds it takes to decrease the reactor power by 20%. No flow reversal occurs during the decrease of the mass flow.
- All boundary conditions remain at their nominal normal operation values.
- The transient is calculated for a total of 72 hours to track the build-up and decay of xenon and the resultant power oscillations, if any. To enable consistent output data comparisons, output data (see Table 43 for definitions) must be provided in the Excel templates at the requested time points.

Table 28: Transient exercise 4 - sequence of events and required data edits

<table>
<tr><th>Time (seconds)</th><th>Description</th><th>Required spatial output data</th><th>Required global output data</th></tr>
<tr><td>0</td><td>Equilibrium steady state completed for the End-of-Life core state.</td><td>Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, xenon concentration/absorption and power density.</td><td rowspan="6">Maximum and average fuel temperature, reactor power, core average xenon concentration, reactivity, axial offset.
Required at every second for the first 120 seconds, then every 10 minutes up to the end of the transient.
(See Excel template).</td></tr>
<tr><td>0 – 60</td><td>Reduce the reactor inlet coolant mass flow and system pressure from nominal to 80% over 60 seconds. The ramps are assumed to be linear.
Reduce the total reactor power from 100% to 80% over 60 seconds. The ramp is assumed to be linear.</td><td>None</td></tr>
<tr><td>60</td><td>100-80% phase completed.</td><td>Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, xenon concentration/absorption and power density.</td></tr>
<tr><td>60-25200</td><td>No changes in input parameters.</td><td>Spatial maps of the xenon concentration /absorption and power density on the intervals specified in the Excel template.</td></tr>
<tr><td>25200-25260</td><td>Initiate the mass flow rate, pressure and power levels ramps back to their 100% values over 60 seconds. The ramps are assumed to be linear.</td><td>None</td></tr>
<tr><td>25260</td><td>80-100% phase completed.</td><td>Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, xenon concentration/absorption and power density.</td></tr>
</table>

Time (seconds)	Description	Required spatial output data	Required global output data
25260-259200	No changes in input parameters.	Spatial maps of the xenon concentration /absorption and power density on the intervals specified in the Excel template.	
259200	Transient case completed at 72 hours.	Spatial maps of the maximum and average fuel temperature, moderator/reflector temperature, xenon concentration/absorption and power density on the intervals specified in the Excel template.	

10.2 CONVERGENCE CRITERIA FOR PHASE II

General convergence criteria guidelines are provided in Table 29. Each participant should ensure that a well-converged result is obtained by performing a sensitivity study on the code-specific input parameters, mesh sizes and acceleration parameters.

Table 29: Suggested convergence criteria – PHASE II

Parameter	Unit	Convergence criteria
Local Fluxes		1.0E-4
Local Temperatures	°C	0.1
Local Flows	kg/s	0.1

11. PHASE III: CROSS SECTION GENERATION

The macroscopic cross section generation phase is intended to show the variation in the data preparation process between benchmark participants. The case definition starts with a simple single cell calculation. The level of complexity is gradually increased and finally a full block depletion is performed. The following data will be compared for each exercise in the cross section preparation phase in two different group structures: kinf, total, absorption, fission, nubar-fission, transport cross section, and scattering cross sections. The group structure that should be used in the collapsing is shown in Table 30.

Table 30: Energy Group Structure

Group	Upper Energy (eV)	Group	Upper Energy (eV)
1	1.49E+07	14	1.370E+01
2	7.41E+06	15	8.320E+00
3	3.68E+06	16	5.040E+00
4	6.72E+05	17	2.380E+00
5	1.11E+05	18	1.290E+00
6	1.93E+04	19	6.500E-01
7	3,355.00	20	3.500E-01
8	1,585.00	21	2.000E-01
9	7.485E+02	22	1.200E-01
10	2.754E+02	23	8.000E-02
11	1.301E+02	24	5.000E-02
12	6.144E+01	25	2.000E-02
13	2.900E+01	26	1.000E-02

11.1 CASE DEFINITIONS

Note that in all exercises the gap between the compact and the block or the sleeve has been omitted.

11.1.1 Exercise 1: Homogeneous Cell

Table 31: Dimensions for Exercise 1

	Dimension	Units
Fuel Compact OR	0.635	cm
Cell Outer radius	0.98686	cm

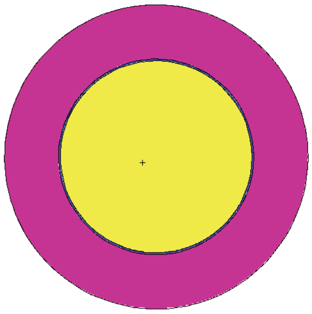

Figure 23: Graphical Representation of Exercise 1

Table 32: Number Densities for Exercise 1

Nuclide	**Number Density (atoms/b·cm)**
U-235	8.39E-05
U-238	4.52E-04
O-16	8.04E-04
C-12 (graphite)	6.53E-02
Si-28	8.05E-05
Si-29	5.34E-05
Si-30	1.31E-04
C-12 Matrix (Graphite)	8.53E-02

11.1.2 Exercise 2: Double heterogeneous Cell

Table 33: Dimensions for Exercise 2

<table>
<tr><th colspan="2"></th><th>Dimension</th><th>Units</th></tr>
<tr><td rowspan="5">TRISO Fuel Particle</td><td>Kernel Radius</td><td>2.125E-02</td><td>cm</td></tr>
<tr><td>Porous Carbon Buffer OR</td><td>3.125E-02</td><td>cm</td></tr>
<tr><td>IPyC OR</td><td>3.625E-02</td><td>cm</td></tr>
<tr><td>SiC OR</td><td>3.975E-02</td><td>cm</td></tr>
<tr><td>OPyC OR</td><td>4.375E-02</td><td>cm</td></tr>
<tr><td colspan="2">TRISO Packing Fraction</td><td>0.35</td><td></td></tr>
<tr><td colspan="2">Fuel Compact OR</td><td>0.635</td><td>cm</td></tr>
<tr><td colspan="2">Cell Outer radius</td><td>0.98686</td><td>cm</td></tr>
</table>

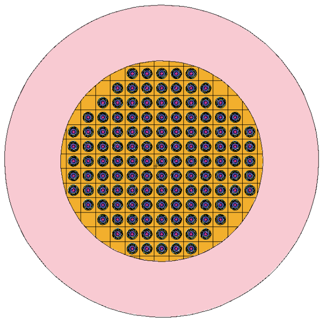

Figure 24: Graphical Representation of Exercise 2

Table 34: Number Densities for Exercise 2

NUMBER DENSITIES		**Nuclide**	**N (at/b-cm)**
TRISO Fuel Particle	Kernel	U-235	3.70E-03
		U-238	1.99E-02
		O-16	3.55E-02
		C-12	1.18E-02
	Porous Carbon	C-12 (Graphite)	5.02E-02
	IPyC	C-12 (Graphite)	9.53E-02
	SiC	Si-28	4.43E-02
		Si-29	2.25E-03
		Si-30	1.49E-03
		C-12	4.81E-02
	OPyC	C-12 (Graphite)	9.53E-02
Compact Matrix		C-12 (Graphite)	8.53E-02

11.1.3 Exercise 3: Block with Homogenized Fuel Compacts

Table 35: Dimensions for Exercise 3

	Dimension	**Units**
Homogenized Fuel Channel OR	0.635	cm
Large Coolant Channel OR	0.794	cm
Small Coolant Channel OR	0.635	cm
Pin pitch	1.88	cm
Block Flat-to-Flat Dimension	36.0	cm

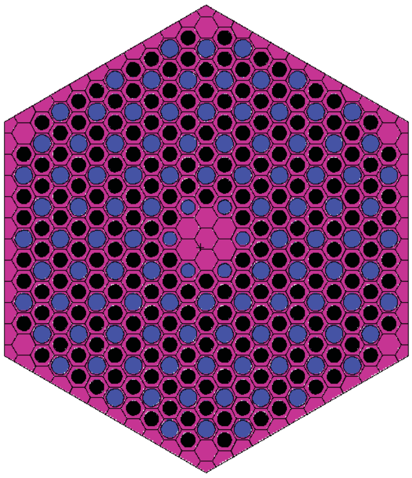

Figure 25: Graphical Representation of Exercise 3

Table 36: Number Densities for Exercise 3

	Nuclide	**N (at/b-cm)**
Homogenized Fuel Region	U-235	1.43E-04
	U-238	7.68E-04
	O-16	1.37E-03
	C-12 (graphite)	7.55E-02
	He-4	2.70E-03
	Si-28	1.37E-04
	Si-29	9.08E-05
	Si-30	2.73E-05
Block Graphite	C-12 (graphite)	9.28E-02
Coolant Channels	He-4	2.46E-05

11.1.4 Exercise 4: Block with Double Heterogeneous Compacts

Table 37: Dimensions for Exercise 4

	Dimension	**Units**
Kernel OR	2.125E-02	cm
Porous Carbon OR	3.125E-02	cm
IPyC OR	3.625E-02	cm
SiC OR	3.975E-02	cm
OPyC OR	4.375E-02	cm
TRISO Packing Fraction	0.35	
Compact OR	0.635	cm
Large Coolant Channel OR	0.794.	cm

Small Coolant Channel OR	0.635	cm
Pin pitch	1.88	cm
Block Flat-to-Flat Width	36.0	cm

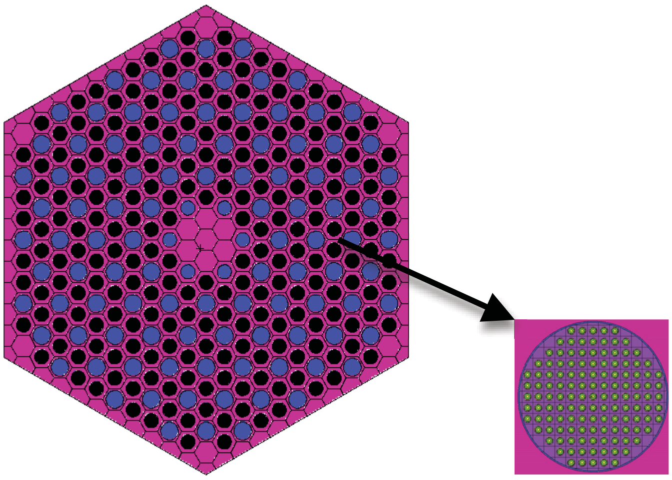

Figure 26: Graphical Representation of Exercise 4

Table 38: Number Densities for Exercise 4

NUMBER DENSITIES		**Nuclide**	**N (at/b-cm)**
TRISO Fuel Particle	Kernel	U-235	3.70E-03
		U-238	1.99E-02
		O-16	3.55E-02
		C-12	1.18E-02
	Porous Carbon	C-12 (Graphite)	5.02E-02
	IPyC	C-12 (Graphite)	9.53E-02
	SiC	Si-28	4.43E-02
		Si-29	2.25E-03
		Si-30	1.49E-03
		C-12	4.81E-02
	OPyC	C-12 (Graphite)	9.53E-02
Compact Matrix		C-12 (Graphite)	8.53E-02
Block Graphite		C-12 (Graphite)	9.28E-02
Coolant Channels		He-4	2.46E-05

11.1.5 Exercise 5: Block with Double Heterogeneous Compacts and BP

The geometry for this case is identical to Exercise 4 with the exception of the presence of 6 FBP, one in each corner of the assembly (shown in Figure 27).

Table 39: Dimensions for Exercise 5

		Dimension	Units
TRISO Fuel Particle	Kernel Radius	2.125E-02	cm
	Porous Carbon Buffer OR	3.125E-02	cm
	IPyC OR	3.475E-02	cm
	SiC OR	3.825E-02	cm
	OPyC OR	4.225E-02	cm
	TRISO Packing Fraction	0.35	
	Compact Radius	0.6225	cm
	Gap Radius	0.635	cm
Burnable Poison Particle	Kernel Radius	1.00E-02	cm
	Kernel Density	2.4696	g/cm^3
	Porous Carbon Buffer OR	1.18E-02	cm
	PyC OR	1.41E-02	cm
	BP Particle Packing Fraction	0.305	
	BP Compact Radius	0.5715	cm
Large Coolant Channel Radius		0.794.	cm
Small Coolant Channel Radius		0.635	cm
Pin pitch		1.88	cm
Block Flat-to-Flat Width		36.0	cm

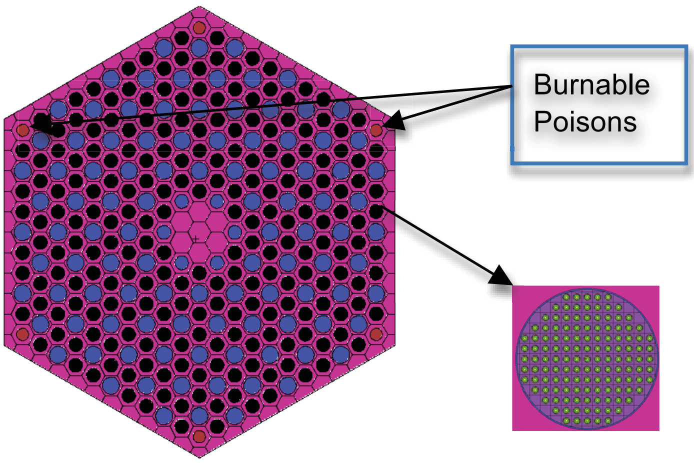

Figure 27: Graphical Representation of Exercise 5

Table 40: Number Densities for Exercise 5

NUMBER DENSITIES		Nuclide	N (at/b-cm)
TRISO Fuel Particle	Kernel	U-235	3.70E-03
		U-238	1.99E-02
		O-16	3.55E-02
		C-12	1.18E-02
	Porous Carbon	C-12 (Graphite)	5.02E-02
	IPyC	C-12 (Graphite)	9.53E-02
	SiC	Si-28	4.43E-02
		Si-29	2.25E-03
		Si-30	1.49E-03
		C-12	4.81E-02
	OPyC	C-12 (Graphite)	9.53E-02
Burnable Poison Particle	Kernel	B-10	2.14E-02
		B-11	8.63E-02
		C-12	2.69E-02
	Buffer	C-12 (Graphite)	5.02E-02
	PyC	C-12 (Graphite)	9.38E-02
BP Compact Matrix		C-12 (Graphite)	6.87E-02
Fuel Compact Matrix		C-12 (Graphite)	8.27E-02
Block Graphite		C-12 (Graphite)	9.28E-02
Coolant Channels		He-4	2.46E-05

11.1.6 Exercise 6: Depletion of a Block with Double Heterogeneous Compacts and BP

This exercise entails the depletion of the detailed fuel block from exercise 5.

Table 41: Number Densities for Exercise 6

Variable	Value
Specific power	70 W/g HM
Burnup	0,50,100,200,300,400,500,540 EFPD

11.2 CONVERGENCE CRITERIA FOR PHASE III

General convergence criteria guidelines are provided in . Each participant should ensure that a well-converged result is obtained by performing a sensitivity study on the code- specific input parameters, mesh sizes and acceleration parameters.

Table 42: Suggested convergence criteria – PHASE III

Parameter	**Unit**	**Convergence criteria**
k-eff		1.0E-6
Local Fluxes		1.0E-5

12. REQUESTED OUTPUT

12.1 MESH DEFINITION FOR REPRESENTATION OF RESULTS

Even though the calculation mesh can be refined to achieve the desired spatial convergence, the reporting mesh will be block-wise in all regions except for the permanent reflector region. In the permanent reflectior the …

TBD

12.1.1 Neutronics results

12.1.2 Thermal fluids results

12.2 PHASE I: STEADY STATE OUTPUT PARAMETERS

Data output and reporting formats are included as MS Excel templates. These templates must be used for all reporting on data for the all cases.

TBD

12.3 PHASE II: TRANSIENT OUTPUT PARAMETERS

Data output and reporting formats are included as MS Excel templates. These templates must be used for all reporting on transient data for the 4 cases. Transient output results are grouped into 2 sections: the time-history values of some important parameters (e.g. total power, maximum fuel temperature, power density, axial power offset), and “snapshot” information on the spatial values of these parameters at specific time-points. The output required for each transient case will be specified as part of the case description. The output parameters are described in Table 43.

Table 43: Output parameter definition – PHASE II

Parameter	Description	Unit
Axial (power) offset	AO = $(TP_{top} - TP_{bottom}) / (TP_{top} + TP_{bottom})$, where TP_{top} = total power produced in the top half of the core, and TP_{bottom} = total power produced in the bottom half of the core	None.
Average and Maximum Fuel Temperature	The “fuel temperature” is defined as the average fuel *compact* temperatures of all the fuel elements present in a single mesh. (This is the value used for Doppler feedback calculations in each mesh point). The maximum fuel temperature of a fuel compact refers to the maximum fuel temperature seen by an $UC_{0.5}O_{1.5}$ kernel, normally in the centre of a fuel compact. The core maximum fuel temperature is the highest of all the maximum fuel temperatures that occurs in the entire spatial map.	^{o}C
Average and maximum moderator temperature	The “moderator temperature” is defined as the average temperatures of all fuel graphite matrix present a single mesh. (This is the value used for moderator temperature feedback calculations in each mesh point). The maximum value that occurs in the 2D spatial map is defined as the “maximum moderator temperature”, and the average of all the spatial moderator temperatures in all meshes is defined as the “average moderator temperature”.	^{o}C
Helium mass flow rate	Mass flow rate of Helium coolant in the system, reported in it directional components (radial and axial).	kg/s
Total power	Total thermal power (i.e. fission power + decay heat).	MW
Power density	Power density that occurs in each of the spatial meshes.	MW/m^3
Core pressure drop	Drop in helium pressure over the core (i.e. $P_{core\ inlet} - P_{core\ outlet}$).	kPa
Reactivity	Global reactivity change, as caused by transient feedback effects.	% $\Delta k/k$
Xenon concentration	Xenon concentration/absorption that occurs in each of the spatial meshes. Since the units for this parameter varies in the different codes, relative xenon values normalised to the steady-state xenon concentration levels in each of the spatial meshes will be used.	None (normalised to steady-state).

12.4 PHASE III: CROSS SECTION GENERATION PARAMETERS

Data output and reporting formats are included as MS Excel templates. These templates must be used for all reporting on cross section data for the all cases. Table 44 includes the data that needs to be reported for the exercises in PHASE III.

Table 44: Output parameter definition – PHASE III

Parameter	Description	Unit
Number of Fine groups	Number of energy groups used in the transport solution	NA
Group Structure	Include the actual group structure used in the collapsing of the cross sections. (Actual boundaries of the 26 group collapse)	NA
k-inf	Infinite medium eigenvalue	NA
Macroscopic Cross Sections		
Total	Total cross section	cm-1
Absorption	Absorption cross section	cm-1
Fission	Fission cross section	cm-1
Nu-fission	Nu-fission cross section	cm-1
Transport	Transport cross section	cm-1
Scattering	Full scattering matrix	cm-1

13. METHOLOGIES FOR COMPARATIVE ANALYSIS

TDB

14. REFERENCES

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“Nuclear Reactor Analysis,” J.J. Duderstadt, L. J. Hamilton, 1 ed., John Wiley & Sons 1976.

16. APPENDIX I: DECAY HEAT CALCULATION

Decay Heat Subroutine for Prismatic HTR Benchmark

By Sonat Sen (July 20, 2010)
Idaho National Laboratories

The decay heat calculations are based on the DIN 25485 standards, which is extracted from PEBBED/THERMIX code. The DIN 25485 standards are applicable to HTRs using pebble-type fuel. *The applicability to HTRs with prismatic fuel is not known*.

The evaluation accounts for:
a) the contribution of the fission products
b) the contribution of the heavy metals
c) the contribution of further isotopes generated by neutron capture of fission products

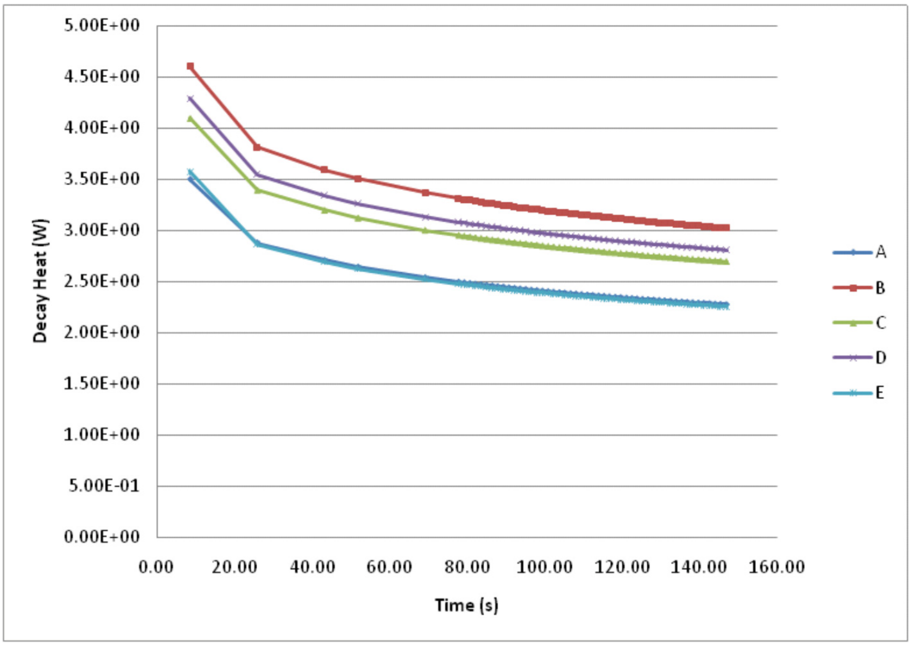
The evaluation includes the local generation of decay heat power as a function of the thermal power history of the fuel located at the considered position as a function of the time elapsed after reactor shutdown or the fuel was discharged. The activation of structural material is ***not*** accounted for.

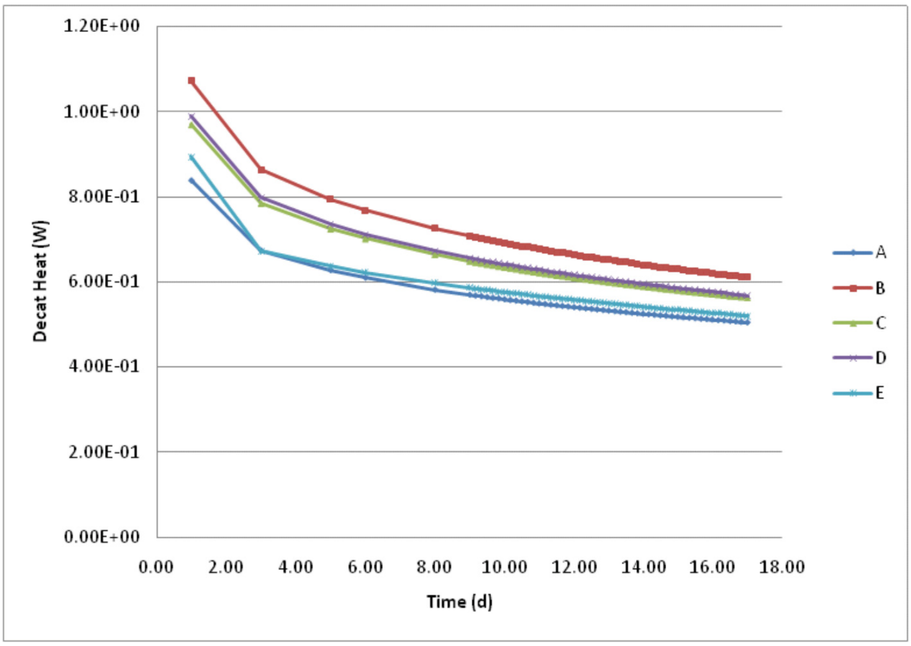
The accuracy of the calculation depends on the detail applied to the definition of the power histogram. The power histogram defines the irradiation history of the considered fuel elements. It divides the operation period of the fuel into time periods, which can be assumed to have constant power, constant sharing of total power between the different fissile isotopes, and as well as a constant neutron capture rate of the fertile materials. If the sharing of the power between the different fissile isotopes is not known to the user, the total power may be assigned to ^{235}U.

Power histogram contains the following data:
1) Length of small time steps
2) Total power in zone
3) Power generated by U-233
4) Power generated by U-235
5) Power generated by U-238
6) Power generated by Pu-239
7) Power generated by Pu-241
8) Ratio of capture rate in Th-232 to total fission rate
9) Ratio of capture rate in U-238 to total fission rate

The limitations of the evaluation of activation effects are:
1) Discharge Burnup of the fuel: 60 MWd/kg ≤ BU ≤ 140 MWd/kg
2) Average power density of the core: 2.5 MW/m^3 ≤ q ≤ 7.0 MW/m^3
3) Average heavy metal load of fuel elements: 4 g/pebble ≤ HM ≤ 12g/pebble

A test calculation for pebbles having 9gr/HM loading with 9.6% enriched and having a discharge burnup of 91 GWD/tHM is performed. 5 different power histograms are considered. The resulting decay heat profiles are represented in the figures below. The power histograms used for the analysis are given at the end of the document.





POWER HISTOGRAM:

A

90.43	3.12E-01	0	0.9803	0.0021	0.0176	0.0001	0	0.2619

85.29	1.33E+01	0	0.9535	0.002	0.0439	0.0006	0	0.2657
80.44	2.29E+02	0	0.9555	0.0022	0.0415	0.0008	0	0.2783
75.86	4.46E+03	0	0.9367	0.0024	0.0596	0.0014	0	0.3024
71.55	2.85E+04	0	0.9273	0.0025	0.0686	0.0017	0	0.3187
67.48	9.20E+04	0	0.9153	0.0026	0.0798	0.0022	0	0.3312
63.65	2.25E+05	0	0.9042	0.0028	0.0901	0.003	0	0.3458
60.03	3.37E+05	0	0.8781	0.0029	0.1138	0.0052	0	0.3588
56.62	3.65E+05	0	0.8403	0.003	0.1466	0.0101	0	0.3715
53.4	3.24E+05	0	0.7869	0.0031	0.1903	0.0197	0	0.3823
50.36	3.48E+05	0	0.7605	0.0033	0.2093	0.0269	0	0.3988
47.5	3.22E+05	0	0.7233	0.0035	0.2353	0.0379	0	0.4168
44.8	2.66E+05	0	0.6875	0.0036	0.255	0.0539	0	0.4246
42.25	3.08E+05	0	0.6792	0.0039	0.2571	0.0598	0	0.446
39.85	3.26E+05	0	0.6503	0.0042	0.278	0.0675	0	0.4673
37.58	2.49E+05	0	0.6195	0.0041	0.2936	0.0828	0	0.4686
35.45	1.95E+05	0	0.5998	0.0043	0.2987	0.0972	0	0.47
33.43	3.11E+05	0	0.6034	0.0047	0.2956	0.0963	0	0.5106
31.53	3.32E+05	0	0.5731	0.005	0.3195	0.1024	0	0.5306
29.74	2.18E+05	0	0.5465	0.0048	0.3358	0.1129	0	0.5225
28.05	1.61E+05	0	0.5335	0.0047	0.3371	0.1247	0	0.4939
26.45	1.42E+05	0	0.5367	0.005	0.3248	0.1335	0	0.5069
24.95	3.09E+05	0	0.5376	0.0056	0.3272	0.1296	0	0.5733
23.53	3.70E+05	0	0.5068	0.0057	0.3533	0.1342	0	0.5937
22.19	2.72E+05	0	0.4804	0.0058	0.3741	0.1397	0	0.6028
20.93	1.73E+05	0	0.4652	0.0058	0.3857	0.1433	0	0.6067
19.74	1.04E+05	0	0.4566	0.0058	0.3918	0.1458	0	0.6072
18.62	5.73E+04	0	0.4521	0.0055	0.3949	0.1475	0	0.5576
17.56	1.16E+05	0	0.4969	0.0045	0.3456	0.153	0	0.4128
16.56	3.01E+05	0	0.4874	0.0049	0.3479	0.1598	0	0.4538
15.62	4.25E+05	0	0.472	0.005	0.3549	0.1681	0	0.4802
14.73	4.67E+05	0	0.4554	0.0051	0.3621	0.1774	0	0.5062
13.9	4.40E+05	0	0.4431	0.0051	0.3679	0.1839	0	0.5234
13.11	3.87E+05	0	0.4319	0.0052	0.3739	0.189	0	0.5373
12.36	3.27E+05	0	0.4225	0.0053	0.3793	0.1929	0	0.5488
11.66	2.69E+05	0	0.4147	0.0054	0.3841	0.1958	0	0.5575
11	2.21E+05	0	0.4092	0.0054	0.3877	0.1977	0	0.562
10.37	1.83E+05	0	0.405	0.0054	0.3905	0.1991	0	0.5656
9.781	1.50E+05	0	0.4015	0.0054	0.3929	0.2002	0	0.5693
9.225	1.25E+05	0	0.399	0.0055	0.3946	0.2009	0	0.5715
8.7	1.05E+05	0	0.3972	0.0055	0.3959	0.2014	0	0.5772
8.206	8.85E+04	0	0.3958	0.0057	0.3968	0.2017	0	0.5883
7.739	7.57E+04	0	0.3948	0.0058	0.3976	0.2018	0	0.6034
7.299	6.14E+04	0	0.3934	0.006	0.3987	0.2019	0	0.6146
6.884	5.79E+04	0	0.3928	0.0059	0.3992	0.2021	0	0.6079
6.493	5.46E+04	0	0.3922	0.0058	0.3998	0.2022	0	0.6006
6.124	4.69E+04	0	0.3914	0.0054	0.4007	0.2025	0	0.5701

5.776	4.54E+04	0	0.3912	0.0054	0.4008	0.2026	0	0.5633
5.447	3.41E+04	0	0.3905	0.0049	0.4017	0.2029	0	0.5033
B								
85.29	1.79E+00	0	0.9803	0.0021	0.0176	0.0001	0	0.2619
80.44	6.83E+01	0	0.9533	0.0021	0.044	0.0006	0	0.2658
75.86	1.09E+03	0	0.9555	0.0022	0.0415	0.0008	0	0.2783
71.55	1.65E+04	0	0.9385	0.0024	0.0578	0.0013	0	0.2999
67.48	1.09E+05	0	0.9282	0.0025	0.0677	0.0016	0	0.3166
63.65	3.63E+05	0	0.9178	0.0026	0.0775	0.0021	0	0.3294
60.03	8.76E+05	0	0.9058	0.0027	0.0887	0.0028	0	0.3431
56.62	1.43E+06	0	0.887	0.0028	0.1058	0.0044	0	0.3551
53.4	1.66E+06	0	0.8606	0.003	0.1291	0.0073	0	0.3662
50.36	1.60E+06	0	0.8104	0.0031	0.172	0.0145	0	0.3773
47.5	1.43E+06	0	0.7656	0.0033	0.206	0.0251	0	0.3901
44.8	1.60E+06	0	0.7496	0.0034	0.2172	0.0298	0	0.4062
42.25	1.43E+06	0	0.7142	0.0035	0.2405	0.0418	0	0.419
39.85	1.16E+06	0	0.6824	0.0037	0.2575	0.0564	0	0.4256
37.58	1.46E+06	0	0.679	0.004	0.2571	0.0599	0	0.4487
35.45	1.51E+06	0	0.6514	0.0041	0.2774	0.0671	0	0.467
33.43	1.18E+06	0	0.6235	0.0042	0.2919	0.0804	0	0.4696
31.53	9.27E+05	0	0.6002	0.0043	0.301	0.0945	0	0.4652
29.74	1.12E+06	0	0.6069	0.0046	0.2917	0.0968	0	0.4964
28.05	1.65E+06	0	0.5926	0.0049	0.3046	0.0979	0	0.522
26.45	1.29E+06	0	0.5596	0.005	0.3291	0.1063	0	0.5327
24.95	9.57E+05	0	0.5444	0.0049	0.3363	0.1144	0	0.5183
23.53	7.24E+05	0	0.5329	0.0046	0.3371	0.1254	0	0.4927
22.19	5.71E+05	0	0.5307	0.0047	0.3297	0.1349	0	0.4864
20.93	1.17E+06	0	0.5449	0.0055	0.3202	0.1294	0	0.5628
19.74	1.71E+06	0	0.5233	0.0057	0.3397	0.1313	0	0.5852
18.62	1.60E+06	0	0.4967	0.0057	0.3614	0.1362	0	0.5979
17.56	1.17E+06	0	0.4769	0.0057	0.3769	0.1405	0	0.6038
16.56	8.21E+05	0	0.4659	0.0058	0.3852	0.1431	0	0.6066
15.62	5.48E+05	0	0.4584	0.0058	0.3906	0.1452	0	0.6078
14.73	3.68E+05	0	0.4545	0.0057	0.3932	0.1466	0	0.6036
13.9	2.12E+05	0	0.4512	0.0052	0.3958	0.1478	0	0.5252
13.11	4.47E+05	0	0.4992	0.0064	0.3435	0.1509	0	0.5807
12.36	1.08E+06	0	0.4884	0.0066	0.3513	0.1537	0	0.6283
11.66	1.68E+06	0	0.466	0.0067	0.3698	0.1575	0	0.6452
11	1.85E+06	0	0.4469	0.0066	0.3858	0.1607	0	0.6544
10.37	1.79E+06	0	0.4308	0.0067	0.399	0.1635	0	0.6608
9.781	1.56E+06	0	0.417	0.0067	0.4104	0.1659	0	0.6633
9.225	1.32E+06	0	0.4072	0.0067	0.4182	0.1679	0	0.6665
8.7	1.11E+06	0	0.4005	0.0066	0.4236	0.1693	0	0.6678
8.206	9.34E+05	0	0.3953	0.0067	0.4277	0.1703	0	0.6682
7.739	7.87E+05	0	0.3911	0.0066	0.4311	0.1712	0	0.6682
7.299	6.12E+05	0	0.3866	0.0066	0.4346	0.1722	0	0.6688

6.884	5.33E+05	0	0.3851	0.0067	0.4357	0.1725		0	0.6684
6.493	4.56E+05	0	0.3832	0.0067	0.4372	0.1729		0	0.668
6.124	3.42E+05	0	0.3814	0.0066	0.4387	0.1733		0	0.6674
5.776	3.21E+05	0	0.381	0.0066	0.4391	0.1733		0	0.6671
5.447	2.07E+05	0	0.3782	0.006	0.4421	0.1737		0	0.5622
C									
85.29	1.24E+00	0	0.9803	0.0021	0.0176	0.0001		0	0.2619
80.44	4.75E+01	0	0.9533	0.0021	0.044	0.0006		0	0.2658
75.86	7.59E+02	0	0.9555	0.0022	0.0415	0.0008		0	0.2783
71.55	1.15E+04	0	0.9385	0.0024	0.0578	0.0013		0	0.2999
67.48	7.57E+04	0	0.9282	0.0025	0.0677	0.0016		0	0.3166
63.65	2.52E+05	0	0.9178	0.0026	0.0775	0.0021		0	0.3294
60.03	6.08E+05	0	0.9058	0.0027	0.0887	0.0028		0	0.3431
56.62	9.92E+05	0	0.887	0.0028	0.1058	0.0044		0	0.3551
53.4	1.15E+06	0	0.8606	0.003	0.1291	0.0073		0	0.3662
50.36	1.11E+06	0	0.8104	0.0031	0.172	0.0145		0	0.3773
47.5	9.95E+05	0	0.7656	0.0033	0.206	0.0251		0	0.3901
44.8	1.11E+06	0	0.7496	0.0034	0.2172	0.0298		0	0.4062
42.25	9.90E+05	0	0.7142	0.0035	0.2405	0.0418		0	0.419
39.85	8.06E+05	0	0.6824	0.0037	0.2575	0.0564		0	0.4256
37.58	1.01E+06	0	0.679	0.004	0.2571	0.0599		0	0.4487
35.45	1.05E+06	0	0.6514	0.0041	0.2774	0.0671		0	0.467
33.43	8.18E+05	0	0.6235	0.0042	0.2919	0.0804		0	0.4696
31.53	6.43E+05	0	0.6002	0.0043	0.301	0.0945		0	0.4652
29.74	7.80E+05	0	0.6069	0.0046	0.2917	0.0968		0	0.4964
28.05	1.15E+06	0	0.5926	0.0049	0.3046	0.0979		0	0.522
26.45	8.96E+05	0	0.5596	0.005	0.3291	0.1063		0	0.5327
24.95	6.64E+05	0	0.5444	0.0049	0.3363	0.1144		0	0.5183
23.53	5.03E+05	0	0.5329	0.0046	0.3371	0.1254		0	0.4927
22.19	3.97E+05	0	0.5307	0.0047	0.3297	0.1349		0	0.4864
20.93	8.10E+05	0	0.5449	0.0055	0.3202	0.1294		0	0.5628
19.74	1.19E+06	0	0.5233	0.0057	0.3397	0.1313		0	0.5852
18.62	1.11E+06	0	0.4967	0.0057	0.3614	0.1362		0	0.5979
17.56	8.10E+05	0	0.4769	0.0057	0.3769	0.1405		0	0.6038
16.56	5.70E+05	0	0.4659	0.0058	0.3852	0.1431		0	0.6066
15.62	3.81E+05	0	0.4584	0.0058	0.3906	0.1452		0	0.6078
14.73	2.56E+05	0	0.4545	0.0057	0.3932	0.1466		0	0.6036
13.9	1.47E+05	0	0.4512	0.0052	0.3958	0.1478		0	0.5252
13.11	2.63E+05	0	0.5009	0.007	0.3419	0.1502		0	0.6797
12.36	6.58E+05	0	0.4896	0.0072	0.3512	0.152		0	0.7265
11.66	1.06E+06	0	0.4648	0.0072	0.3737	0.1543		0	0.7314
11	1.16E+06	0	0.4439	0.0072	0.3929	0.156		0	0.7347
10.37	1.11E+06	0	0.4262	0.0072	0.4091	0.1575		0	0.7372
9.781	9.72E+05	0	0.4114	0.0071	0.4226	0.1589		0	0.7377
9.225	8.04E+05	0	0.4009	0.0072	0.432	0.1599		0	0.7374
8.7	6.72E+05	0	0.3938	0.0071	0.4383	0.1608		0	0.7374

8.206	5.66E+05	0	0.3885	0.0071	0.443	0.1614	0	0.7372
7.739	4.80E+05	0	0.3841	0.0071	0.4468	0.162	0	0.7367
7.299	3.57E+05	0	0.3794	0.0071	0.451	0.1625	0	0.7354
6.884	3.11E+05	0	0.3779	0.0071	0.4523	0.1627	0	0.7346
6.493	2.65E+05	0	0.376	0.0071	0.454	0.1629	0	0.7335
6.124	2.02E+05	0	0.374	0.007	0.4559	0.1631	0	0.7226
5.776	1.90E+05	0	0.3735	0.007	0.4564	0.1631	0	0.7195
5.447	1.28E+05	0	0.3687	0.006	0.462	0.1633	0	0.539
D								
85.29	2.40E+00	0	0.9803	0.0021	0.0176	0.0001	0	0.2619
80.44	9.16E+01	0	0.9533	0.0021	0.044	0.0006	0	0.2658
75.86	1.46E+03	0	0.9555	0.0022	0.0415	0.0008	0	0.2783
71.55	2.21E+04	0	0.9385	0.0024	0.0578	0.0013	0	0.2999
67.48	1.46E+05	0	0.9282	0.0025	0.0677	0.0016	0	0.3166
63.65	4.86E+05	0	0.9178	0.0026	0.0775	0.0021	0	0.3294
60.03	1.17E+06	0	0.9058	0.0027	0.0887	0.0028	0	0.3431
56.62	1.92E+06	0	0.887	0.0028	0.1058	0.0044	0	0.3551
53.4	2.23E+06	0	0.8606	0.003	0.1291	0.0073	0	0.3662
50.36	2.15E+06	0	0.8104	0.0031	0.172	0.0145	0	0.3773
47.5	1.92E+06	0	0.7656	0.0033	0.206	0.0251	0	0.3901
44.8	2.15E+06	0	0.7496	0.0034	0.2172	0.0298	0	0.4062
42.25	1.91E+06	0	0.7142	0.0035	0.2405	0.0418	0	0.419
39.85	1.56E+06	0	0.6824	0.0037	0.2575	0.0564	0	0.4256
37.58	1.96E+06	0	0.679	0.004	0.2571	0.0599	0	0.4487
35.45	2.02E+06	0	0.6514	0.0041	0.2774	0.0671	0	0.467
33.43	1.58E+06	0	0.6235	0.0042	0.2919	0.0804	0	0.4696
31.53	1.24E+06	0	0.6002	0.0043	0.301	0.0945	0	0.4652
29.74	1.51E+06	0	0.6069	0.0046	0.2917	0.0968	0	0.4964
28.05	2.21E+06	0	0.5926	0.0049	0.3046	0.0979	0	0.522
26.45	1.73E+06	0	0.5596	0.005	0.3291	0.1063	0	0.5327
24.95	1.28E+06	0	0.5444	0.0049	0.3363	0.1144	0	0.5183
23.53	9.70E+05	0	0.5329	0.0046	0.3371	0.1254	0	0.4927
22.19	7.65E+05	0	0.5307	0.0047	0.3297	0.1349	0	0.4864
20.93	1.56E+06	0	0.5449	0.0055	0.3202	0.1294	0	0.5628
19.74	2.29E+06	0	0.5233	0.0057	0.3397	0.1313	0	0.5852
18.62	2.14E+06	0	0.4967	0.0057	0.3614	0.1362	0	0.5979
17.56	1.56E+06	0	0.4769	0.0057	0.3769	0.1405	0	0.6038
16.56	1.10E+06	0	0.4659	0.0058	0.3852	0.1431	0	0.6066
15.62	7.34E+05	0	0.4584	0.0058	0.3906	0.1452	0	0.6078
14.73	4.94E+05	0	0.4545	0.0057	0.3932	0.1466	0	0.6036
13.9	2.84E+05	0	0.4512	0.0052	0.3958	0.1478	0	0.5252
13.11	4.31E+05	0	0.5016	0.0067	0.3416	0.1501	0	0.6437
12.36	1.15E+06	0	0.4924	0.0067	0.3487	0.1522	0	0.6589
11.66	2.00E+06	0	0.4703	0.0066	0.3675	0.1556	0	0.6398
11	2.21E+06	0	0.4524	0.0066	0.3824	0.1586	0	0.646
10.37	2.14E+06	0	0.4375	0.0067	0.3947	0.1611	0	0.6514

9.781	1.87E+06	0	0.4249	0.0066	0.4052	0.1633	0	0.6533
9.225	1.58E+06	0	0.4159	0.0066	0.4126	0.1649	0	0.656
8.7	1.33E+06	0	0.4097	0.0066	0.4176	0.1661	0	0.6573
8.206	1.12E+06	0	0.405	0.0066	0.4214	0.167	0	0.6577
7.739	9.38E+05	0	0.4011	0.0065	0.4246	0.1678	0	0.6576
7.299	7.28E+05	0	0.3969	0.0066	0.4279	0.1686	0	0.6581
6.884	6.35E+05	0	0.3955	0.0067	0.429	0.1688	0	0.6579
6.493	5.45E+05	0	0.3938	0.0065	0.4305	0.1692	0	0.6577
6.124	4.09E+05	0	0.392	0.0066	0.432	0.1694	0	0.6571
5.776	3.83E+05	0	0.3915	0.0065	0.4325	0.1695	0	0.6568
5.447	2.59E+05	0	0.3878	0.0059	0.4367	0.1696	0	0.5509
E								
90.43	5.16E-01	0	0.9803	0.0021	0.0176	0.0001	0.26	0
85.29	2.20E+01	0	0.9535	0.002	0.0439	0.0006	0.27	0
80.44	3.79E+02	0	0.9555	0.0022	0.0415	0.0008	0.28	0
75.86	7.38E+03	0	0.9367	0.0024	0.0596	0.0014	0.3	0
71.55	4.72E+04	0	0.9273	0.0025	0.0686	0.0017	0.32	0
67.48	1.52E+05	0	0.9153	0.0026	0.0798	0.0022	0.33	0
63.65	3.72E+05	0	0.9042	0.0028	0.0901	0.003	0.35	0
60.03	5.58E+05	0	0.8781	0.0029	0.1138	0.0052	0.36	0
56.62	6.04E+05	0	0.8403	0.003	0.1466	0.0101	0.37	0
53.4	5.37E+05	0	0.7869	0.0031	0.1903	0.0197	0.38	0
50.36	5.76E+05	0	0.7605	0.0033	0.2093	0.0269	0.4	0
47.5	5.33E+05	0	0.7233	0.0035	0.2353	0.0379	0.42	0
44.8	4.40E+05	0	0.6875	0.0036	0.255	0.0539	0.42	0
42.25	5.10E+05	0	0.6792	0.0039	0.2571	0.0598	0.45	0
39.85	5.39E+05	0	0.6503	0.0042	0.278	0.0675	0.47	0
37.58	4.13E+05	0	0.6195	0.0041	0.2936	0.0828	0.47	0
35.45	3.23E+05	0	0.5998	0.0043	0.2987	0.0972	0.47	0
33.43	5.14E+05	0	0.6034	0.0047	0.2956	0.0963	0.51	0
31.53	5.49E+05	0	0.5731	0.005	0.3195	0.1024	0.53	0
29.74	3.61E+05	0	0.5465	0.0048	0.3358	0.1129	0.52	0
28.05	2.66E+05	0	0.5335	0.0047	0.3371	0.1247	0.49	0
26.45	2.36E+05	0	0.5367	0.005	0.3248	0.1335	0.51	0
24.95	5.12E+05	0	0.5376	0.0056	0.3272	0.1296	0.57	0
23.53	6.13E+05	0	0.5068	0.0057	0.3533	0.1342	0.59	0
22.19	4.51E+05	0	0.4804	0.0058	0.3741	0.1397	0.6	0
20.93	2.86E+05	0	0.4652	0.0058	0.3857	0.1433	0.61	0
19.74	1.72E+05	0	0.4566	0.0058	0.3918	0.1458	0.61	0
18.62	9.48E+04	0	0.4521	0.0055	0.3949	0.1475	0.56	0
17.56	9.66E+04	0	0.4998	0.0053	0.3443	0.1506	0.54	0
16.56	3.02E+05	0	0.4923	0.0051	0.3482	0.1544	0.51	0
15.62	5.41E+05	0	0.4776	0.005	0.3568	0.1606	0.47	0
14.73	6.35E+05	0	0.4633	0.0051	0.363	0.1686	0.49	0
13.9	6.16E+05	0	0.4534	0.0052	0.3667	0.1747	0.5	0
13.11	5.54E+05	0	0.4444	0.0052	0.3707	0.1797	0.51	0

12.36	4.74E+05	0	0.4368	0.0052	0.3745	0.1835	0.52	0
11.66	3.95E+05	0	0.4304	0.0052	0.3779	0.1865	0.53	0
11	3.32E+05	0	0.4258	0.0053	0.3804	0.1885	0.53	0
10.37	2.80E+05	0	0.4221	0.0054	0.3825	0.19	0.54	0
9.781	2.33E+05	0	0.419	0.0053	0.3845	0.1912	0.54	0
9.225	1.94E+05	0	0.4166	0.0054	0.386	0.192	0.54	0
8.7	1.63E+05	0	0.4149	0.0054	0.3872	0.1925	0.55	0
8.206	1.40E+05	0	0.4135	0.0054	0.3882	0.1929	0.55	0
7.739	1.22E+05	0	0.4123	0.0055	0.389	0.1932	0.56	0
7.299	9.93E+04	0	0.4109	0.0057	0.39	0.1934	0.57	0
6.884	9.15E+04	0	0.4104	0.0057	0.3904	0.1935	0.58	0
6.493	8.39E+04	0	0.4097	0.0059	0.3908	0.1936	0.59	0
6.124	7.34E+04	0	0.4092	0.0055	0.3914	0.1939	0.56	0
5.776	7.14E+04	0	0.409	0.0055	0.3915	0.194	0.55	0
5.447	5.73E+04	0	0.4094	0.0047	0.3912	0.1947	0.46	0

17. APPENDIX II: CALCULATION OF THE THERMAL CONDUCTIVITY

AMEC Model for Radial Effective Thermal Conductivity of Fuel Blocks

The model is based on Maxwell's theory of the conductivity of composite materials. The original theory is derived for two materials, but it is extended to three materials for the HTR applications by AMEC/NSS []

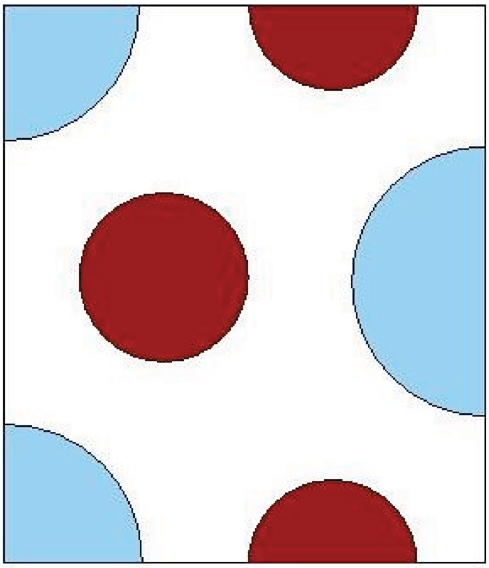
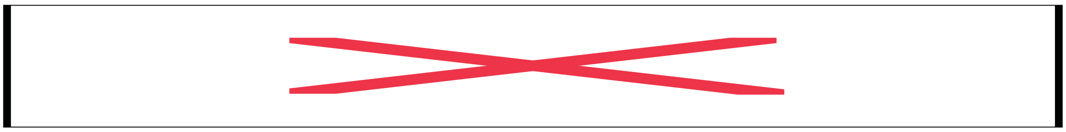


Figure 28: Unit cell of MHGTR fuel block

The effective conductivity of a unit cell shown in Figure 28 is given with the following expression.



where ,

k_{eff}	=	effective thermal conductivity
k_s	=	thermal conductivity of the solid material
k_{por}	=	thermal conductivity of the pore material
k_{FC}	=	thermal conductivity of the fuel compact
α_1	=	volume fraction of gap material
α_2=	volume fraction of fuel compacts	

18. APPENDIX III: SIMPLIFIED CROSS SECTION SET SPECIFICATIONS

The simplified cross section set follows the

MATERIAL #

Group #	Scalar Flux	Σ_T	D	$\nu\Sigma_f$	χ

Scattering Profile (g = 1 – ngroup columns)

Start Group ID   End Group ID

P0 Scattering Matrix (g = 1 – ngroup columns)

$\Sigma_{g\text{->}g'}$ (g' = Starting Group ID, End Group ID)

/MATERIAL #

EXAMPLE:

MATERIAL 1

```
   1     3.58035E-01   8.47674E-02   6.19258E+00   6.90407E-04   1.93826E-04   1.60238E-02
   2     3.63022E+00   1.29507E-01   4.12089E+00   3.46316E-04   1.15298E-04   1.56004E-01
   3     3.95931E+01   1.67893E-01   2.33150E+00   1.45391E-04   5.51244E-05   6.74210E-01
   4     3.97063E+01   2.80867E-01   1.30405E+00   5.30253E-05   2.15953E-05   1.41962E-01
   5     3.16133E+01   3.24262E-01   1.09423E+00   7.55829E-05   3.11679E-05   1.07022E-02
   6     3.42833E+01   3.33706E-01   1.05831E+00   1.35402E-04   5.56597E-05   1.02248E-03
   7     1.48419E+01   3.36593E-01   1.04815E+00   2.33543E-04   9.59776E-05   5.07248E-05
   8     1.44365E+01   3.36986E-01   1.04656E+00   3.46232E-04   1.42290E-04   1.60332E-05
   9     1.68575E+01   3.36968E-01   1.04629E+00   5.84350E-04   2.40147E-04   5.84031E-06
  10     1.41736E+01   3.39672E-01   1.03673E+00   8.07851E-04   3.31998E-04   1.52720E-06
  11     1.23947E+01   3.41096E-01   1.03051E+00   1.10442E-03   4.53878E-04   5.04733E-07
  12     1.05766E+01   3.43719E-01   1.02161E+00   1.70935E-03   7.02481E-04   1.87770E-07
  13     1.08476E+01   3.42918E-01   1.03192E+00   1.76131E-03   7.23835E-04   9.06619E-08
  14     7.21005E+00   3.38015E-01   1.04020E+00   2.83076E-03   1.16334E-03   2.71380E-08
  15     6.92379E+00   3.49343E-01   9.98276E-01   7.49090E-04   3.07849E-04   1.28989E-08
  16     1.04157E+01   3.36215E-01   1.05418E+00   6.95873E-04   2.85979E-04   9.96776E-09
  17     8.97560E+00   3.36741E-01   1.04463E+00   6.63248E-04   2.72458E-04   4.13045E-09
  18     1.22014E+01   3.38148E-01   1.03841E+00   2.53169E-03   1.03924E-03   2.39520E-09
  19     2.21957E+01   3.39996E-01   1.02854E+00   4.30883E-03   1.76867E-03   1.14767E-09
  20     2.76384E+01   3.43207E-01   1.01136E+00   7.10354E-03   2.91583E-03   5.16419E-10
  21     2.23942E+01   3.46031E-01   9.96345E-01   7.56553E-03   3.10546E-03   2.67791E-10
  22     1.52552E+01   3.54061E-01   9.57541E-01   9.89203E-03   4.06043E-03   1.66375E-10
  23     9.00096E+00   3.66530E-01   8.97767E-01   1.32131E-02   5.42366E-03   1.11661E-10
  24     6.37390E+00   3.84827E-01   8.22483E-01   1.94927E-02   8.00125E-03   1.24179E-10
  25     6.15849E-01   3.60003E-01   9.09646E-01   3.30833E-02   1.35799E-02   2.93871E-11
  26     1.92771E-01   4.05119E-01   7.25440E-01   5.28391E-02   2.16891E-02   2.73242E-11
Scattering Profile
      1 10
      2 11
      3 10
      4  7
      5  9
      6  8
      7  9
      8 10
      9 11
     10 12
     11 13
     12 14
     13 16
     14 16
     15 17
     15 24
     16 25
     16 26
     17 26
     17 26
     18 26
     18 26
     18 26
     18 26
     18 26
     18 26
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