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HTGR REACTOR PHYSICS, THERMAL-HYDRAULICS AND DEPLETION UNCERTAINTY ANALYSIS: A PROPOSED IAEA COORDINATED RESEARCH PROJECT

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ABSTRACT

The continued development of High Temperature Gas Cooled Reactors (HTGRs) requires verification of HTGR design and safety features with reliable high fidelity physics models and robust, efficient, and accurate codes. The predictive capability of coupled neutronics/thermal-hydraulics and depletion simulations for reactor design and safety analysis can be assessed with sensitivity analysis and uncertainty analysis methods. In order to benefit from recent advances in modeling and simulation and the availability of new covariance data (nuclear data uncertainties) extensive sensitivity and uncertainty studies are needed for quantification of the impact of different sources of uncertainties on the design and safety parameters of HTGRs. Uncertainty and sensitivity studies are an essential component of any significant effort in data and simulation improvement.
In February 2009, the Technical Working Group on Gas-Cooled Reactors recommended that the proposed IAEA Coordinated Research Project (CRP) on the HTGR Uncertainty Analysis in Modeling be implemented. In the paper the current status and plan are presented. The CRP will also benefit from interactions with the currently ongoing OECD/NEA Light Water Reactor (LWR) UAM benchmark activity by taking into consideration the peculiarities of HTGR designs and simulation requirements

Key Words: uncertainty analysis, sensitivity analysis, high temperature reactors, PBMR, GT-MHR, pebble-bed, prismatic.

1. INTRODUCTION

The continued development of High Temperature Gas Cooled Reactors (HTGRs) requires verification of HTGR design and safety features with reliable high fidelity physics models and robust, efficient, and accurate codes.

The predictive capability of coupled neutronics/thermal-hydraulics and depletion simulations for reactor design and safety analysis can be assessed with sensitivity analysis (SA) and uncertainty analysis (UA) methods. Uncertainty originates from errors in physical data, manufacturing uncertainties, modeling and computational algorithms. SA is helpful for ranking the various sources of uncertainty and error in the results of core analyses. SA and UA are required to address cost, safety, and licensing needs and should be applied to all aspects of reactor multi-physics simulation. SA and UA can guide experimental, modeling, and algorithm research and development. Current SA and UA rely either on derivative-based methods such as stochastic sampling methods or on generalized perturbation theory to obtain sensitivity coefficients. Neither approach addresses all needs.

In order to benefit from recent advances in modeling and simulation and the availability of new covariance data (nuclear data uncertainties) extensive sensitivity and uncertainty studies are needed for quantification of the impact of different sources of uncertainties on the design and safety parameters of HTGRs. Only a parallel effort in advanced simulation and in nuclear data improvement will be able to provide designers with more robust and well validated calculation tools to meet design target accuracies.

Uncertainty and sensitivity studies are an essential component of any significant effort in data and simulation improvement. These studies also can be credible and can be used in a convincing and effective manner to perform design optimization and to assess safety features and design margins, but only if the uncertainty information is of the highest quality, and reliability and science-based.

In February 2009, the Technical Working Group on Gas-Cooled Reactors (TWG-GCR) recommended that the proposed IAEA Coordinated Research Program (CRP) on the HTGR Uncertainty Analysis in Modeling (UAM) be implemented.

This CRP is viewed as a natural and logical continuation of the previous IAEA and NEA/OECD international activities on Verification and Validation (V&V) of available analytical capabilities for HTGR simulation for design and safety evaluations [1], [2]. Within the framework of these activities different numerical and experimental benchmark problems were performed and insight was gained about specific physics phenomena and the adequacy of analysis methods.

The CRP will also benefit from interactions with the currently ongoing OECD/NEA Light Water Reactor (LWR) UAM benchmark activity by taking into consideration the peculiarities of HTGR designs and simulation requirements [3].

In the next section the main objectives of the CRP on the HTGR UAM are introduced. In Section III the current efforts in the OECD/NEA LWR UAM benchmark will be summarized.

Sections IV and V describe the design configurations to be analyzed in the CRP with the different phases and exercises defined. The conclusions and future work, including the next steps to get the CRP officially off the ground, complete this paper.

2. OBJECTIVES OF THE COORDINATED RESEARCH PROJECT (CRP).

Sensitivity Analysis (SA) and Uncertainty Analysis (UA) capabilities must be further developed for comprehensive coupled (multi-physics and multi-scale) simulations with nonlinear feedback mechanisms. SA and UA methods need to be considered as an integral part of the development of coupled code methods. Of particular importance are innovative methods that address nonlinearity, can predict the probability distributions in output parameters, can treat discrete events, and handle simultaneously large input data and response fields in a computationally efficient manner.

In the proposed comprehensive IAEA CRP on the HTGR UAM different SA and UA methods will be compared, further developed and their value assessed including the validation of the methodologies for uncertainty propagation in HTGR modeling. The uncertainty propagation will be estimated through the whole simulation process on a unified benchmark framework to provide credible coupled code predictions with defensible uncertainty estimations of safety margins at the full core/system level. The proposed program will help to utilize the created community of experts during the previous IAEA and OECD HTGR-related activities and expand it by mixing expertise in physics (neutronics and thermal-hydraulics) and in SA and UA. The CRP will allow not only to compare and to assess the current SA and UA methods on representative applications but also will stimulate further development of efficient and powerful SA and UA methods suitable for complex coupled code simulations and will help to formulate recommendations and guidelines on how to utilize advanced and optimized SA and UA methods in “best estimate” reactor simulations in HTGR licensing practices.

The objective is to determine the uncertainty in HTGR calculations at all stages of coupled reactor physics/thermal hydraulics and depletion calculations. In order to accomplish this objective a benchmark platform for uncertainty analysis in best-estimate coupled code calculations for design and safety analysis of HTGRs will be defined and utilized. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) will be tested on a number of benchmark exercises with maximum utilization of the available experimental data, published benchmark results and released design details. Two main HTGR types are selected and will be followed, based on previous benchmark experiences and available data – the prismatic (block) and the pebble bed HTGRs.

The comparative analysis results of the completed OECD PBMR-400 Coupled Code Benchmark have demonstrated the need of such an HTGR UAM program [4]. Figure 1 shows a comparison of different computer code predictions of core maximum fuel temperature for Depressurized Loss of Forced Cooling (DLOFC) without scram transient scenario. The deviations between participants’ results range between 50°C to 80°C during the transient having in mind that for this benchmark all participants use the same cross-section library i.e. without taking into account cross-section uncertainties.

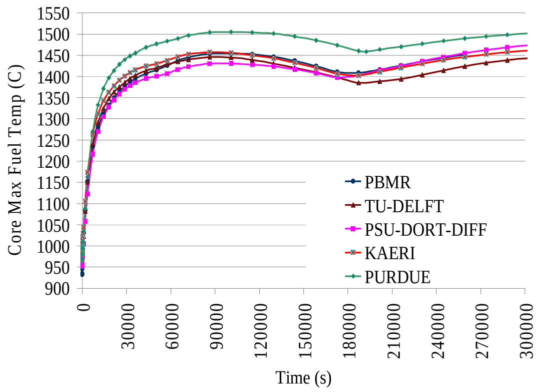

Figure 1. Core maximum fuel temperature vs. time.

Such an effort can be undertaken within the framework of a program of international co-operation that would benefit from the coordination of the IAEA and by interfacing with the NEA/OECD activities as well as with international and national activities on Generation IV Very High Temperature Reactor (VHTR).

3. METHODOLOGY.

The proposed technical approach is to establish and utilize a benchmark for uncertainty analysis in best-estimate coupled HTGR modeling and analysis, using as a basis a series of well defined problems with complete sets of input specifications and reference experimental data.

The principal idea is to subdivide the coupled system calculation into several steps, each of which can contribute to the total uncertainty and to identify input, output, and assumptions for each step. The resulting uncertainty in each step will be calculated by taking into account all sources of uncertainties including propagating the related uncertainties from previous steps. As part of this effort, the development and assessment of different methods or techniques to account for the uncertainties in the calculations will be investigated and reported.

The above-described approach is based on the introduction of three phases, which allows for developing a benchmark framework which mixes information from the available integral facility and Nuclear Power Plant experimental data with analytical and numerical benchmarking. Such an approach compares and assesses current and new uncertainty methods on representative applications and simultaneously benefits from different methodologies to arrive at recommendations and guidelines. Each phase will contain two Exercises which will be performed in parallel. The three phases are as follows:

Phase I (Stand-alone Modeling):

a) Exercise 1 (I-1): “Stand-alone Neutronics” focused on the derivation of the multi-group microscopic cross-section libraries, followed by the derivation of the few-group macroscopic cross-section libraries and core steady state stand-alone neutronics calculations

b) Exercise 2 (I-2): “Stand-alone Thermal-hydraulics” focused on core and system thermal-hydraulic modeling.

Phase II (Design Calculations):

a) Exercise II-1: “Coupled Steady-State” focused on coupled steady-state neutronics/thermal-hydraulics core performance

b) Exercise II-2: “Coupled Depletion” focused on coupled depletion neutronics/thermal-hydraulics core performance

Phase III (Safety Calculations):

a) Exercise III-1: “Coupled Core Transient” focused on coupled neutronics/thermal-hydraulics core transient performance with boundary conditions

b) Exercise III-3: “Coupled System Transient” focused on coupled core/thermal-hydraulic system transient performance

4. DEFINITION OF THE REPRESENTATIVE REACTOR DESINGS.

It was agreed that the CRP will follow the LWR Uncertainty Analysis benchmark as close as possible. For test-cases, the PBMR 400 MW benchmark definition will also be used as a basis for the specification of the CRP test cases. The following designs were selected:

- The MHTGR (an earlier General Atomics 350MWth design considered for NGNP) was adopted as the main prismatic reference design.
- The HTR-Module-based design, upgraded to 250MWth will be the reference pebble-bed design with some simplifications introduced.
- The GT-MHR design will be adopted as the second reference prismatic design for the CRP with the provision that more than two parties express interest and will make contributions to this configuration.

In each case the designs will start with detailed Uranium fuel specification but this will be complemented with a plutonium fuel definition to include especially the cross section uncertainties of both fuel cycles.

4.1 The 350MWth MHTGR Design

The MHTGR-350 MW is a General Atomics (GA) design that has existed since the 1980’s. 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 [5]. The pressure-retaining components are constructed of steel and designed using existing technology. Currently the Prismatic Core Transient Benchmark Working Group (PCTBWG), consisting of different organizations in USA, is developing coupled code prismatic HTR transient benchmark specifications [6]. The purpose of the collaboration is to generate data and documentation to help benchmark computer codes for use in the analysis of prismatic high temperature reactors. The benchmark document will be submitted to the Organization for Economic Co-operation and Development (OECD) - Nuclear Energy Agency (NEA) for sponsorship. The reference design is based on the 350 MWth MHTGR. The HTGR UAM benchmark activity will utilize these core design specifications being developed by the PCTBWG.

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 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 plan view is shown in Figure 2. The active core consists of hexagonal graphite fuel elements (see Figure 3) 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.

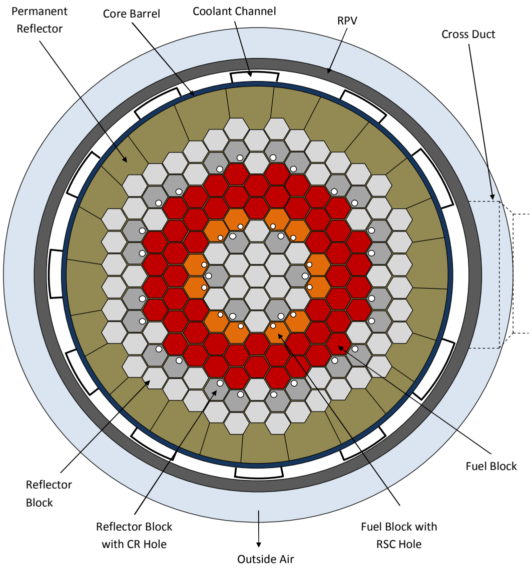

Figure 2: MHTGR Reactor Unit Layout – Plane

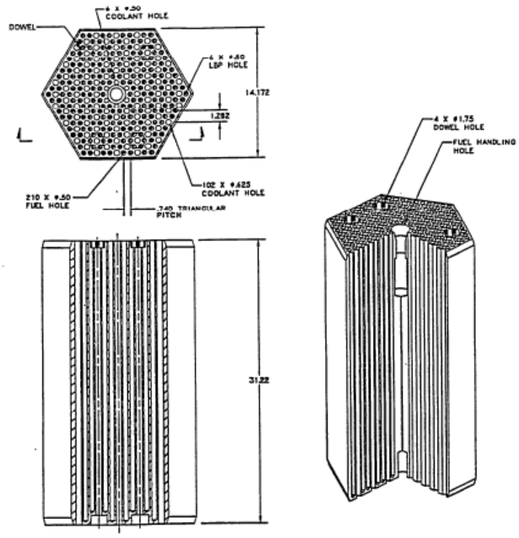

Figure 3: Hexagonal fuel element

4.2 The 250MWth Pebble Bed Design

The HTR-Modul core design, from the Interatom Company in Germany in the late 1980's, represented the first modular HTR pebble bed reactor design with inherent safety characteristics **Error! Reference source not found.**. The HTR-Modul power plant consists of standardized reactor units with a power rating of 200 MWth each, in order to maintain the inherent safety features of the small high-temperature reactors. The power conversion is via a helical tube steam generator with the helium coolant being circulated by the primary circuit blower.

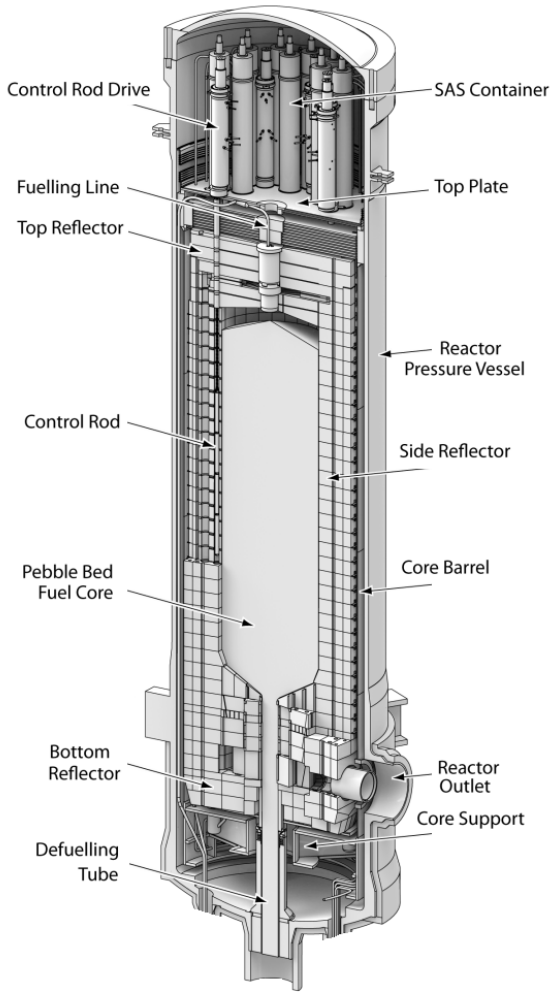

Figure 4: Core geometrical representation for the PBR250 design

The HTR-Modul design was selected as the basis of the 250 MWt Pebble Bed Reactor design (PBR-250) defined for this CRP. The only major changes are the increase of the power and the axial height. This is in broad terms the same changes made by the HTR-PM designers in China **Error! Reference source not found.** and the PBMR Company after the change in project strategy **Error! Reference source not found.**, including the design proposed for the NGNP project

The geometry of PBR250 and its major design parameters are shown in Figure 4 and Table 1, respectively. The specifications of a TRISO particle fuel spheres are also listed in Table 1 and are essentially same to those of the HTR-Module. In the CRP specification the need for other fuel types (Plutonium, MOX, Thorium, etc.), higher temperatures (VHTR applications) or parameters that highlight other sensitivities or uncertainties will be added to the PBR250 specification.

The model approximations and final specification will be finalized at the next planning meeting of the activities of the CRP and will include the final specification of the core height, inclusion of top and bottom cones and representative cell calculations (infinite or mini-core to include the environment). The details of the control and shutdown system (number, position and modeling method) must still be decided.

Table I: Major design parameters.

Reactor Design Specifications		
Parameter	**Value**	**Unit**
Thermal power	250	MW
Fuel core diameter	300	cm
Core height	~ 10-10.5	m
Side reflector thickness	100	cm
Inlet temperature	250	^{0}C
Outlet temperature	750	^{0}C
Pressure	60	bar
Mass flow	96	kg/s
Fuel: Pebble type		
Pebble radius	3.0	cm
Thickness of fuel free zone	0.5	cm
Density of graphite in matrix/fuel free zone	1.74	g/cm^3
U-235 enrichment of uranium	8.0	wt%
HM mass	7.0	g
Coated particles:		
Kernel diameter	500	μm
Kernel density	10.4	g/cm^3
Coating material	C / C / SiC / C	
Layer thickness	95 / 40 / 35 / 40	μm
Layer densities	1.05 / 1.90 / 3.18 / 1.90	g/cm^3

4.3. The GT-MHR Design

General Atomics in the USA and Experimental Design Bureau of Machine Building (OKBM) in the Russian Federation have jointly developed a gas turbine modular helium reactor (GT-MHR) [10].

The 600 MW(t) reactor is cooled by helium at a pressure of 7 MPa. The reactor core consists of five rings of inner reflector to five rings and three rings of fuel. The amount of fuel columns in the active core region is to 102 columns using the prismatic block design with 1020 fuel elements –see Figure 5.

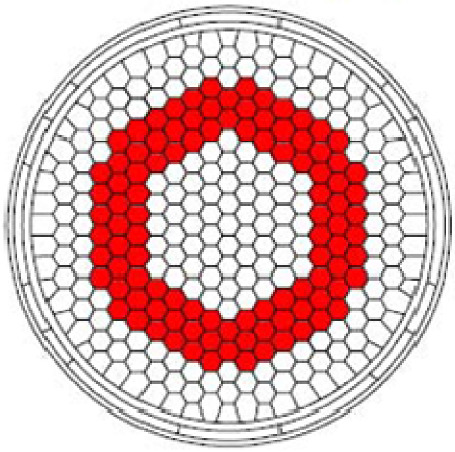

Figure 5: GT-MHR core layout

5. UNCERTAINTY BENCHMARK EXERCISES

The work is divided in several phases as described below.

5.1. Phase I: Stand-Alone Physics Phenomena Modeling

The benchmark specification for Phase I is being developed to include the following exercises.

5.1.1. Exercise 1 (I-1): "Stand-alone Neutronics"

"Stand-alone Neutronics" is focused on the derivation of the multi-group microscopic cross-section libraries, followed by the derivation of the few-group macroscopic cross-section libraries and core steady state stand-alone neutronics calculations.

5.1.1.1. Exercise 1a (I-1a) – Cell Physics: Derivation of the multi-group microscopic cross-section libraries.

The major Uncertainty (U) parameter to be propagated is U-1 (multi-group cross-section variance / covariance matrix). The following items will be addressed in the benchmark specifications:

- Define unit cells for PBR and PMR
- Use SCALE 44 group data as source of co-variance data (INPUT) and expand (up to 4 times using ANGELO and LAMDA (for checking)) to group structure of interest used [3].
- Can use different evaluated data sets as the basis for evaluation.
- New updated co-variance in future based on ENDBF-VII.

Unit cell is only the basis to compare the cross section uncertainties. The definition of the unit cell has to be derived (geometry and number densities provided). Participants can make their

own approximations as needed in the analysis. This unit cell is not the basis for the next assembly calculation. For pebble bed case, a 1-D unit cell or pebble can be used (refer to MCNP detailed cell vs. simplified cell yielding good comparison in University of Michigan slides). For prismatic, 2-D core (see GT-MHR CRP5 and presentation of Boyarinov or the triangular lattice cell that include 3 pins / coolant channel) can be used. In all cases equilibrium fuel specification must be used. MICROX (1-D) or SCALE6.0 can be used as reference calculation.
The requested output includes k-inf, fission and capture rates in U235, and U238. Comparisons between different methods (with defined 1-group spectrum uncertainties) will be carried out.
The target is to expand SCALE 6.0 44-group including resonance self-shielding, to own multi-group covariance data for two defined for two different spectra and two different temperatures.

5.1.1.2. Exercise 1b (I-1b) – Lattice Physics: Derivation of the few-group macroscopic cross-section libraries

The major Uncertainty (U) parameter to be propagated is U-2 (few-group cross-section variance / covariance matrix). The input uncertainty parameters include covariance data; manufacturing variations, environment effect / leakage (modeling and numerical uncertainties later), kinetics parameters.

The requested output includes uncertainties in k inf/k-eff, reaction rates (homogenized over region), effective 1-group cross section, and kinetics parameters.

The target is to obtain few group parameters, variance / co-variance matrix for all homogenized including diffusion coefficients, Assembly Discontinuity Factors (ADFs) and kinetic parameters.

One important issue in this exercise is the definition of a lattice.

For pebble reactor design the following considerations will be taken into account:

- 1D cut (slab or cylindrical) through the core – infinite height (random packing) with reflectors (mini-core)
- Different constant packing ratio and / or temperatures / grey curtain to simulate control rods as variation per case (sensitivity study to show how spectrum change to be used as input to decide the cases)
- One group effective cross sections for two fuel temperatures (250C and 750C)
- Uranium fuel and plutonium fuel.

For the prismatic reactor designs the following cases will be considered:

- Single assembly with symmetry 1/6th)
- Single assembly with symmetry including BP (1/6th)
- Color-set with control rods / non symmetric
- Reflector
- (plutonium case – one of the above)

5.1.1.3. Exercise 1c (I-1c) – Core Physics: Criticality (steady state) stand-alone neutronics calculations

The major Uncertainty (U) parameter to be propagated is U-3 (uncertainties in k-eff, power peaking factors, rod worth). First 2-D calculations will be performed before 3-D calculations. The input uncertainty parameters include few group cross-section and kinetics parameters variance / covariance matrices, modeling and numerical uncertainties.

The request output and target of the analysis of this exercise are k-eff, reaction rates, power distribution uncertainties, and sensitivity analysis (what contributes to the uncertainties).

One important issue in this exercise is the definition of the core.

For the pebble bed reactor design the following considerations will be taken into account:

- Single-pebble core with a mixture of uranium and plutonium to simulate equilibrium core and a realistic spectrum.
- Calculate fresh and burned core
- HZP and HFP temperatures (250C and 750C)
- Include all fission products of interest.
- Specify control rods position
- Include uncertainties on burnup distribution due to different pebble flow rates or do a sensitivity study.

For the prismatic reactor designs the following cases will be considered:

- 250C (523K) uniform;
- Hot case with reflector 900K, core 1200K;
- Startup core with one fuel type.

5.1.1.4. Exercise 1d (I-1d) – Core Physics: stand-alone kinetics without feedback

This exercise models time dependent neutronics / kinetics without thermal feedback with an objective to identify uncertainties in kinetics parameters, delayed neutron fractions, delayed and prompt neutron fractions, inverse velocities, etc.

Starting from Exercise 1c (hot case) and using 3D core model, control rod insertion and withdrawal simulations will be performed. The input uncertainty parameters include cross sections and kinetics parameters, solution method, covariance tables

The requested output and target of the analysis of this exercise are power evolution and power distribution. Simulated rod movements will be limited in extent and speed.

5.1.2. Exercise 1 (I-2): "Stand-alone Thermal-Hydraulics"

5.1.2.1. Exercise I-2a: "Stand-alone Thermal-hydraulics" focused on core and system thermal-hydraulic modeling (Normal operation)

When specifying the input uncertainty parameters the following recommendations will be taken into account:

- Bypass flow is recommended to be specified ("Bypass flows": Wall effects, side reflector blocks, side reflector and barrel, cooling of discharge fuel, control rod cooling)
- Specify flow characteristics with uncertainties
- Define geometry and calculate by-pass flow and uncertainties
- Boundary conditions should be system pressure, mass flow rate and inlet temperatures
- Flow areas – coolant channels in prismatic designs, and packing fractions in pebble-bed designs.
- Uncertainties in thermal conductivities, temperature and power distribution
- Convective heat transfer coefficient uncertainties in pebble-beds and coolant channels in prismatic.
- Non-uniform flow caused by non-uniform pebble packing patterns on the wall ("wall effect")
- Uncertainties in wall friction coefficient and in effective conductivity
- Irradiation induced geometry changes and the resultant bypass flow changes (input from structural analysis). For example for MHGTR: 11 % average bypass; between blocks 1-1.5%; reflector blocks 3.5%, Side reflector and barrel 3%; engineered flow 3%.

The requested output includes:

- Fuel and moderator temperatures (average)
- Gas temperature (average)
- Pressure drop
- Gas temperature distribution at bottom of the core
- Reflector temperatures

The target to obtain fuel and moderator temperature supplemented with uncertainties and compared to experimental data where possible.

5.1.2.2. Exercise I-2b: "Stand-alone Thermal-hydraulics" focused on core and system thermal-hydraulic modeling (DLOFC transient)

This case can be 1D radial with adiabatic boundary conditions top / bottom. The input uncertainty parameters are the power distribution, decay heat with uncertainty distribution, pebble-bed effective thermal conductivity, pebble bed to wall effective thermal conductivity, and emissivity in the core barrel and vessel (include sensitivity studies). The conditions for the analysis are the adiabatic axial boundary conditions at the top and bottom of reflectors, RCCS prescribed boundary condition, 1.1 bar, include natural connection, system pressure stay 1.1 bar, zero flow.

The request output and target of the analysis of this exercise are fuel temperature (time dependence of the histogram), reflector temperatures, RCCS heat load, and fuel temperature uncertainties.

5.1.3. Exercise I-3: "Localized fuel thermal response" "Stand-alone Fuel Thermal heat removal / thermal response"

5.1.3.1. Exercise I-3a - Case 1: Steady State

Two sub cases – single pebble with heat source or two different fuel pebbles with different powers and thermal properties- will be modeled. Volume weighting or flux (reaction rate) weighting will be used as part of the feedback.

The input uncertainty parameters are geometry of fuel design (inner buffer layer and the 1st PyC layer gap, compact / block gap), fuel graphite thermal conductivity as function of fluence, gas temperature, and average Reynolds Number.

The requested output parameters are kernel temperatures, Doppler feedback temperatures for each of the two pebble types, and temperature distribution (moderator temperature).

The target uncertainty parameter is the temperature for feedback purposes (Doppler).

A detailed finite element numerical benchmark will be specified for this exercise.

5.1.3.2. Exercise I-3b - Case 2: Stand-alone power excursion transient Case (simulating RIA)

The input uncertainty parameters for this exercise are the same as for Exercise I-3a plus power excursion history- power up and down; variations in kernel and coated layer properties variation.

The requested output parameters include time dependent kernel temperatures, and temperature distribution (moderator temperature).

The target uncertainty parameter is temperature for feedback purposes (Doppler)

A detailed finite element numerical benchmark will be specified for this exercise also.

Appropriate experimental data is being reviewed and collected for the purposes of Phase I. The experiments to be used include:

- ASTRA critical experiments;
- One of the GT-MHR cases selected, namely: criticality of core configuration; worth of control rods and calculation of kinetics parameters;
- HTR-PROTEUS experiments;
- HTR-10 first criticality case;
- KAHTER data;
- HTTR data;
- HTTF data:
- AVR data.

6. CONCLUSIONS AND FUTURE WORK

This project is a natural and logical continuation of previous international HTGR V&V activities and responds to needs of estimating confidence bounds for results from simulations and analysis in real applications. Among the expected results of this project are:

- Systematic ranking of uncertainty sources;
- Systematic consideration of uncertainty and sensitivity methods in all steps. This approach will generate a new level of accuracy and will improve transparency of complex dependencies;
- All results will be represented by reference results and variances and suitable tolerance limits;
- The dominant parameters will be identified for all physical processes;
- Support of the quantification of safety margins;
- The experiences of validation will be explicitly and quantitatively documented;
- Recommendations and guidelines for the application of the new methodologies will be established.

The IAEA HTGR CRP will establish an internationally accepted benchmark framework to compare, assess and further develop different uncertainty analysis methods associated with the design, operation and safety of HTGRs. As a result the IAEA HTGR UAM CRP will help to address nuclear power generation industry and regulation needs and issues related to practical implementation of HTGR regulation. The realistic evaluation of consequences must be made with best estimate coupled codes, but to be meaningful, such results should be supplemented by an uncertainty analysis.

The use of coupled codes minimizes unnecessary penalties due to incoherent approximations in the traditional decoupled calculations, and to obtain more accurate evaluation of margins regarding licensing limits. This becomes important for licensing new plants, which are very different form the LWR NPPs currently dominating nuclear power generating market. Establishing such internationally accepted HTGR UAM benchmark framework offers the possibility to accelerate the HTGR licensing process when using best estimate methods and contributes to establishing a unified framework to estimate safety margins, which would provide more realistic, complete and logical measures of reactor safety.

The organization arrangement was made with three working groups responsible in the three identified HTR designs, Pebble Bed working group, Prismatic Reactors Working Group (MHTGR 350MWth and GT-MHR).

A second planning meeting is planned for October 2010 (just after this conference) where the different proposal will be finalized. The next step is then to invite participation from the member countries after which, assuming sufficient interest is indicated, the CRP will be officially launched. The CRP duration is planned to be three years only, followed by the publication of an IAEA TECDOC.

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Nine experts from seven Member States attended the first meeting to plan the activities of the CRP. Their contributions made the proposal and this paper possible.
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- C. Parks, United States of America (ORNL)
- J. Kelly, United States of America (US NRC)
- K. Ivanov, United States of America (Penn State University)
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