SUMMARY OF SESSIONS

SESSION I

ANALYTICAL METHODS, COMPARISON OF PREDICTIONS WITH RESULTS FROM EXISTING HTGRs, UNCERTAINTY EVALUATIONS

H. Yasuda
Chairman

The results of experimental and theoretical analysis of spent AVR fuels were presented by Dr. Woloch of OEFZS - Seibersdorf. Burn-up was measured at Jülich and Seibersdorf by gamma spectrometric and mass spectroscopic methods where contents of heavy metals including Cs and Nd were measured. Theoretical analyses were made for these fuel pebbles to give heavy metal inventories by using the new HTR-2000 program system including TOMKU and HTROGEN with the libraries MUPO and WIMS-D. The system can solve the equations for 1156 nuclides taking into accounts the flow path of pebbles in the core during many refueling period by using 25 spectral zones and 139 regions in the core. An equivalence principle was introduced to take account the resonance absorption of heavy metals. Comparisons of heavy metal inventories between calculation and experiments were shown for HEU and LEU pebbles at various burn-up stages up to 10.5 % FIMA.

A French experience in thermohydraulics calculations of gas-cooled reactors were presented by Ms. Boutard of CEA. The LOTE code has been developed in CEA to be applied to the analysis of transient behaviour in accidental conditions up to several months. The code consists of independent modules, modelling the primary flow circuit and main heat exchangers, shut down heat exchanger, blowers and the different materials making up the core. Details of the code, i.e., assumptions, geometrical modelling, and basic equations were explained. Comparisons of core exit temperature and flow rate between calculation by LOTE code and readings of Chinon-3 power plant showed very good agreement in the medium (-1h) and long (-4h) terms. The results suggested the special importance of (1) axial and radial zoning of reactor, (2) Heat exchanger transient modelling and (3) Blower characteristics at low speed.

Extensive work on time dependent neutronics and temperature code (TINTE) was presented by Dr. Scherer of KFA. The code is able to model the primary circuit of HTR plant using modern numerical techniques and relevant physical variables. The HTR core is treated in 2-D R-Z geometry, 2 energy group diffusion theory and 14 decay heat groups model. Gamma ray heating effect can be considered in the calculation by using local and nonlocal heat sources. Validation of TINTE was conducted using the various transient tests, such as rod drop, rod withdrawal, blower speed change and shut-down Xe effect, data obtained at AVR. The results showed good accordance between experiment and calculation. Further validation work is also planned for natural convection and core heat up experiments. The effects of water ingress on reactivity, fuel element corrosion and cooling properties are now being incorporated into the code.

SESSION II

ANALYTICAL METHODS, PREDICTIONS OF PERFORMANCE OF FUTURE HTGRs, UNCERTAINTY EVALUATIONS - PART 1

W. Scherer
Chairman

The session was made up by 5 papers from 3 countries, namely Japan, China and USSR. Three papers were dedicated to the Japanese HTTR project.

The first paper was presented by K. Tokuhara (Fuji Electric Co. Ltd.). The doppler- and moderator-temperature coefficients were calculated using a whole core model by uniform changing of the temperatures. Control rods were assumed to be withdrawn. It was shown that the doppler coefficient lies in the range of $-5.0*10^{-5}$ to $-1.5*10^{-5}$ C^{-1}. The moderator coefficient covers the area between $-17*10^{-5}$ up to $+1*10^{-5}$ C^{-1}. Despite of this positive value the power coefficient stays negative in all situations. The kinetic parameters effective delayed neutron fraction and prompt neutron lifetime were shown to be $beta_{eff}$ = 0.0047 to 0.0065 and $lambda_{prompt}$ = 0.00067 to 0.00078 sec. respectively. These data are necessary for kinetic calculations using point kinetics techniques.

The second paper was presented by T. Nakata (Kawasaki Heavy Ind. Ltd.). The design procedure and the results of the power distribution evaluation for the HTTR were given. One of the main design goals was to minimize the maximum fuel temperature. Therefore a careful fuel zoning and loading of burnable poisons was performed. By combining the results of different core models using the diffusion code CITATION the local power peaking factor was obtained to be about 7% slighty decreasing in time. The effect of local power spikes due to the block end graphite was found to be in the order of 4% at the point of maximum fuel temperature. The accuracy of the calculational methods was tested using Cu reaction rate measurements in the critical facility VHTRC with good results. It was stressed that the methods would meet all the design requirements.

The third paper was presented by Y. Shimakawa (Mitsubishi Atomic Power Ind. Inc.). The ASURA code was developed to design the HTTR plant control systems. It models the main components and systems of the HTTR and simulates the transient behaviour of the whole plant. The verification of this code was performed by evaluation of Fort St. Vrain dynamics experiments and comparison with the validated dynamics code THYDE-HTGR. Results were shown to be adequate and applicable for the HTTR control system design.

The fourth paper was dedicated to the 10 MW Test Module reactor project in China. The general design of this reactor was discussed and special design problems were pointed out. To keep the reactivity effect of water ingress accidents small the heavy metal loading was reduced to 5 g HWM/fuel ball. A reduction of the core diameter to 180 cm was found necessary to keep the shut down reactivity margin in an adequate bandwidth during operation. The reduction of control rod worth by water ingress was considered to be of major importance and additional detailed investigations in this field were announced.

The fifth paper of the session came from USSR and was presented by Yu. P. Sukharev. The effects of variations in the pebble flow velocity and the pebble bed porosity on reactivity, power distribution and temperatures were analyzed for the USSR HTGR projects VG-400 and VGM. It was shown that assumptions on the porosity variations near the core/reflector boundary and above the fuel discharge tubes based on previous theoretical analyses could lead to substantial modifications of the temperature distribution in the core. The maximum fuel element temperature may thus increase by some 200°C. In earthquake situations pebble bed compaction could lead to reactivity effects of about 0.4% as was explained using data from related experiments. The necessity of studies in this field at critical assemblies was emphasized.

In total the session gave an overview of status and problems in todays HTTR projects. It was felt that the Japanese HTTR licensing requirements are fulfilled by the methods available and the analyses yet performed. For the Chinese Test Module fields for future detailed analysis were defined and concerning the USSR projects additional experiments were asked for.

SESSION III

ANALYTICAL METHODS, PREDICTIONS OF PERFORMANCE OF FUTURE HTGRs, UNCERTAINTY EVALUATIONS - PART 2

S. Pelloni
Chairman

Session III included six papers from the different participating countries.

The first paper is from Japan, and was presented by R. Shindo. Neutronics calculations of the Very High Temperature Reactor Critical Assembly (VHTRC) based on the DELIGHT/TWOTRAN/CITATION codes were performed in connection with ENDF/B-IV based data libraries. The characteristics of the VHTRC are similar to those of the High Temperature Engineering Test Reactor (HTTR) of 30 MW thermal power. The core of the VHTRC consists of 280 fuel rods of UO_2 with 4% enriched uranium coated particles. The agreement with experimental values of the calculated eigenvalue k_{eff}, power distribution and of the calculated poison rod worths is good within 1%, 2.9%, and 1% respectively. Thus, the licensing requirements for the HTTR are met.

The second paper is from the USA, and was presented by B. Worley. First, the sensitivity and uncertainty analysis capabilities of the reactor physics models used at ORNL were summarized. Second, a new, generalized formalism for performing sensitivites studies by implementing direct and adjoint techniques into existing FORTRAN computer codes was described. This methodology is used in the code GRESS (Gradient Enhanced Software System). GRESS has been tested mostly for non nuclear applications. The code and the related documentation can be requested without any restriction from RSIC.

The third paper is from Germany, and was presented by U. Ohlig. Extensive physics calculations with the VSOP code system of a pebble bed HTR with LEU fuel characterized by 4% enriched uranium coated particles show that there are large discrepancies (up to 2.5$) between the eigenvalue k_{eff} of the initial core estimated using an old ENDE/B-II based library and a newer library based on ENDF/B-IV, ENDF/B-V, and JEF-1. The discrepancy between the eigenvalues almost vanishes for the equilibrium cycle, due to compensating effects arising from the build up of plutonium and fission products. Safety related properties of the reactor are only weakly affected by the turn to the new library.

The fourth paper is a common paper from Switzerland and from Germany, and was presented by S. Pelloni. A series of two-dimensional discrete-ordinates transport theory k_{eff} calculations with an (r,z) geometric model have been performed for a simple, typical LEU HTR configuration characterized by LEU AVR non irradiated fuel containing 6 grams of 16.7% enriched uranium per pebble, using different methods and data. For a 13 neutron group calculation in the well tested structure from HRB the eigenvalue ranges from 0.9896 for P_0 modified cross sections coming from MICROX-2 to 1.0083 for P_0 modified JEF-1 cross sections from WIMS-D obtained using a new method developed by Segev. Either the use of P1 modified cross sections, or that of P_0 cross sections (suitably modified) in connection with a finer group structure, reduces the discrepancies between eigenvalues of small HTR cores.

The fifth paper is from the USA, and was presented by A.M. Baxter. The paper gives a very detailed review of the important, safety related, physics parameters for the low-enriched Modular High-Temperature Gas-Cooled Reactor (MHTGR), and estimates are presented of the uncertainties in the calculated values of these variables. The MHTGR core contains 20% enriched fissile uranium oxycarbide and fertile thorium oxide fuel. It is pointed out that the core physics experimental data base is adequate to ensure that the temperature coefficient obtained using a broad class of methods and data libraries available in the USA can be calculated with an uncertainty of less than 20%. The present uncertainties of the control rod bank reactivity worth, of the local power distributions, of the core reactivity, of water ingress effects, and of the production of decay heat are of less than 10%, 13%, 1.5%, 10%, and 10% respectively.

The sixth paper is from the USSR and was presented by Yu. P. Sukharev. Uncertainties from calculations of low enriched HTR cores characterized by water ingress accidents were described. The reactor analysis was made with the code system NEKTA/VIANKA/GAVROSH in connection with WIMS-D cross sections. Results of the analysis show that the water inventory in the core as a result of an accident varies from between 10 Kg and 80 Kg to 400 Kg. Corresponding changes of the positive reactivity are between 0.05% and 2%. The discrepancy of water ingress effects predicted by different codes and libraries is about 30%. Since the assessed uncertainties are different in various papers, it was recommended to define a simple LEU HTR Benchmark to be calculated in the frame of the cooperative work.

SESSION IV

CRITICAL EXPERIMENTS - PLANNING AND RESULTS, UNCERTAINTY EVALUATIONS

R. Chawla
Chairman

There were five presentations in this session - four being contributed papers and the fifth being a summary report by W. Scherer, acted as the Chairman of the 2-day Research Co-ordination Meeting held at PSI earlier in the week.

In the first paper, presented by H. Yasuda, an account of the experimental procedures employed at the VHTRC critical facility in Japan for determination of the temperature coefficient over several different temperature steps between 8 and 200°C is given. The importance of correcting for various effects in the measurements was stressed. Lower values of the temperature coefficient were found in the lower part of the temperature range investigated. The agreement with calculations based on the SRAC code was within ~2%.

In the second paper presented by D. Mathews, the current status of work in the HTR-PROTEUS program is described. The safety report for the planned experiments is being reviewed by the Swiss authorities, and measurements are scheduled to begin in Summer 1991. During the currently planned phase (~3 years of experimentation), first-of-its-kind integral data should be made available for (a) criticality of LEU pebble-bed configurations with different moderator/fuel pebble ratios and lattice geometries, (b) neutron balance changes upon water ingress and (c) control absorber effects.

An overview of the HTR-relevant calculational methods and data available at PSI was provided by S. Pelloni in the third paper. The cell codes employed are MICROX-2, WIMS-D (with a specially developed double-heterogeneity treatment) and TRAMIX. The data libraries most commonly used are based on the JEF-1 file. Whole-reactor calculations are usually carried out with the 2-D transport code TWODANT.

The fourth contributed paper was presented by E.S. Glushkov and described how uncertainties for LEU pebble-bed systems have been assessed in the USSR. For K_{eff} predictions an uncertainty of ~2% has been estimated, this being based on (a) comparisons of calculated and experimental results from the ASTRA and GROG critical facilities and (b) an explicit consideration of tolerances on various technical specifications in these experiments. Uncertainty estimates were also presented for temperature coefficient, water ingress, Xe-poisoning, control rod worths and power distributions - the various values being broadly consistent with those given by other speakers at the Meeting.

The fifth paper was presented by W. Scherer who summarized findings from the 1st IAEA CRP Meeting on "Validation of Safety Related Physics Calculations for Low-Enriched HTGRs", held at PSI on May 7-8, 1990. The CRP agreement has been signed by China, Japan, Switzerland, the USSR and the Federal Republic of Germany, and the USA is expected to join soon. Viewpoints and needs are somewhat different in the individual countries, but there was unanimous expression of the need for the planned HTR-PROTEUS program which will form the focal point for activities within the CRP. Experimental results from past and on-going LEU/HTR experiments (VHTRC in Japan and ASTRA/GROG in the USSR) will also be made available, and these should form an additional basis for intercomparisons of methods/data. As a first step, a calculational benchmark exercise will be carried out by the CRP members on the basis of a specified HTR-PROTEUS configuration. Priorities on the types of measurements to be carried out in PROTEUS (paper entitled "Present Status of the PROTEUS HTR Experiments") were discussed and found to be somewhat constrained by the amount of fuel available. The prospect of a second phase of experiments was raised with the suggestion that the USSR might later provide a larger number of LEU (21% enriched) fuel pebbles. Finally, the delegation of scientific staff to PSI for direct participation in the HTR-PROTEUS program was recognized by all as being an important feature of the CRP agreement.