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EURATOM

1. Final publishable summary report

1.1 Executive summary

The Collaborative Project ARCHER (Advanced Reactor for Cogeneration of Heat & Electricity-R&d) is funded by the European Commission under the 7th Framework Programme of the European Atomic Energy Community (EURATOM). ARCHER was a four year project which ran from the 1st of February 2011 until the 31st of January 2015. The consortium consisted of 14 partners from the private sector, 12 R&D institutes, 6 Universities and 2 Technical Support Organizations from Europe. The activities performed were imbedded in the international framework via the Generation IV International Forum (GIF), and direct collaboration existed within the project with international partners from the US, China, Russia and cooperation with IAEA. The ARCHER project's aims and achievements are in line with the Sustainable Nuclear Energy Technology Platform's (SNETP) plans.
The ARCHER project aims to move the High Temperature Reactor (HTR) technology base forward by extending knowledge of state-of-the-art European HTR's by supporting a nuclear cogeneration demonstration. High temperature gas-cooled nuclear reactors (HTRs) combine high safety with high efficiency. They are particularly suitable for operating in cogeneration mode, by supplying of heat and electricity specifically for the European process industry, as an alternative for fossil fuel burning. The HTR technology has been established mainly by former European programs, which were active until the end of the 1980's. ARCHER builds on this, and on the large European HTR technology base (re-)established in previous EU Framework Programmes (FP) such as the FP7 projects RAPHAEL and EUROPAIRS and the nuclear cogeneration working group in SNETP.

The R&D activities in ARCHER were split into four key technological areas. The integration of a HTR cogeneration system in connection to industrial processes system was evaluated and assessed. Different options to couple a process heat application to an HTR are evaluated, and from these coupling studies it can be concluded that an HTR cogeneration plant can support process heat demand requiring process steam at temperatures up to 600°C (or 550°C as the case may be) considering existing or near term technology. The studies on safety aspects focussed on resolving open questions regarding confirmation of HTR safety, as preparation for a licensing framework. These matters varied from different design concepts to core thermal and neutronics behaviour and advanced studies for water and air ingress experiments and modelling. The experimental work on ingress and dust mobilization resulted in an significant extension of the current knowledgebase. Other research focussed on the fuel behaviour. Irradiated fuel pebbles and particles were available from previous projects and were subjected to extensive characterization. From the analysis a better fundamental understanding is obtained, resulting in completely novel approaches to fission product release concepts. The structural material and component work focussed on several areas. The large database that exists in Europe on graphite behaviour was extended by a new irradiation experiment in the High Flux Reactor, which added to the previous work done in the RAPHAEL program. Also other materials were extensively characterizes such as the Alloy 800H. Results of which were directly used in the design an built of a mock-up of a gas-to-gas Intermediate Heat Exchanger (IHX) which is a critical component that can provide improved efficiency and economy of the reactor.
During the project there was a strong focus on communication and dissemination of the work performed. Younger generation of engineers and scientists were included by two EUROCOURSE schools on HTR technology. Education in HTR technology has been supported by the writing of a textbook. To ensure the availability to the Nuclear Cogeneration Industrial Initiative (NC2I) of the SNETP, the ARCHER reports have been transferred to NC2I to be available for its members, and for future consortia working on HTR technology.

The ARCHER project is completed and the 4 years of work resulted in a total number of 114 delivered reports that contribute to maintaining, strengthening, and expanding the extensive HTR knowledge base in

Europe. ARCHER has set up a project webpage at www.archer-project.eu where further information is made available.

1.2 Summary description of project context and objectives

High temperature gas-cooled nuclear reactors (HTRs) combine high safety with high efficiency. They are particularly suitable for operating in cogeneration mode, by supplying of heat and electricity specifically for the European process industry, as an alternative for fossil fuel burning. Europe has a strong history of HTR development and has focused the developments on robust safety, appropriate size and high temperature heat generation. The ARCHER project aims to move this technology base forward by extending knowledge of state-of-the-art European High Temperature Reactors (HTRs) by supporting a nuclear cogeneration demonstration.

The ARCHER project's aims are in line with the Sustainable Nuclear Energy Technology Platform's (SNETP) plans, set out in its Strategic Research Agenda (SRA) and Deployment Strategy (DS). The state-of-the-art of HTR technology has been established mainly by the former European and United States HTR programmes, which were active until the end of the 1980s. ARCHER builds on this, and on the large European HTR technology base (re-)established in previous EU Framework Programmes (FP) such as the FP7 projects RAPHAEL and EUROPAIRS ('End User Requirement fOr Process heat Applications with Innovative Reactors for Sustainable energy supply') and the nuclear cogeneration working group in SNETP.

The R&D activities in ARCHER are split into four key areas:

1. *evaluating and assessing the system integration of an HTR in connection to industrial processes;*
2. *resolving open questions regarding confirmation of HTR safety, as preparation for a licensing framework;*
3. *providing a basic insight into fuel behaviour to support fuel performance code and support robust demonstrator design;*
4. *progressing HTR and VHTR technology developments on materials and component technology by developing, building and testing an advanced heat exchanger prototype putting Europe at the forefront of advances in this technology;*

Four subprojects were defined to address the key areas. The first subproject had a focus on the system integration. The main purpose of SP1 is to identify and solve issues that come up when a multitude of systems and components are combined with the objective of safe and economic operation whilst meeting end user needs. It shall establish a design schematic of a nuclear cogeneration system connected to industrial processes, with sufficient detail for performance assessment of the coupled system, including a gap and SWOT analysis. Additionally a European system code inventory will be established, from which the requirements for integration, development and qualification/validation are assessed, in view of design, performance evaluation, and safety and licensing of a nuclear cogeneration demonstration. Two main tasks were defined to address this;

- *A case study in which the coupling of an HTR to a real industrial site is examined, and its technical viability and performance is assessed*
- *Assess the multitude of various thermalhydraulic, neutronic and thermomechanical codes, that can (partly) describe an HTR system, identify the characteristics and define next steps for code integration and code validation*

The main purpose of SP2 is to investigate further the Key Safety Aspects identified by the Safety Advisory Group of the RAPHAEL project and to contribute to their resolution. Main focus of the work is on air and water/steam ingress into HTGR reactor cores and on the role of graphite/carbonaceous dust in the radiological source term analysis chain from the reactor core to the environment. Collaboration with international and national programmes is considered where opportunities arise.

- *Advance the knowledge regarding the thermal behaviour of HTR pebble beds*

- *Provide experimental data regarding air ingress tests*
- *Provide experimental information regarding the chemical interaction between steam and graphite*
- *Code development and code performance assessment regarding thermalhydraulics, dust modelling, pebble ordering and source term, including validation/comparison with experimental data, both existing and generated in ARCHER*
- *Provide the approach to assess the HTR pressure boundary in licensing context*

The SP3 on fuel capitalizes on the irradiated material available with state of the art PIE, for improved fundamental fuel behaviour understanding and further advancement of fuel performance codes. It will prepare an essential fuel performance validation test in international framework, and will strengthen and extend the unique position Europe holds in HTR fuel back end R&D.

- *Experimental data from Post Irradiation Examination of HTR pebbles irradiated in previous HTR R&D programs (HFR-EU1), and provide insight regarding fission product behaviour, also under accidental conditions*
- *Experimental data from Post Irradiation Examination of surrogate particles irradiated in previous HTR R&D programs (PYCASSO-I and –II), to provide essential material data and validation information for the existing fuel performance codes*
- *Experimental study regarding the behaviour of HTR fuel waste under long term disposal conditions, with focus on radiological impact and fission product diffusion behaviour*

The main purpose of SP4 is to study the materials and components that have reached a maturity level which promotes them as potential candidates for a demonstrator. It will address still pending priority materials issues for the HTR deployment. Graphites that should be used for the reactor core will be specified and design data will be provided. It will make recommendations and a gap analysis regarding nickel based materials and welds, for use in high temperature heat exchangers and components where the code data is lacking. For components it will establish the basis for the introduction of the steam generator as the initial HTR primary heat exchange component of a demonstrator, and it will test a plate IHX mock-up in order to validate a design option that would allow for an extension of the application of HTR to higher temperatures.

- *Graphite irradiation completing the graphite design curves, appropriate for initial assessments of HTR core behaviour*
- *Extending the existing material databases in conventional industry to nuclear HTR applications*
- *High temperature instrumentation and alloys investigation*
- *Novel IHX design, mock-up construction and transient testing*
- *Steam generator assessment, coupled to system integration in SP1*

Next to the technical content of the program, another key target is securing the European HTR knowledge base accompanied by proper dissemination, to facilitate access to the younger generation of engineers and scientists. Inform the public and politics of the details of this technology and the benefit it might bring to society in an accessible way, and ensure the R&D performed finds its way to the international community (for example the Generation IV Forum).

1.3 Description of the main S&T results/foregrounds

1.3.1 SP1 Main S&T results/foregrounds

The main purpose of SP1 is to identify and solve issues that come up when a multitude of systems and components are combined with the objective of safe and economic operation whilst meeting end user needs.

1.3.1.1 WP11 Coupled System Specification

Reference HTR system
With the purpose of performing an assessment of the coupling of an HTR system to a process heat application, reference HTR plant characteristics have been provided by Westinghouse as reported in the table below.

Characteristics Reference HTR

Plant Characteristic	Value
Thermal Power	2x260 MWth
Electric Power	127 MWe
Process Heat Power	400 t/hr
Availability	90%
Primary System Pressure	70 bar
Number of Pebbles	317500
Steam Generator Inlet Temperature	700°
Steam Generator Outlet Temperature	250°
Helium Mass Flow	111 kg/s

A 600 MW nuclear reactor with steam generator outlet temperature of 600°C is specified for the conceptual design. The system is designed to deliver process steam at two pressure levels: the high pressure process steam at 500°C (or 550°C if there is no intermediate reboiler) and low pressure process steam at 300°C. Different values of steam supply rate can be selected depending on the desired mix of process heat/electricity production. Study of the system performance criteria shows satisfactory results.

Fictive Realistic Process Heat Application
The reference ARCHER HTR system is coupled to a typical European process heat application provided in-kind by the Polish engineering company Prochem. For this purpose, the following fictive but realistic process heat application has been assumed. The application is representative for a large chemical complex, more specifically an oil and refinery complex. The plant uses steam as the energy carrier between the different units. Steam was identified as the obvious heat transfer medium if HTR energy is introduced in the conventional industries, with a minimum of modifications in existing plants operating well established industrial processes.
The following main characteristics for the process heat application are defined:

- Steam is mostly used at 3 different pressure levels
    - high pressure (~30 bars, 260°C),
    - medium pressure (~16 bars, 220°C),
    - low pressure (~4 bars, 200°C)

- The steam is mostly generated in a dedicated Power and Steam Station, at high conditions (~130 bars, 540°C). The steam is first expanded in turbines to produce electricity, then admitted into the different pipe headers.
- The steam is frequently in direct contact with the product
- Most of the steam condensates are discharged into the environment, only a fraction returns to the boilers water preparation plant
- A large part of the steam consumption is linked to heating of equipment and infrastructures which must be maintained at a sufficient temperature in winter. As a result, steam consumption is markedly reduced in summer
- Stable steam conditions are most frequently compulsory for a safe and smooth operation. Sudden interruptions may lead not only into production problems, but also into emergency measures. For this reason, the power plant always maintains several boilers in operation, with at least another one in standby mode ready to take-over.

Long Term Potential Coal Processing

As a longer term high potential option for cogeneration, coal processing options have been assessed. It is concluded that at the demonstrator level of heat production with an HTR, an additional system will be needed to elevate the temperature. To this purpose a mechanical heat pump could be used or so-called clean gas combustion.

Assessment of Coupling Options

Different options to couple a process heat application to an HTR are evaluated and conclusions are drawn from this assessment. An indirect steam cycle allows more than 550°C of steam temperature on the secondary side before materials issues appear. These conditions enable the currently best available power conversion efficiencies and fully correspond to the existing heat market demand, but the nuclear heat source is not used to its full potential, because primary coolant outlet temperatures of approximately 700°C are required. Lowering the primary coolant outlet temperature would allow making more robust material and design choices with possible advantages in terms of safety, economy and technology readiness. Issues to be considered in this configuration relate to possible steam ingress into the primary system.

If a customer process requires heat at a temperature above 550°C, the primary coolant circuit would need to operate at a higher temperature, so that an indirect gas cycle is preferred then (because steam would cause material issues at these temperatures). The IHX acts in that case as the barrier for separating the nuclear island from the conventional secondary side. The high temperature fraction of the energy could be delivered to the customer process while the lower temperature fraction is directed to a bottoming cycle for steam generation with either process steam production or electricity generation depending on demand. However, doubts are expressed regarding the economic viability of adding a heat exchanger for a secondary gas cycle, to maximize the usage of the primary high temperatures.

Another, although more costly and less efficient option is a direct helium cycle with a Brayton topping cycle for electricity generation and a steam generator as the bottoming application. In many of the considered configurations, there are options to tap low temperature heat from different places in the secondary, or in case relevant, tertiary system. In all cases, leakage and tritium permeation must be minimized.

A review of available cogeneration cycle concepts is made, mainly from existing experience with fossil fuel fired power plants. This review shows that the condensing steam turbine concept is a suitable application because it offers a wide range of design flexibility and performance specification. They are designed so that steam for the thermal load is obtained by extraction from one or more intermediate stages at the appropriate pressure and temperature. The remaining steam is exhausted to the pressure of the condenser, which can be as low as 0.05 bar with a corresponding condensing temperature of about 33°C.

An assessment has also been made of the feasibility of coupling an HTR cogeneration system with industrial processes by considering the nature of industrial heat demand as well as the technical, economic and safety requirements. An HTR cogeneration plant can support process heat demand requiring process steam at temperatures up to 600°C (or 550°C as the case may be) considering existing or near term technology. However there may be need for further studies on a steam generator using helium as the hot stream. There seems to be no technical impediments to coupling nuclear reactors to industrial process applications, although safety-related studies of coupled systems may still be necessary.

Design Schematics and Analysis
A conceptual cogeneration steam cycle configuration was set up. Thermodynamic analysis and performance evaluation of the nuclear cogeneration system connected to industrial processes was carried out following the design schematics. A system analysis has been carried out of the ARCHER HTR reference system coupled to a secondary system as defined in the ARCHER case study based on a fictive but realistic site using a suitable secondary circuit configuration selected out of the options studied separately before. Thus, this provides an example, using real end user data from an oil refining and petrochemical complex of how an HTR could be used in a cogeneration plant. A mass and heat balance of the entire system have been derived and reported. The main conclusions are that a nuclear power facility will have to be built close to an important industrial complex (thus probably adding difficulty to the licensing process). The economic viability of an HTR cogeneration project is considered as one of the main challenges for the technology in the future.

Key indicators and comparison to a modern gas plant
A key performance indicator assessment is performed in which a number of key performance indicators are defined as a basis for comparison between HTR cogeneration and a reference gas-fired cogeneration plant. The purpose is to identify pathways towards improved employment of HTRs, specifically with respect to the performance when being used for cogeneration. The main outcomes are summarized below:

Plant Characteristics
The plant characteristics are selected such that the CCGT plant and the HTR plant are to a large respect very similar, i.e. with respect to the power level, the availability, the plant life time, and the space requirements. Obviously, differences are found in the time to market and the construction duration.

Environment
The assessment shows that when waste is considered, the HTR system produces more radioactive waste. With respect to all other forms of waste, the HTR produces less or even much less waste. Typically, radioactive waste is associated with very long lifetimes. This cannot be neglected. However, one should realize that some of the conventional or chemical waste will never be taken out of the environment and will always remain dangerous to living creatures. Where to put more emphasis is not so much a technical issue, but rather a political question. Further quantification with regard to the economic parameters associated with external costs is also provided and from a technical-economic point of view a nuclear system would have the preference.

Safety
With respect to safety a comparison based on actual facts is hard to achieve as statistics are lacking for nuclear, let alone for an HTR. If the HTR system will be able to fulfil its promising safety characteristics, the HTR system might be a little safer than a CCGT which as such is relatively safe when compared with many other forms of energy generation. A rather different aspect of safety is the licensing process. For HTR, the licensing of a plant in combination with cogeneration has never been shown. Furthermore, much of the licensing expertise and knowledge from the HTRs which have been operated has been lost and needs to be retrieved and updated in line with current licensing practices.

Economics
The economic assessment leads to some obvious conclusions that nuclear is expensive with respect to the construction costs and therefore is very sensitive to the interest rate during construction. On the other hand the operation and maintenance costs and the fuel cycle costs are relatively low. When the energy

generation costs are considered for plants which only produce electricity, one can show that the costs for both CCGT and nuclear are comparable at a discount rate of around 10%. When lower discount values are considered, nuclear becomes more attractive, on the other hand, when higher values are considered, CCGT becomes more attractive economically. Further, it is shown that the energy generation costs for an HTR employed with cogeneration in a plug-in market will approximately be of the same order of magnitude as CCGT. However, the assessment has also shown that when the costs are compared to the benefits, i.e. the price of the energy (electricity and process heat) sold to the market, it becomes clear the HTR is only viable when it is employed in combination with cogeneration and will not be economically attractive as an electricity only production facility. It should be noted that this situation may vary with respect to the country where the HTR system is foreseen. In this assessment, no attempt has been made to identify components which are critical with respect to their costs in order to determine where major cost reductions could be achieved. For such an evaluation, a bottom-up cost estimating approach would be more suitable.

Generic Outcome
An attempt has been made to provide an objective overview of the different indicators important for decision makers. One should realize that the decision to construct an HTR or another energy generation facility are not only technical, not only economical, but are highly influenced by public opinion and politics. Therefore, it is impossible to draw firm conclusions. However, a generic outcome can be summarized as follows:

- An HTR system can be considered economically viable in Europe when it is employed for the production of electricity and process heat.
- A demonstration of an HTR with cogeneration will be a very important step, as it will remove the current uncertainties associated with licensing and time to market.
- The environmental impact of an HTR is minimal, as long as the safety features claimed are true.
- There seems to be a lower boundary of the break-even ratio of process heat and electricity price. This means that the price for process heat should be at least a certain percentage of the price of electricity for a coupled system to become profitable. This value may vary from case to case.

South-African Process Heat Application:
As a typical South-African process heat application, a coal to liquid process currently in operation in Secunda was taken. The reference HTR system defined in the ARCHER project was coupled to this plant. In the analysis, a techno-economic baseline for an existing coal to liquid process using the Aspen Plus chemical modelling software was performed to determine mass and energy balances. The economic performance of a typical 80,000 barrels per day synthetic crude oil plant was determined from first principles. The techno-economic baseline model was validated with reference to published product output data and audited financial results of a coal to liquid plant located at Secunda which uses relatively cheap available South-African coal to provide the required process heat and electricity, as reported for the 2012 financial year.
A number of schemes were identified to couple the reference HTR plant to the coal to liquid case study. Two schemes were studied in detail. For the study, two key performance indices were used. Firstly, the Internal Rate of Return of a nuclear supported coal to liquid plant and secondly the reduction in CO2 emissions resulting from the introduction of nuclear energy. The case where nuclear cogeneration replaced electrical power bought from the grid and all the steam currently produced by plant's internal coal fired steam plant, revealed the following conclusions.
The case study plant would need a total of 16 reference HTRs. The coupling scheme would reduce the emission by approximately 1650 ton/hr CO2 (14.5 million ton/yr) or 50% of the current emissions. The economic feasibility challenge in the specific South-African situation with cheap coal at hand for large scale deployment of nuclear energy in a coal to liquid application is to construct such a facility at an all-inclusive overnight cost not exceeding \$3400/kWe. It should be noted that for a cogeneration system, a cost

indication in $/kWe is misleading. One should realize that apart from electricity also process heat may be produced which changes the interpretation of this figure.

1.3.1.2 WP12 Technology Gap and SWOT Analyses

Westinghouse has performed a technology gap analysis mainly based on the experience of the company being the legal successor of the German HTR companies involved in AVR and THTR. The gap analysis discusses possible gaps in all major components and additionally provides supply chain information where available. Apart from the technical gaps, also the regulatory framework and the commercial impact and gaps are shortly discussed.

Many of identified gaps address (economic) efficiency rather than safety issues. Some gaps address improvements in analysis tools (experimental techniques and simulation approaches). The main gaps are related to economic performance and re-establishment of the supply chain. Within the EU framework (RAPHAEL, ARCHER), some ‘fundamental’ technology gaps have been analysed, e.g. graphite qualification, integrated modelling package, dust formation and transport, and stratification loads.

A SWOT analysis was performed by the industrial partner EA starting from their own experience and the work of Prof. Kugeler and LGI ordered by JRC and provided as in-kind contribution to ARCHER. They approached the SWOT analysis from a technical, economic, strategic, legal, and societal point of view. The main conclusions of the SWOT analysis are:

- Large market forms opportunity for low CO2 emission HTR technology.
- Large European knowledge and experience base.
- Energy from an HTR is affordable, reliable (i.e. security of supply and safe), and clean (low carbon) which fits European energy policies.
- HTR heat provision is flexible (large range of pressures and temperatures).
- HTR fits very well in small grids. Many heat intensive industrial complexes are located in such areas.
- Nuclear safety concepts are difficult to explain to the public at large. Nuclear risks and liabilities are hindering deployment.
- European energy is divided and different in each member state.
- A consistent and up-to-date regulatory framework is lacking.

Finally, system integration guidelines were derived taking into account the outcomes of the technology gap and SWOT analysis. These guidelines also include the R&D challenges towards demonstration, i.e. safety demonstration, licensing support, and technology Innovations. Also a statement on the R&D required to reduce costs of HTR applications was made together with a recommendation to work on a business case which is mature enough to approach private investors have been included.

1.3.1.3 WP13 Primary System Code Integration

In order to enable technical analysis of a nuclear cogeneration system, a multitude of computer codes will be needed. Therefore, a code inventory is established of codes being used in Europe and South Africa for which the requirements for integration, development and qualification are assessed. To this purpose, the information of system codes of different companies was obtained by sending out a survey to the ARCHER partners. The data collected from the survey, including references, was thoroughly checked and categorized. The survey comprised the ‘system codes’ which are capable of predicting parameters for the whole primary and secondary circuit and their respective couplings to neutron physics codes and/or containment codes. In order to evaluate the capabilities of the various system codes, different categories were introduced into an evaluation table comprising neutronics capabilities, thermal fluid dynamics within the reactor pressure vessel, thermal fluid dynamics for the primary and secondary circuit and also for processes within the containment. This categorisation helps to identify and highlight the capabilities of the different code systems and display possible code improvement needs in order to establish a full primary and secondary circuit simulation (optionally also for containment) for high temperature gas-cooled reactors. Next to this, code development efforts in Europe were assessed with the purpose to arrive at a European HTR simulation platform. Apart from an elaboration of how this may be achieved and which codes could be included, recommendations are provided on the design of the codes, on how to deal with legacy codes, on how to deal with code version management, on how to agree on conventions, e.g. on which unit system to use in all the codes, on how the documentation of codes should be structures and provided, and on how to achieve an integrated development environment.

1.3.2 SP2 Main S&T results/foregrounds

SP 2 dealt with key safety aspects of the primary and secondary circuit of an HTR. It built on results of the RAPHAEL Project as documented in the Recommendations of the RAPHAEL Safety Advisory Group, on experience from past and recent licensing activities for HTGR and on public discussion of HTGR safety aspects. Accordingly, the subproject was subdivided in five work packages on air ingress, water or steam ingress, the source term analysis chain, thermal safety issues, and the integrity concept of the primary (helium) pressure boundary. In these work packages, experimental tasks to broaden the data base for future code validation or safety assessments as well as computational and other safety analyses were performed.

1.3.2.1 WP21 Air Ingress

This work package dealt with the safety issues related to a potential ingress of air into an HTGR reactor core at operating temperatures or elevated accident temperatures. It intended to enhance the respective experimental database by conducting experiments with the existing NACOK-II test facility in Jülich, Germany and performing respective pre- and post-calculations.
Since the cooperation with INET in work package WP24 could not be realized as planned the work has been enhanced with a second NACOK experiment and some small-scale thermal hydraulic experiments on pebble beds in the existing INDEX facility. In addition, the scope of the NACOK experiments has been focused on the secondary Boudouard reaction at high temperatures, since a lack of knowledge about this reaction was identified in the literature while the primary oxidation reaction of graphite appears well investigated.
The two NACOK experiments, one with a stack of graphite blocks (Figure 2-1) with internal cooling channels and one with a pebble bed have been performed as well post-calculations with the computer code MGT-3D. While a good agreement between experimental and numerical results could be observed in the beginning of the test runs, the chemical reactions were more intense in the experiments than in the calculations in later phases of the experiments. The probable reason for this, which is under further investigation at FZJ, lies in the increasing porosity of the graphitic specimen with the elapsed time, caused by the corrosion in the pores of the graphite which characterizes the Boudouard reaction in the temperature regime at about 1200 °C. The small scale experiments with the INDEX facility have been performed and the results are available for validation of CFD computer codes modelling pebble beds.

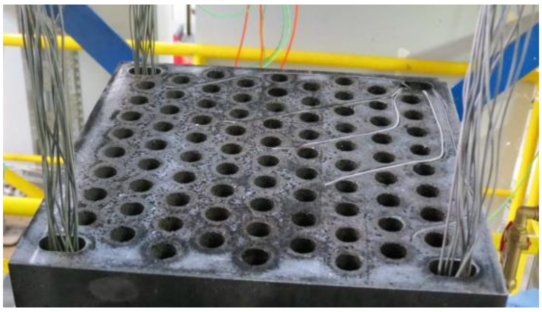


Fig. 2-1: Oxidized block from the NACOK test

1.3.2.2 WP22 Water Ingress

This work package dealt with the potential for formation of explosive gas mixtures and with corrosion of the fuel elements due to an ingress of water or steam into an HTGR core at operating temperatures or elevated accident temperatures. It comprised both experiments and computer analysis in three different tasks.

Task 2.2.1 Potential for Explosive Gas Mixtures, Flammability Limits

The objective of this task was to determine flammability limits of gas mixtures of hydrogen, carbon monoxide and methane which could be produced in an HTR after water ingress, when these gas mixtures are released into the atmosphere of the reactor building. The experiments have been performed in a “spherical bomb” facility at CNRS. A laser was used for ignition of the gas mixtures.
Flammability diagrams (Figure 2-2) have been determined for previously defined gas mixtures of hydrogen, methane, carbon monoxide and steam in continuous mixtures in air and in the presence of the diluents carbon dioxide or helium. The flammability diagrams have been determined at ambient temperature and both ambient and increased pressure (5 bar). Originally intended experiments at elevated temperatures of 150 °C were cancelled in favour of the determination of complete flammability diagrams which by far extended the original scope of the task.

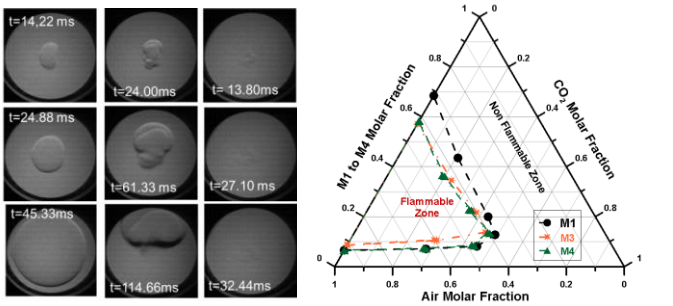


Fig. 2-2: Flammability detection and diagrams

Task 2.2.2 Analysis of Potential for Explosive Gas Mixtures

This task had the objective to improve the capabilities of the engaged partners with regard to computational analysis of water ingress accidents. For this purpose, existing computer codes were enhanced with correlations for corrosion and with the capability to treat nuclear reactivity effects of the water ingress simultaneously with the corrosion effects.
The enhanced computer codes have been applied to a modular HTR corresponding to the Chinese HTR-PM design. For the design basis event: instantaneous rupture of one steam generator tube, it could be shown that former assumptions of 600 kg of steam entering the core and reacting there were conservative. The detailed calculations performed now show that less than half of this value would need to be considered.

Task 2.2.3 Water/Steam Ingress Corrosion Experiments

The objectives of this task were to determine experimentally if and under which conditions the exposure of reactor graphite to humidity from a steam generator leak would lead to combustible or explosive gas mixtures, and compare the results with thermodynamic predictions and results from historic irradiation tests. Originally the tests were planned in a respectively licensed laboratory at JRC-IE in Petten. When it turned out that the experiments could not be performed there, the KORA facility of JRC at ITU in Karlsruhe was selected and planning of the experiments there was started. At the end of the second reporting period, a reassessment of all JRC activities in the ARCHER project showed that due to uncertainties in the timely completion of the KORA facility and the experiments, and due to the necessary concentration of JRC/ITU workforce on other ARCHER tasks in SP 3, task 2.2.3 had to be cancelled.

1.3.2.3 WP23 Source Term Analysis Chain

This work package dealt with gaps in the HTGR source term analysis chain from analysis of fission product releases from the fuel elements to the analysis of environmental dispersion of fission products and activation products including evaluation of dose and risk to the public. In a recent licensing process (PBMR) the emission of contaminated carbonaceous dust has been identified as a potential main contributor to early releases from modular HTGR after leaks or breaks of the primary pressure boundary. Six tasks of this work package therefore concentrated on dust production, contamination, and deposition in the primary circuit as well as dust remobilization in the primary circuit and its behaviour in the reactor building in the course of depressurization events. A separate task dealt with the potential issue of Tritium permeation from the primary to the secondary circuit and subsequently to the environment.

Task 2.3.1 Simulation and Validation of Fission Product Release as well as Estimation of Graphite/Carbonaceous Dust Production in the HTR-10

The objective of this task was to evaluate the fission product release and dust generation data of HTR-10. As a precondition for fission product release calculations with the computer code STACY, the steady state condition of the HTR-10 core with regard to temperatures and burn-up was calculated. In the further assessment of HTR-10 data, however, it became obvious that, due to prolonged outages in the past, the HTR-10 core has not yet reached its equilibrium state. Therefore a decision has been taken to evaluate the current state of the HTR-10 core based on information published by INET. The respective calculations have been performed together with predictive calculations of the fission product release from the core with the code STACY.
The work intended to evaluate experimental data from the HTR-10 and compare them with the calculated results had to be cancelled because the intended collaboration with INET could not be achieved within the ARCHER timeframe. The low amount of publicly available data did not justify a detailed comparison. The budget dedicated for this sub-task was therefore spent for the calculation of the HTR-10 burn-in phase.

Task 2.3.2 Graphite Dust Production and Deposition

The dust production in an HTR core in a small pebble mill with moving graphite spheres was simulated. The produced dust should be characterized with regard to shape and size distributions and, if possible, dust be made available for other experiments. The abrasion experiments should be performed under ambient air and under Helium atmosphere. The experiments with different abrasive partners (number of interacting spheres, wall; different graphite / carbonaceous materials) have been performed and the results have been documented. Characteristic dust parameters, depending on graphite grades and friction forces, have been determined. Based on their number, most dust particles exhibited a diameter smaller than 5 μm. However, the particle size distribution showed a tendency to shift to higher values with increasing contact forces. Most particles had aspect ratios between 0.5 and 0.8, i.e. the particles were non-spherical, but not rod-shaped either. It was therefore concluded that it should be possible to use existing correlations for spherical particles for their transport behaviour if shape factors are taken into account.
The last part of the task was to investigate the deposition behaviour of the generated dust behind flow obstacles and in dead end areas. The experiments have been performed in a small Helium test. The comparison of the particle size distribution of dispersed dust with that deposited as a single layer on a flat plate showed an increase of the mean particle size suggesting larger particles are more susceptible to deposition. The creation of multilayer deposits in front of a forward facing step was investigated as well. A recirculation area in front of the obstacle with size and flow characteristics depending on the Reynolds number was observed with the particle layer below it growing at constant velocity. The particle deposition

velocity increased almost linearly with the Reynolds number. Simultaneously, the density of the layer increased, while the volume decreased.

Task 2.3.3 Dust Mobilisation Experiment

The objective of this task was to contribute to the assessment of the potential for dust mobilisation in case of an accidental leak or break of the primary circuit. The planned experiments aimed at providing dust mobilisation data for later computer code validation. For this purpose the BISE facility at IRSN laboratories in Saclay was used after appropriate additional instrumentation, which is a small-scale wind tunnel in which a pile of powder is exposed to a horizontal air flow. In this facility the parameters governing the re-suspension phenomena of loose dust deposited in eddy zones can be studied. Dust parameters as size distribution and particle shape have been selected in accordance with task 2.3.2, however, since under that task it was not possible to produce enough graphite dust for task 2.3.3, alumina powder was used. Since the task aimed at providing data for code validation, this deviation is acceptable. For the shape of the heap of powder a cone has been selected because flat multi-layer deposits can already be sufficiently described with existing numerical models.
It turned out that the free-stream velocity and the particle size distribution were the most influential parameters, and a strong positive interaction was observed between them. The experiments also revealed that the resuspension process is a threshold phenomenon: there is a free-stream velocity threshold under which no measurable resuspension occurred. The results were also compared with two models found in the literature, which were found to be not valid for the geometry of dust pile investigated in this task. Therefore a dedicated semi-empirical relationship was proposed.

Task 2.3.4 Dust Deposition and Remobilisation in a Pebble Bed Experiment

Task 2.3.4 also aimed at providing experimental data for future computer code validation. For this purpose the deposition of dust in a pebble bed and its remobilisation at increased gas flows was studied. For the experiments an existing dust transport facility was used at HZDR. The aerosol particles to be deposited in the pebble bed were radioactively labelled with the isotope fluorine 18 and the spatiotemporal distribution of the particles was recorded by means of Positron Emission Tomography (PET). With this method, the resuspension of dust which had been deposited in the pebble bed before could be monitored online and, depending on the characteristics of the depositing and of the remobilising gas flow, threshold values for the remobilisation could be determined. It turned out that above a remobilization flow threshold velocity of about twice the deposition velocity, about 40 % of the deposited dust are soon remobilized while the rest adheres to the pebbles.

Task 2.3.5 Validation of CFD Approaches (RANS, LES) for Dust Behaviour

This task concentrated on the validation of the so-called Reynolds Averaged Navier-Stokes (RANS) and Large Eddy Simulation (LES) CFD approaches for dust deposition and resuspension in the geometrically more complex components of HTR primary circuits. Currently available experimental data, new experimental data generated within the ARCHER project and data from detailed numerical experiments performed using Direct Numerical Simulation (DNS) were used for this purpose.
The first part of the task comprised the generic validation of the RANS and LES CFD approaches to predict the particle transport and deposition. Detailed literature review was performed to collect and evaluate the available experimental and DNS data pertaining to the particle deposition validation. In case of RANS modelling, several uncertainties regarding the near-wall modelling were addressed. A robust and well validated RANS model coupled with a specific near-wall correlation was proposed for the accurate particle transport and deposition predictions. On the other hand, the capability of a more advanced model such as the LES was validated by performing two test cases.

In the second part of the task, based on a combined system code – CFD code approach, a step was made towards accurate analysis of the considered dust phenomena in an HTR, utilizing among others the experimental results from tasks 2.3.2, 2.3.3, and 2.3.4. It is shown that a RANS approach for regular pebble arrangements shoes good agreement with the much more expensive DNS approach, which opens a passable route for future CFD simulations of coolant flow and dust behaviour in pebble beds. In a final step the RANS model has been applied to coolant flow and dust deposition in a randomly packed pebble bed (Figure 2-3).

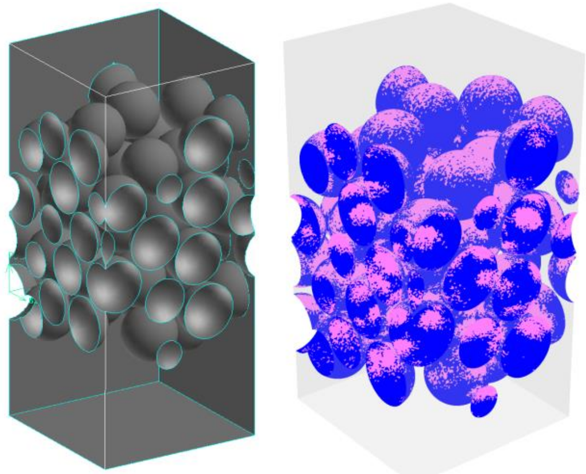


Fig. 2-3: Model and particle deposition from the RANS CFD computations

Task 2.3.6 Dust Behaviour in Building and Environment

The overall objective of task 2.3.6 is the numerical simulation of dust behaviour in an HTR building. A principal software tool for performing calculations of aerosol behaviour in reactor buildings is the COCOSYS code which has been extensively used to study containment phenomena of severe accidents of LWR. Within task 2.3.6, the aerosol models already implemented in COCOSYS were expanded to cover all aspects of graphite dust behaviour in HTR which in some cases differ from those of the LWR aerosols, and detailed calculations for an exemplary HTR building were be performed. In order to provide realistic information for the confinement calculation, it was necessary to include a simulation of the thermal hydraulics inside the primary circuit during depressurisation. For this purpose the finite-differential thermal hydraulics code DIRECT developed at FZJ was used.

The necessary improvements of the physical models in the codes have been implemented, a calculation model has been created and first calculations have been performed. In the second part of this task, the above mentioned calculation has been repeated with an updated Version of the general COCSYS code and with a refined model of the reactor building. Calculations have been performed for the HTR-Modul design basis event “DN 65 break of a pressure equalisation line” and for a beyond design basis 960 cm² break at the bottom of the Reactor Pressure Vessel. For the design basis break only a very small amount of dust (~10 g) was released from the primary pressure boundary into the reactor building, whereas during the postulated hypothetical event about 4 kg would be released. In both cases about half of the dust released into the building would be retained there, the rest would be released into the environment. No accident dust filtration was assumed.

Task 2.3.7 Tritium Permeation in Heat Exchanging Components

The aim of this task was to perform a parameter study indicating which combination of the different tritium control options would be capable of meeting the safety requirements, complemented by suggestions for a viable tritium control strategy. For this purpose, a conceptual steam generator lay-out based on the Fort Saint Vrain design has been assumed to generate the required steam generator temperature profiles and surfaces. On their basis, using also assumptions on the tritium source term of an HTR core, coolant chemistry, steam generator tube thicknesses and tritium permeabilities through the concerned materials, the acceptable tritium partial pressure in the primary circuit, requirements for gas chemistry and helium clean-up systems and performance of possible tritium permeation barriers have been determined.

It is concluded there that a viable tritium control strategy would not be problematic to achieve, provided plant designers build in some flexibility in operating conditions to allow compensating for various uncertainties. This applies both to a closed steam cycle for electricity generation where the accumulating tritium concentration would be decisive boundary condition for radioprotection reasons of workers, and to an open steam cycle where the contamination of the process steam would be limiting. Although this limits is very low, this condition will actually be more easily met than for a closed steam cycle. Obviously the elaboration of a more detailed tritium control strategy will require the knowledge of the licensing conditions for new-build HTR.

1.3.2.4 WP24 Thermal Safety Issues

This work package dealt predominantly with the issue of correct prediction of temperature distributions in (pebble bed) HTR cores. Two analytical tasks and one combined analytical/experimental task were dedicated to this issue. A separate task was dedicated to the liaison with the ISTC project “Hot Astra”.

Task 2.4.1 Melt Wire Experiment at HTR-10 and Comparison with Thermal Core Analysis

The objective of this task was to prepare and conduct a melt-wire experiment in the Chinese HTR-10. Graphite spheres instrumented with set of wires melting at specified temperatures should be inserted into the core of the HTR-10 and later on be examined for the maximum temperatures they had experienced on their way through the core. A similar experiment had been performed at the AVR in the 1980s. The task should also comprise pre-calculations of the temperature profiles in the core of the HTR-10 for comparison with results obtained with the melt-wire spheres.

Due to the occupation of INET with the detailed design of the HTR-PM which is currently under construction, the performance of the melt-wire experiment turned out to be not possible within the time frame of the ARCHER project. Therefore task 2.4.1 had to be reduced in scope to the pre-calculation of the HTR-10 equilibrium core, which was also needed in task 2.3.1, and to the generation of a specification for the production of melt-wire spheres and a proposal for the conduct of the experiment, the latter two thought as a preparation for an experiment possibly to be conducted after completion of the ARCHER project.
The first part of the redefined task 2.4.1, the calculation of the HTR-10 equilibrium core, has been performed and the results have been documented in deliverable D24.11. The specification of the melt-wire spheres and the proposal for conduct of the experiment have been provided, but the experiment in the HTR-10 and the comparison of the results with the pre-calculations, had to be cancelled.

Task 2.4.2 Analysis of Hot Spots/Areas in HTGR Cores

The objective of this task was to perform numerical calculations in order to investigate the potential for hot spots respectively hot areas in pebble bed reactor cores, either by local variations of the pebble bed density or by incidental loading of clusters of fresh fuel. The resulting thermal hydraulic and nuclear effects were assessed in combination. The calculations were performed with a respectively updated version of the code system ATTICA-3D/TORT-TD.

The model improvements were mainly related to the consideration of the fact that the fuel spheres pass several times through the core before they have accumulated their discharge burn-up. Therefore a mixture of fuel spheres with different burn-up, power and reactivity exists in every sub-volume of the pebble bed. With this model, steady state calculations of densified areas in the pebble bed have been performed. With these local porosity variations as a starting point, passive heat removal scenarios with core heat-up have been investigated as well as control rod withdrawal and ejection scenarios.

While local densifications of the pebble bed in the region of the axial core power maximum lead to a local increase of steady state core temperatures which is still visible several meters below the densification, the maximum operational fuel temperatures are not increased because they at the bottom of the core. A densified zone at the bottom of the core has no sensible effect on the maximum operational fuel temperatures either because the local power density is low at the bottom of the core.
The higher local starting temperatures for passive heat removal with a densification in the axial power maximum equalize during the slow transient, therefore the effect on the maximum fuel temperatures during passive heat removal is insignificant.
No violation of fuel temperature limits, i.e. the 1600 °C limits was observed in control rod withdrawal scenarios or after withdrawals (with operational speed) or even fast ejection of three control rods. Only in the hypothetical cases of fast ejection of all control rods a violations of the 1600 °C criterion was observed for fuel spheres in their first and second pass through the core. This is a new insight since in the past such calculations have been performed with an average fuel sphere at average burn-up. Such scenarios must therefore be ruled out by appropriate design measures, e. g. demonstration of incredibility of failure of control rod housings if they are outside the primary pressure boundary.

Task 2.4.3 Liaison with ISTC Project #685.2 "HOT ASTRA"

The objective of this task is to maintain the liaison with ISTC project #685.2 ("HOT ASTRA") to ensure that project results concerning HTGR core physics at elevated temperatures are being transferred to EU FP project partners, through the (established) western collaborators (AREVA and NRG). This task thus depends on the progress made with the HOT ASTRA project. During the first periods of the ARCHER project it became obvious that HOT ASTRA struggled with funding problems regarding necessary enhancements of the facility and with the wish of the Russian partners to finalise the project already in 2013. Actually HOT ASTRA has been closed in 2013, the enhancement of the facility was achieved, but the criticality experiments at high temperatures could not be performed until the end of the ARCHER project.

Task 2.4.4 Investigation in Porosity Variations of Pebble Beds

The objective of this task was to calculate the pebble distribution in the core of a pebble-bed reactor and to explore the possibility of simulating the pebble flow in such reactors. The variation in density was experimentally investigated in both radial and axial directions in the core. Furthermore the stochastic pebble-bed stacking calculations were repeated many times to investigate the variations in density between different realisations.

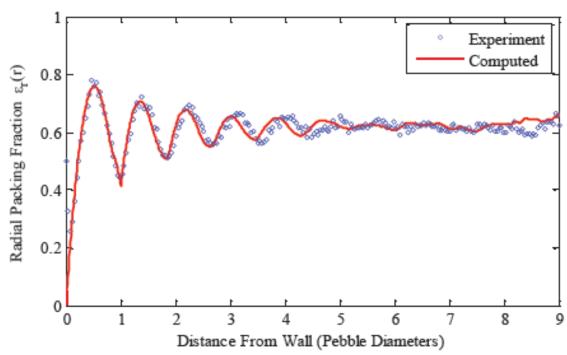


Fig. 2-4: Radial packing fraction of pebble bed measured with PebBEx

Results for the computed bed were found to be in good agreement with the experiments (Figure 2-4) and with literature, giving confidence that the method is capable of generating beds with realistic packing structures, although the experimental results for the microscopic stacking properties in the near wall region are of insufficient quality for a meaningful comparison. Analysis of the various results showed different stacking properties near the wall than in the bulk of the bed, indicating the stacking is anisotropic near a boundary forming semi-ordered layers parallel to the wall with hexagonal-like stacking properties, which implies flow and heat transfer might be anisotropic near the wall and could need different models near the wall than in the bulk to be accurately described.

Finally, the probability distribution of packing fractions of small clusters of around 45 pebbles showed that the local packing fraction inside a packed bed can vary strongly, both in the bulk and near the wall, which might significantly affect flow rates and could result in hotspots. The results informed the selection of densified pebble bed zones in task 2.4.2.

1.3.2.5 WP25 Integrity Concept of the Helium Pressure Boundary

This work package deals with the basic application of the “Integrity Concept” developed for LWR to components, e.g. pressure vessel and piping, of HTR or to components of other advanced nuclear reactors approaching the creep regime.

Task 2.5.1 Loading & Boundary Conditions, Regulatory Aspects

The objective of this task was to provide the operating and transient conditions of steel pressure boundaries of recent modular HTR designs and to identify relevant fault conditions.
Operating and transient conditions were provided for three different modular HTR designs: firstly the German HTR-Modul (applicable also to the Chinese HTR-PM and a potential European Demonstrator in the 200 MW_{th} class or lower) which used LWR pressure vessel steel and remained in the temperature regime for that material both during normal operation and design basis accidents, secondly the PBMR 400 (applicable also to other concepts of similar power output) which also used LWR steel, but would exceed the temperature range of LWR experience during design basis accidents, and thirdly some pre-conceptual NGNP proposals which featured a “hot vessel design” operating in the creep range already during normal operation and intending to use modified 9% Cr steel or 2¼ % Cr steel.

Task 2.5.2 Identification of open Questions

The objective of this task, based on the input provided by task 2.5.1, was to describe the LWR Integrity Concept and to identify additional steps that are necessary for the application of the concept to HTR pressure boundaries taking into account concepts of “cold” pressure boundaries (same material and operational temperature range as for LWR, accident temperatures higher for a limited period of time, but below creep range) and of “warm” pressure boundaries (operational and accident temperatures in the creep range, use of other materials such as 9% C steels).

From the results it was concluded that or those components that operate at temperatures below the creep range the LWR Integrity Concept (IC) can be used taking additionally into account the HTR-specific damage mechanisms irradiation (RPV only, for HTR-Modul reduced below threshold by a radial zone of boronated carbon bricks) and corrosion in HTR-helium.
For components operating at higher temperatures where time dependent effects must be taken into account the LWR IC must be extended to assure that the causes of possible operational damage mechanisms (creep and/or creep-fatigue processes) do not impact the component integrity. Due to the complexity of ongoing and competing damage mechanisms advanced simulation methods and appropriate material data must be used to assure the safe operation. The application of the IC in this range must be based on a reliable and sufficient materials data base.

Task 2.5.3 Methods for safety assessment -input from conventional power plant technology

The objective of this task was to investigate modern conventional power plant concepts with live steam temperatures between 600 °C and 700 °C to identify potential benefit for the integrity of HTR primary pressure boundaries operating at high temperatures or aiming at similar live steam conditions.

In addition to the principles of the basic safety concept BSC (Quality through production as basic safety and multiple parties testing, worst case, documentation and validation as independent redundancies) and to the established design rules for components operated in the regime of time independent behaviour, advanced inelastic analyses as well as damage and failure assessment for components operated in the time dependent regime must be included in an integrity concept for such a temperature regime.
Experience from developments for fossil plants shows that a reduced ductility and deformation capability has to be taken into account. A high deformation capacity is an additional safety margin and required in the BSC. Hence, the deformation behaviour has to be investigated and taken into account in the assessment procedure. Additional fracture mechanics investigations must be included in the safety concept to ensure sufficient crack resistance in case of reduced deformation capacity of a material, and the influence of multiaxiality on the damage development has to be taken into account.

1.3.3 SP3 Main S&T results/foregrounds

The SP3 on fuel capitalizes on the irradiated material available with state of the art PIE, for improved fundamental fuel behaviour understanding and further advancement of fuel performance codes.
Creep and swelling properties of the SiC layers due to the neutron dose as much as elastic modulus and hardness measurement (by HT nano-indentation of SiC within up to 500 °C) were one of the focuses. Some preliminary work (irradiation feasibility and tailored shape particles) concerning an international irradiation (CPSTRESS) to assess the performance of mechanical stress codes for irradiated material was successfully achieved.
The diffusion process of the main FPs of interest was made possible by different set-ups and fuel manufacture. Fuel from past German programmes irradiated in HFR Petten undergone some safety test and associated PIE was performed jointly with some code qualification. Some specific long term tests and discussion concerning waste management were introduced to strengthen and extend the unique position Europe holds in HTR fuel back end R&D.

1.3.3.1 WP31 Fuel Basic Properties

WP3.1 provided underpinning science based on data enabling a better understanding of the fundamentals of coated particle fuel. The approach relied on separate effect studies both in and out of pile, largely with non-active kernels to eliminate artefacts due to fission.

The key data produced during the period were:

- Creep data on coated particle layers (especially PyC) under normal and internal operational temperature: X-Ray tomography analyses concerning the PYCASSO particles made available at NRG
- Mechanical properties of irradiated SiC by high temperature nano indentation techniques at UMAN: HT nano-indentation of SiC up to 500 °C achieved
- New model for Ag transfer through the SiC to pin down Ag diffusion: UMAN finalized a new model concerning the migration of Ag in the SiC layer
- Particles fabrication for separate effect irradiation tests (CPSTRESS): a viable path to manufacture some CPSTRESS particles investigated jointly at TU Dresden, UMAN and JRC-IET

The X-Ray tomography work performed at NRG on the irradiated particles focused on the knowledge on the densification kinetics by determining the swelling and shrinkage of the separate layers as a function of fluence. The particles investigated in this work were provided by CEA (France) and consisted of various coating materials and combinations for TRISO coated particle fuel. Close to 500 particles have been measured 24/7 by X-Ray Tomography.

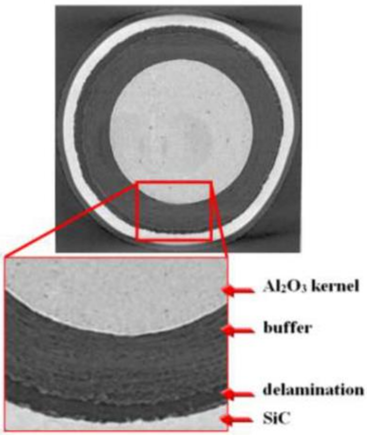


Fig. 3-1: XRT of irradiation TRISO Particle

The XRT measurements performed at NRG and the analysis performed in University of Manchester show a clear difference between the two types of studied particles. The particles that consisted of a kernel, buffer layer and OPyC layer clearly showed shrinkage in the two graphite layers. The buffer is known to shrink, but without any restraint, the OPyC layer follows the buffer layer, shrinking up to 3.7% in radius, whilst increasing up to 3.5% in thickness. The particles with a SiC layer (see figure 3-1) between the graphite layer restrains the PyC from shrinking, resuling in residual stresses in the PyC.

Even though nano-indentation is an established technique at room temperature, significant developmental efforts were required for in-situ measurements at elevated temperatures. This work resulted in the first report on elastic modulus and hardness measurement by HT nano-indentation of SiC within up to 500 °C.
The elastic modulus from the CEA particles irradiated in PYCASSO was found to decrease steadily with temperature over the measured range, while the hardness showed a more pronounced drop, which was in accordance with literature data obtained by other techniques. The elastic modulus of the irradiated samples was slightly higher than in the pristine state. There was an obvious increase in the hardness of the irradiated samples, which was around 7% at room temperature and more than 20% for the highest measurement temperatures suggesting that irradiation hardening has occurred. This data is the first of this kind and it will help to improve the database for current fuel performance models of HTR fuel. So far these codes had to rely on data generated on dummy specimens, since the TRISO coating geometry does not allow the application of standardized mechanical testing.

The work conducted at the University of Manchester follows two approaches to study the interaction of silver and silicon carbide in its fundamentals. At first, pellets consisting of silver and silicon carbide powders were fabricated, which were subsequently heat-treated at temperature well above the regular reactor operating conditions. The second part was a trial study of entrapping a thin layer of silver between two SiC coatings in a TRISO particle. Based on the experimental findings and thermodynamic considerations a new model describing the migration mechanism in silver through silicon carbide has been proposed. It is concluded that silver can move through SiC by a dissolution-precipitation mechanism, a theory supported by extensive thermodynamic calculations. The experiments and calculations show that in its liquid state silver dissolves SiC by forming a silver-silicon alloy and a separate carbon phase. Tests including silver containing TRISO particles showed that silver penetrates the SiC coating rapidly and in large quantities causes recrystallization of SiC grains.

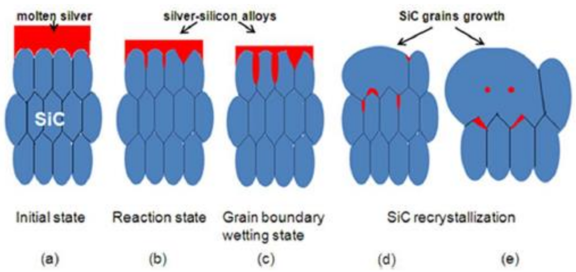


Fig. 3-2: Ag diffusion mechanism through SiC

This proposed mechanism (Fig. 3-2) is completely novel approach to look at the cause of silver release from intact SiC coatings in TRISO fuel. Former reports have focused on diffusion theories or vapour transport models through nanocracks in SiC. A migration mechanism that is founded on a reaction might be unavoidable, but still could be mitigated by a careful engineering of the SiC microstructure since grain boundary characteristics will influence the silver release rates observed in irradiation experiments.

The diffusion process of the main FPs of interest (Ag, Cs, I) was investigated by ARMINES by the mean of a sealed tube placed under vacuum and heated up in tubular furnace at different temperatures (in the range 1050°C to 1300°C) for annealing time from 24 to 85 hours. Although the significant amount of investigation techniques, the measurements performed on the batch of coated particles do not produce up the expected profile results.

NRG initiated a feasibility study concerning a new Pycasso-type irradiation named CPSTRESS including internal pressurization of the coating by the addition of B_4C to the kernel to generate helium during the irradiation process. A report jointly with INL providing a feasibility study of the irradiation within the HFR Petten has been issued. This international support proved the importance of such an irradiation in the future to qualify mechanical codes assessing the integrity of the irradiated particles at high pressure (i.e. 350 MPa and 1200 °C). The production of doped B_4C and Ag particles using pulsed laser deposition (PLD) was performed at TU Dresden with the support of UMAN. In order to analyse the different batches of particles produced a collaboration with JRC-IET was introduced. The PLD coating process was optimized along the two rounds of particles that have been characterized by SEM, EDX, XRD and CT.
As a result of this study, TRISO-like surrogate kernels delivered by Manchester University have been successfully coated with B_4C and Ag-SiC layers by PLD at Technical University Dresden. Homogeneous coatings of the spherical kernels were achieved by introducing a rotating crucible as holder in the PLD set-up. It can be confirmed that particles required for separate effect irradiation tests can indeed be fabricated. This will allow further progress in fuel performance assessment which is a required step in the licensing procedure of a future HTR demonstrator.

1.3.3.2 WP32 Integral Fuel Element Irradiation Performance and Safety Testing

Reliable fuel will remain the most essential requirement for HTR deployment. This can only be achieved by irradiation testing, both for fuel qualification and qualification of the empirical type engineering codes, an essential component for licensing.

Fuel samples from past German programmes (AVR) were unloaded from the HFR Petten (HFR-EU1 irradiation). NRG performed gamma spectrometry scans on the pebbles at NRG Hot Cells on both pebbles and graphite structure. A NRG fuel transport to JRC-ITU concerning two HFR-EU1 AVR pebbles took place at mid-February 2013.

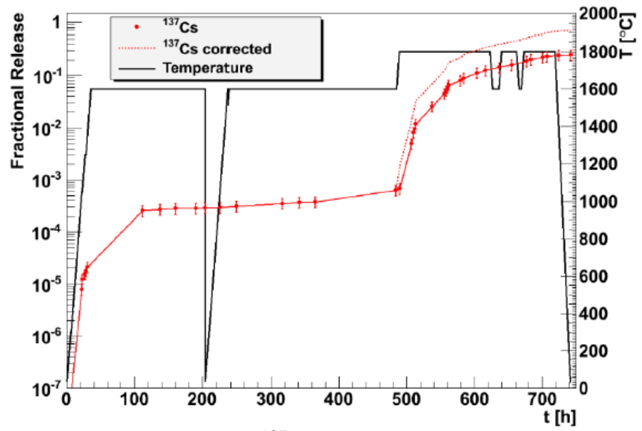


Fig. 3-3: Cumulative ^{137}Cs release from HFR-EU1/3 during the heating test

Both HFR-EU1/3 and HFR-EU1/4 pebbles were gamma counted in the JRC-ITU Hot Cells with a dedicated Denal pinhole collimator and the associated burnup was calculated form the Cs-137 content. The safety test (representative of a Loss Of Forced Cooling accident) was performed on the HFR-EU1/3 pebble using mainly a two plateaus profile (figure 3-3) (first plateau at 1600 ^{0}C for 300 hours and second at 1800 ^{0}C for 200 hours). Throughout the 1800°C phase, the entire Ag-110m inventory was released and the Cs release increased to about 70 %, significantly exceeding all previous heating tests carried out at JRC-ITU. Furthermore, a total fraction of (6 ± 2) × 10^{-4} Eu-154, the release of Kr-85 became significant and amounted to (1.263 ± 0.006) × 10^{-3} at the end of the campaign, corresponding to 12.1

coated particle inventories. The tested fuel retains fission products remarkably well at a nominal accident reference temperature of 1600°C. However, for temperatures towards 1800°C the retention capability decreases significantly.

A new model taking into account the variation as a function of burn up of the isomeric branching ratio for isomeric states (such as Am-242m or Ag-110m) has been implemented by IRSN in the VESTA software. This has been applied to the HFR-EU1 irradiation experiment. The HFR fuel spheres have been modelled using unperturbed neutron spectra for each of the three reactor positions in which the irradiation rig was placed. The comparison of the calculations with the experimental values (mainly Ru-106, Ag-110m, Sb-125, Cs-134, Cs-137, Ce-144, Eu-154, Eu-155 and Am-241) is very satisfactory with the exception of the results for Am-241. The overestimation of the Sb-125 activity by 50 % is most likely a cross section problem.

The FZJ computer code PANAMA simulates the mechanical performance of TRISO coated fuel particles under normal operation and accident conditions. Calculations of fuel performance and metallic fission product release during the HFR-EU1 irradiation based on the operational data, were performed as much as a comparison with predictive calculations from the past. The calculated failure probabilities of the TRISO particles indicated that no particle failure is expected to occur for the Chinese sphere, while for the AVR sphere, failure fraction is approaching the level of one failed particle towards the end of the test which is consistent with the very low FP release encountered during the irradiation phase. The failure fraction predicted to occur during the safety test after a subsequent 200 hours heating phase at 1600°C were also provided.
A calculation has been done with the aim to model radioactive fission product release-to-birth of Xe-133, Kr-85m and Kr-88 of the capsule containing HFR-EU1 /3, /4 and /5 fuel pebbles, using the CEA/AREVA code ATLAS V3.0.3. The calculation showed a really good agreement of ATLAS calculation with experimental data for Xe-133, but a poor agreement for Kr-85m and Kr-88.

At JRC-ITU, three samples (representative of outer, mid-radius and central zones of the pebble) were obtained by carrotage through the centre of the AVR pebble (HFR-EU1/3) previously KüFA tested fuel pebble. Macrographs and micrographs of the cut sections by OM and SEM revealed microstructure evolution of the coated particles during irradiation. The boundaries between the kernel and all coating layers were well-defined, with gaps between the kernel and the buffer layer and some delamination phenomena between the low and high density PyC. On all the investigated particles, small cracks on the SiC layer were detected that are enough to lead to a complete release of highly volatile fission products. This is in agreement with the results achieved on the HFR-EU1/3 KüFA test where the entire Ag-110m inventory of the pebble and up to 70% of the Cs was released. SIMS mappings were performed on kernels and should be completed by complementary electron probe microanalyses in the months to come. The acoustic microscopy technique (to assess some mechanical properties as the Young modulus and to give access to small cracks in the SiC) will also complement the PIE results.

1.3.3.3 WP33 HTR Fuel Back end

The objective of this WP focused on the stability of the coating and the integration of existing HTR spent fuel performance models. The suitability of the irradiated fuel for direct disposal as much as other disposal or reprocessing options were discussed and recommendations were provided..

The key data produced during the period were:

- Limitation of the coating stability: under air at 90 °C, formation of SiO_2 and a clay-like Mg–silicate
- Recommendations concerning V/HTR spent fuel management and public acceptance

The alteration of BISO (with pyrolytic carbon) and TRISO (with SiC) particles under geological conditions were simulated at ARMINES at temperatures of 50 and 90 °C and in the presence of synthetic groundwater. Solid state (scanning electron microscopy (SEM), micro-Raman spectroscopy, electron probe microanalyses (EPMA) and X-ray photoelectron spectroscopy (XPS)) and solution analyses (ICP-MS, ionic chromatography (IC)) showed oxidation of both pyrolytic carbon and SiC at 90 °C. Under air this led to the formation of SiO_2 and a clay-like Mg–silicate, while under reducing conditions (H_2/N_2 atmosphere) SiC and pyrolytic carbon were highly stable after a few months of alteration. At 50 °C, in the presence and absence of air, the alteration of the coatings was minor. Due to their high stability in reducing conditions, HTR fuel disposal in reducing deep geological environments may constitute a viable solution for their long-term management.
For V/HTR spent fuel management, specific additional considerations have to be undertaken due to the fact that the graphite moderator is an integral part of the fuel element. The large graphite fraction of the V/HTR fuel is the 'key' to the inherent safety features of V/HTR and the 'price' to be paid.
The direct disposal of spent V/HTR fuel would possibly not be acceptable in case of a larger V/HTR fleet, because of the large associated volumes and large amounts of steel for the containers.
In case of direct disposal of V/HTR spent fuel, it was shown that the fission product release characteristics are mainly governed by the corrosion resistance of the pyrocarbon and the SiC coating layers. In addition, engineered barriers, containers for the disposal of spent V/HTR fuel, the potential embedding of the coated particles in glass or SiC and the choice of the fuel kernel (UO_2 vs. UCO) have a major impact on the release mechanisms in a final repository.

It is questionable whether new designs of V/HTR can only be based on the fission product retention characteristics of the coated particle fuel. Advanced containment and filter designs as well as catalytic recombiners to cope with flammable gases in case of air or water ingress need to be developed to achieve enough confidence in the safety assessment organizations and in the public. Much more R&D is necessary to underline the benefits of the V/HTR safety features / advantages.
The management, transport and storage of V/HTR fuel is not perceived in a different way, as compared to other radioactive waste from other reactor systems. The mere volume of the spent fuel from V/HTR and the relative high uranium enrichment raises additional concerns, in the general and political public.

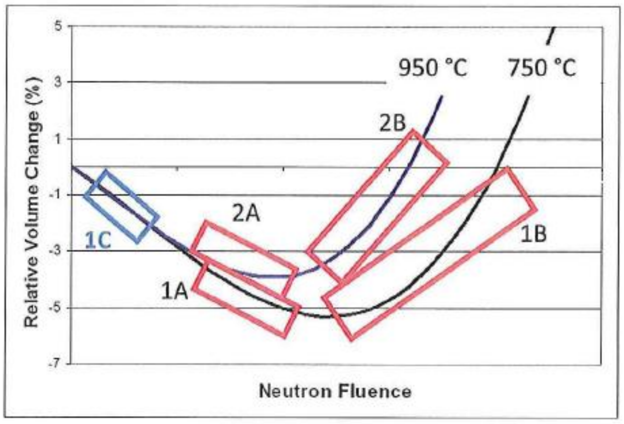
Some recommendations could be summarized as follow:

- Steps need to be undertaken to reduce the ^{14}C generation in V/HTR by limiting the nitrogen content within the graphite and the primary system as far as possible. Leaching of long-lived activation products, such as ^{14}C and ^{36}Cl, from the spent fuel matrix also needs to be considered, because these activation products play a major role in performance assessments of final repositories
- Develop test processes for the separation of coated particles and/or fuel compacts from the graphite matrix
- Avoid using thick-walled containers. Fully ceramic containers (e.g. SiC) made of the tails from irradiated graphite may be an adequate approach.

1.3.4 SP4 Main S&T results/foregrounds

1.3.4.1 WP41 Graphite

For the design of the core of a High Temperature Reactor (HTR) it is important to know the full irradiation behaviour of the available graphite grades in the temperature region of interest. Obtaining this information currently requires irradiation experiments to be performed at an early stage in the design process to have the data available when it is needed in the detailed design phase. This work package examines the irradiation behaviour of available graphite grades and provides a down-selection for use in the HTR Core. One or more of the grades (when fully qualified) may be used in the next generation HTR and to make the



1. INNOGRAPH-1A 750°C, low/medium dose (FP5)
2. INNOGRAPH-1B 750°C, high dose (RAPHAEL)
3. INNOGRAPH-2A 950°C, low/medium dose (RAPHAEL)
4. INNOGRAPH-2B 950°C, high dose (RAPHAEL)
5. INNOGRAPH-1C 750°C, low dose (ARCHER)

Fig. 4-1: Schematic overview of the Graphite Irradiation Experiments performed in HTR-M1, RAPHAE-IP & ARCHER.

selection it is important to determine the irradiation behaviour of each graphite tested, and then to down-select from these the better graphites from a design viewpoint. To this end, the graphites were assessed mainly in terms of their peak shrinkage, the dose at which the peak shrinkage, or shrinkage "turn-around" occurs, the dose to reach original dimensions/volume, the anisotropy in the dimensional change behaviour and the scatter in the data. The work in the first task includes the post irradiation examination (PIE) of the high dose experiments performed in RAPHAEL at 750°C and 950°C (INNOGRAPH-1B and INNOGRAPH-2B, see figure 4-1)) and the assessment of the data for each graphite taking into account results from previous RAPHAEL irradiations 750°C and 950°C (INNOGRAPH-1A and INNOGRAPH-2A) at low to medium dose rate. In addition strength tests and micro-structural and x-ray diffraction measurements have also been carried out to provide information for evaluation and modelling.

As a first step, screening results were produced for the INNOGRAPH-1B and INNOGRAPH-2B experiments to assess general trends and to see if any of the grades behaved significantly differently to others. The results were plotted against other results at 750°C and 950°C previously obtained in RAPHAEL to provide an overall view of behaviour combining the low to medium dose results with the high dose results which reached to ~23 dpa at 750°C and ~16dpa at 950°C. The Full PIE investigations of the high dose experiments were then completed and results obtained for the major and minor graphite grades tested. The experiment included graphite grades currently being used in test reactors operating or being constructed around the world plus the best available grades recommended by the manufacturers. The table 4-1 lists the major and minor graphites in the INNOGRAPH-1B and INNOGRAPH-2B experiments.

Manufacturer	Grade	Coke	Grain size	Process	Classification
GrafTech	PCEA	Petroleum	Medium	Extrusion	Major
	PPEA	Pitch	Medium	Extrusion	Major
	PCIB-SFG	Petroleum	Super-fine	Iso-moulding	Minor
	LPEB/BAN	Petroleum	Medium	Extrusion	Minor
SGL	NBG-10	Pitch	Medium	Extrusion	Major
	NBG-25	Petroleum	Fine	Iso-moulding	Minor
	NBG-18	Pitch	Medium	Vibro-moulding	Major
	NBG-17	Pitch	Medium	Vibro-moulding	Minor
Toyo Tanso	IG-110	Petroleum	Fine	Iso-moulding	Minor
	IG-430	Pitch	Fine	Iso-moulding	Minor

Table 4-1: Major and minor graphites in the INNOGRAPH-1B and INNOGRAPH-2B experiments.

The full PIE tests on selected grades were generally consistent with graphites irradiated in the past and the dimensional change data showed that for both temperatures most of the samples had gone well beyond shrinkage turn-around. The results also showed that a significant number had exhibited positive growth/volume change (i.e. swelling beyond original dimensions/volume). The Dynamic Young's Modulus data exhibited the expected increase at lower dose to a plateau, followed by a further increase at medium dose, and finally a decrease at high dose. The coefficient of thermal expansion results were found to have decreased and reached a stable plateau at medium dose, with indications of a slight increase at high dose.

Also, as expected, the thermal diffusivity and conductivity values, which typically fall rapidly at low dose, had reached a plateau at medium dose.

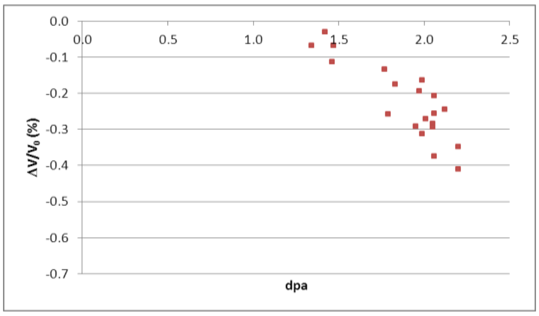


Fig. 4-2: Graphite results at low dose 750°C

In terms of further evaluations on the irradiated graphite twenty samples from the 750°C irradiation tests were examined using optical microscopy with nine of these also scanned using the SEM. Bright field and polarisation micoscopy images were also obtained for selected areas and magnifications of up to x10,000 using SEM. The results form a catalogue of images to be used in later investigations. In addition diametric compression tests were carried out on forty samples from the 750°C irradiation tests and thirty non-irradiated graphite specimens covering with and against grain orientations to assess the effect of dpa on strength. The results show the tensile strength to increase with increasing dpa up to and around turn-around and then to decrease again. In some cases the strength almost returns to the unirradiated values. Fractography was also peformed using a scanning electron microscope on ten of the tested specimens. The results add to the catalogue of images to be used as part of the microstructure and modelling developments and graphite behaviour interpretation.

An assessment of the full PIE results was performed addressing the findings from the 750°C graphite irradiation experiments INNOGRAPH 1A, 1B and 1C and the 950°C experiments INNOGRAPH 2A and 2B. The assessment looked at the dimensional change behaviour of the graphites tested, together with changes in dynamic Young's modulus (DYM), coefficient of thermal expansion (CTE) and thermal conductivity (TC). In terms of properties, the assessment shows that all the graphites behaved in a similar way but with significant differences in the dimensional change behaviour of the different graphites, which is the most important attribute from a core design point of view. There were large differences between the extruded and iso-moulded graphites in the 750°C experiments and between the extruded and vibro-moulded graphites and the iso-moulded graphites in the 950°C experiments attributed to the differences in grain size rather than the manufacturing method. There were also significant differences found between the with-grain (WG) and against-grain (AG) directions, which was not expected for some graphites. The extruded and vibro-moulded graphites had a medium grain size (2mm down to 0.7mm), whereas the iso-moulded graphites had a fine to super-fine grain size (0.1mm down to 5µm).

In terms of dimensional change behaviour, the critical factors are the rate of initial shrinkage, the peak shrinkage (i.e. at turn-around), the fluence at which this occurs, and the fluence at which cross-over occurs, which is when the dimensions/volume return to their pre-irradiation values. For any particular graphite, this is considered to be the limiting fluence for the core as it is impractical to design a core structure which can accommodate significant, if any, nett growth. For the graphites tested at 750°C , the fluence for cross-over ranges from 8.5 dpa to 13.0 dpa for the AG direction, and 14.5 dpa to 22.5 dpa for the WG direction. The corresponding results at 950°C are from 7.0 dpa to 9.4 dpa for the AG direction, and 8.0 dpa to 11.2 dpa for the WG direction. For each graphite tested, therefore, the fluences to reach cross-over was found to be dramatically reduced when increasing the temperature, generally by a factor of ~2. This could have implications for the VHTR, where the graphite temperatures will be significantly higher than for a HTR, and therefore it is essential that the highest temperatures in the graphite do not coincide with the highest fluences. The designer needs to take all the changes in properties with irradiation into account taking full advantage of the favourable behaviour/attribute of a particular graphite and minimising the effects of the unfavourable ones.

A further task involved a low dose experiment INNOGRAPH C at 750°C (see figure 4-1). This utilised a 3 column arrangement which was inserted in the HFR core and subjected to three loading cycles. The objective of the experiment was to obtain data up to 1-2dpa for those properties that are more rapidly affected by fast neutron irradiation (Young's modulus, strength and thermal diffusivity/conductivity) in this region. It was considered that the micro-structural changes that arise can have an effect on their medium and high dose behaviour and a fuller understanding of the changes that occur might lead to a better understanding of property changes with irradiation and help the development of new graphite grades with improved irradiation tolerance at medium to high doses. The design details and results for the major and minor graphites featured were reported and the results showed dimensional shrinkage to start at around 1 dpa. These results were also compared with the other 750°C data to provide a better description of the full curves for the different graphite properties. The DYM values increase between 0 and 1 dpa to a plateau at approximately 1.5 times the virgin value before it continues to increase at around 5 dpa. The CTE value increases before dropping to a stable level at around 10 dpa which is lower than the virgin CTE value. The TD and TC values fall to a stable level two to three times slower than the virgin values within 1-2 dpa for the graphite grades tested in the INNOGRAPH programme. Seven graphite samples irradiated in the INNOGRAPH-1C irradiation rig were prepared for optical microscopy and scanning electron microscopy. Both optical microscopy and SEM were successfully performed. The XRD results show changes in lattice parameters and coherence length. The neutron damage causes the X-Ray coherence length to be reduced in both a- and c-directions.

This work in ARCHER is the completion of a major & unique irradiation program for HTR Graphite Core development that covers both the prismatic and pebble bed designs. The programme provides elevated temperature results for all the graphites featured in current test reactors and provides vital new information on other available graphites recommended by manufacturers.

1.3.4.2 WP42 High-Temperature Alloys and Instrumentation

Materials for key components such as the Intermediate Heat Exchanger (IHX) require industrial materials to operate for significant periods of time at high temperature. The effects of creep and environment are therefore critical, also the manufacturing process and any interaction with cyclic operation (fatigue). Within ARCHER the focus of the work on metals is towards Alloy 800H addressing time and cycle dependent effects on material behaviour (creep/fatigue) and developing an understanding of the influence of different manufactured forms on material behaviour and application. Aspects such as condition monitoring (miniaturized samples), performance in a corrosion environment (loop tests), and sub-critical crack growth and in-service inspection issues were investigated.

Alloy 800H has been examined and tested to understand the material's capability at temperatures between 650<850°C with specific work on thin plate and welded joints undertaken for the IHX. In addition to conventional testing, a literature review of the effects of heat treatment, welding process and welding parameters on different types of cracking has been performed. This includes solidification cracking in the weldment, liquation cracking and/or ductility dip cracking in the heat affected zone and relaxation cracking in large welded components due to residual stresses in air, steam and helium (He). Thin plate and welded material has been investigated with regard to issues of residual stress and environment and the need for reliable prediction for long term operation.

A specific task on material procurement and data collation has been performed which feeds into the existing database with the purpose of expanding the available information to include feedback from the involved industrial partners. The procurement and transfer of material for the tests involved two blocks of Alloy 800H (dimension: 500mm x150mm x16mm) plus two WIG-welded plates (dimension: 500mm x150mm x16mm) which was provided and forwarded to partners involved in testing. Tests were performed on the WIG-welded plate of Alloy 800H, also four relaxation tests on cross-weld specimens and four Gleeble tests with simulated heat treatment followed by slow strain rate tensile tests (10^{-6}/s). Tensile tests at room temperature (RT), 700°C, 800°C, showed a significant reduction in strength between 700 and 800°C and that the fatigue life is substantially affected by tensile holds. A good agreement was obtained for the strength of welds comparing results with recent information produced by ASME. For the evaluation against material codes and gap analysis, gaps in the materials data were identified, and recommendations given. Key issues addressed ranged from data for a 60 year operation life; data for the primary pressure boundary; corrosion influence; joining and long term operation at temperature.

With regard to corrosion, a survey of corrosion data generated in former HTR and related programmes was performed to provide information on candidate materials and mechanisms of degradation in V/HTR helium coolant. Experiments utilizing a high temperature furnace (HTF), see Fig 4-3, were carried out for different materials (P91, Alloy 800H including weld, 316SS) at 750-760°C for exposure times up to 1500h and a pressure of 1 bar with a low flow rate in a simulated helium environment (helium containing minor impurities – CO, CO_2, CH_4, H_2, H_2O). Post evaluation of the results included microstructure investigations, mass change, general corrosion, hardness and fracture toughness changes. In addition some tests were performed in a high temperature helium loop (HTHL) on Alloy 800H (parent and weld) and P91 steel at pressures up to 4.5 bar with a high flow rate. However frequent problems with the HTHL (new device) meant that the extent of these results was limited. For the tests in the HTF the 316SS material showed a significant decrease in fracture toughness; for P91 the decrease was moderate. The main influence was temperature. The corrosion layer for P91 comprised of chrome and manganese oxide. For Alloy 800H, after exposure, surface and subsurface layers were examined and the change in weight of the specimens recorded which showed the highest mass gain of the tested materials. An extreme flooding event meant that further tests had to be rescheduled and a planned in-pile test would not be available before the end of the ARCHER program, also the planned testing of a new thermometry device could not go ahead because of resource reductions. A report on the In-pile and out-of-pile tests was however produced to give the available position and status of the experimental equipment.

Fig. 4-3: High Temperature Furnace (HTF)

For the work on Condition Monitoring some initial information was provided with a review of approaches for active and passive components given. For the HTR it is likely that the latest most developed condition monitoring approach and methods will be applied. Periodic investigation of small samples taken from well-defined locations in the plant (micro-pillars) could provide an attractive tool for damage assessments and such an approach has been investigated in ARCHER for a vessel material. The results show that further work is necessary to understand the discrepancy between the yield stress measured with micro-pillars and the ones determined with conventional samples for annealed material. This conclusion fits in with the current understanding and development of the approach and supports the recommendation to investigate this type of damage condition monitoring further before it can be considered an alternative approach to the currently developed methods. For the work on the development of a best practice proposal for In-service Inspection a review of such techniques was performed considering the requirements of the IHX including the compact design to provide some initial information. Additionally the results from low cycle fatigue tests were used to investigate the stress-strain behaviour of Alloy 800H and to develop material parameters for a Chaboche type visco-plastic material model which can be used for design optimization and code assessment of components made of Alloy 800H operating at elevated temperatures such as the compact IHX and SGU. The parameters for this model were provided for future analysis and investigation.

1.3.4.3 WP43 Intermediate Heat Exchanger

This WP addresses the needs and development of a compact and modular Intermediate Heat Exchanger (IHX). The gas to gas IHX is a critical component that can provide improved efficiency and economy of the reactor but requires considerable development to achieve a robust and compact arrangement that is capable of resisting the high temperature environment for long periods. The work focused on the design and development, manufacture and testing a mock-up of a promising PSHE under realistic temperature conditions. The work was broken down into five tasks covering design calculations, welding and machining, manufacture, tests in the CLAIRE Loop at CEA and materials tests on representative welded features.

For the design activity iterative sizing and computer fluid dynamics (CFD) and finite element analysis (FEA) were carried out to minimize thermal and mechanical stresses under steady and transient conditions so that the length and height could be chosen to provide smooth thermal gradients throughout. The header position for the IHX and mock-up was optimized and the shape of the plate carefully studied to limit the occurrence of any flow recirculation areas. Plate thicknesses were selected according to pressure differences between fluids and the outside and the width so that external welding would be sufficient to withstand the pressure differences for different load cases. Depending on the loading, the parts would be subjected to tension, compression or bending and the worst damage effect was used to size and optimize the component.

The mock-up was designed (figure 4-4) and manufactured (figure 4-5) using Alloy 800H with laser welding used for the manufacture. The welding and machining processes were optimized to ensure a high quality for the component. Aspects such as the machining of the thin plates, the development of custom tools to avoid deformation and the final forming of the designed plates and their laser welding were addressed and successfully completed. The mock-up has the same dimensions as the real IHX module, but contains fewer plates (20 instead of 300), without loss of representation. The manufacturing work used to produce the IHX mock-up represents a significant achievement and strongly indicates that such a geometry representative of a larger IHX can be satisfactorily produced.

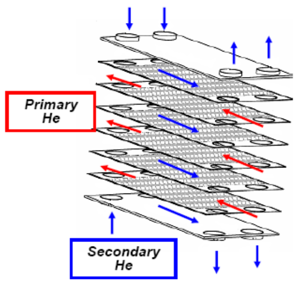


Fig. 4-4: Schematic of IHX plate assembly

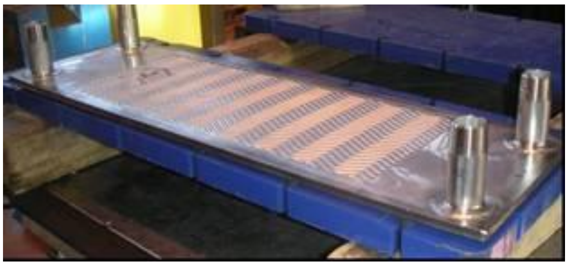


Fig. 4-5: Stainless steel mockup to check CFD thermal and Hydraulic laws

Both CFD and thermo-mechanical calculations were carried out to provide a comparative platform to assess the behaviour and lifetime of the mock-up and its implications for the full size IHX module. The heat exchanger was submitted to a significant numbers of shocks during testing with temperatures increasing from cold conditions initially to establish the test rig to above service elevated temperature conditions (800°C) with measurements taken of its performance. The calculations were compared against the test results and temperature measurements, to determine the effectiveness of the mock-up modelling and its simulation. From a thermal point of view, the heat exchanger provides the expected thermal efficiency (of 90%) in the whole range of Reynolds number tested. From an integrity point of view the mock-up behaved satisfactorily throughout the complete test period and provided a strong basis for further development and investigation.

For the lifetime assessments a review of data for Alloy 800H design properties for thin and thick sections was completed. Although a significant amount of material test and property data are available for thicker section Alloy 800H (typically plate or bar material exceeding 15mm section), the amount of testing on Alloy 800H or other variants of Alloy 800 in sections of 5mm or less is low. The main technological limits for the IHX are creep (long term effect) and thermal fatigue which are so far limiting applications to <850°C. The testing has been carried out in air at ~750-800°C representative of operating conditions and to provide a conservative testing envelope for the mock-up. The output from the calculation and test investigations include limits of operation and recommendations for improving the design and for establishing and extending the operational envelope of the proposed full size IHX module.


Fig.4-6 Tests in the CLAIRE_Loop

Some representative laser welded samples were produced for mechanical testing, and specific tests carried out as part of this investigation on the mock-up material and joints to provide direct information for the mock-up lifetime estimation. These include: tensile properties of plate and welded structures (covering specific features in the mock-up heat exchanger to check against thicker section plate properties); creep tests on plate and welded structures [650–850°C] including tests on laser welded joints and thin section plate. Comparison with European Creep Collaborative Committee (ECCC) data and testing on machined specimens taken from welded features plus longer term creep data were used to understand the strength and behaviour at elevated temperature. Tests focus on creep and strength testing of welds and parent plate to determine weld factors for life time predictions.

The work on the Intermediate Heat Exchanger (IHX) focusses on demonstrating the feasibility through the manufacture and testing of a representative IHX mock-up to assess the compact IHX (PSHE) lifetime. It is considered that this objective has been satisfactorily achieved and valuable information has been obtained for future development.

1.3.4.4 WP44 Heat Transport Circuit

Steam Generator concepts are currently in use in several HTR demonstrators overseas. All seven constructed HTR, including prototypes and the grid-connected THTR-300, deliver their heat to steam generators operating at temperatures significantly lower than the reactor's capabilities. Steam Generators are also in use in the UK AGR's which operate with gas outlet temperatures ~640°C. The objective of this work package is to provide an assessment of the technological limits of the steam generator and its associated heat transport circuit, to identify the main requirements and risks associated with SGU deployment on HTRs and to recommend key areas of improvement and future development to ensure a robust design for process heat application. Individual tasks were performed to provide information on operating and transient conditions, on the development of steam generator design options, on manufacture and inspection issues, and on steam generator thermal hydraulic and structural integrity issues. The picture was completed with a cost/benefit analysis of the chosen lead concept. This WP brings together designers of steam generators and HTR and takes full advantage of industrial developments and advances in conventional technology applications.

A review of Steam Generator operating conditions using information provided by Alstom from their existing experience of SG's in operation and from AREVA on their HTR Module experience was carried out. Thermal data, key requirements and earlier concepts have been described along with material requirements and lifetime issues. The information on the Module includes a description of the primary and secondary side and summarizes the conduct of operations and transients following a main heat transfer malfunction. Later tasks address key aspects of manufacture (tube bundle, including tube to tube-plate joints, etc.), issues of integrity, failure mechanisms, manufacturing risks. Detailed work on these concentrate on the issues associated with the Lead SG Concept chosen within the project which was the Helical design.

The Selection of the Helical design as the Lead Concept was based on an understanding of plant operating / transient conditions and technical / reliability risks associated with the choice of the materials and cost effectiveness of different concepts for producing the required thermal and mechanical performance. The results of the cost analysis also provided an important input in the selection process. The helical-coil tube bundle is capable of accommodating thermal expansion without excessive mechanical stress, has high resistance to flow-induced vibrations, and can be designed to have a high thermal performance. The straight-tube was discarded because of the high loads due to thermal expansion caused by temperature transients, mainly compressive forces developed between the feed and steam headers.

The use of appropriate materials and their data is an important issue and of primary importance are the materials used in the superheater sections, which must withstand the highest temperatures and the target lifetimes of 500k hours. The need to avoid stress corrosion cracking has influenced material selection at lower temperatures, and the shell material is similarly required to withstand prolonged durations, whilst maintaining the pressure boundary due to its connection to the reactor vessel. Mitigating actions have been discussed and proposed to address future materials property needs and associated information for building a (V)HTR Steam Generator Unit.

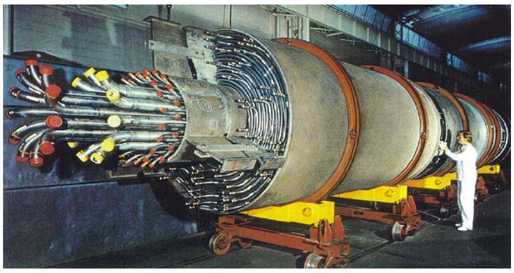


Fig.4-7 HTR Cooler THTR 300

Managing the safety and operational aspects of the Steam Generator requires the implementation of an effective conditioning assessment programme capable of detecting and identifying degradation or developed flaws. Such a programme ensures that the integrity and functional capability can be maintained throughout the whole of the plant service life consistent with the original design intent. For the HTR Steam Generator, aspects of condition monitoring have been provided by Westinghouse using their experience on the THTR Reactor. some initial information has also been provided based on a review of experience on code guidance and capability requirements plus information coming from LWR experience and practice on inspection techniques, monitoring and descriptions of inspection equipment technology. It is considered that certain aspects of the examination and inspection issues for the HTR will be different to those of the LWR because of environmental differences and because of the lead concept chosen. Conclusions are drawn and some issues identified that should be considered when assembling an approach for the HTR Steam Generator tubes and shell structures.

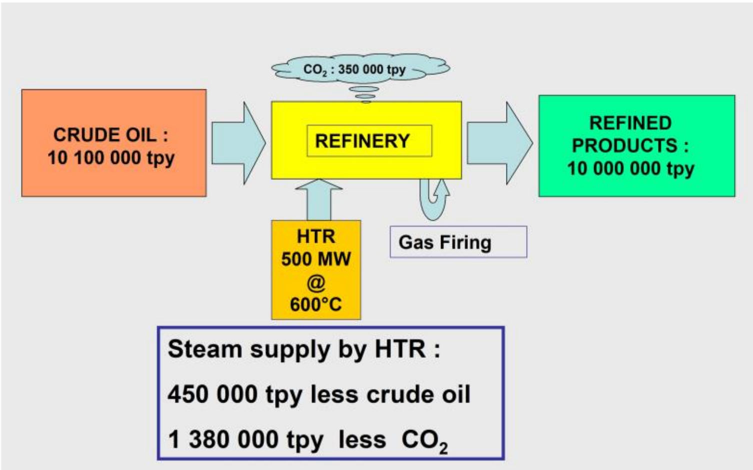
With regard to an investigation of advanced technologies a thermal-hydraulic calculation of the steam generator, a FEM analysis of tube to tube plate joints under transient conditions, and a FEM analysis of a hot gas transition piece were performed to understand needs and issues. The calculations were performed based on the THTR-300 design and the results confirm the acceptability of the arrangement and demonstrate the type of investigations needed to develop and provide an effective design.

Within the Materials and components work of SP4 key activities and results have been performed and achieved to further the advancement of the HTR for cogeneration application completing key areas of technology and understanding and helping to forward the HTR development towards the establishment of a demonstrator for cogeneration.

1.4 Potential impact

Nuclear CHP Markets & Environmental Impacts

Industrial process heat and district heat cogeneration applications span a wide range of temperature, from below 100°C to more than 1000°C. The power requirements remain limited to a few hundred megawatts on a single site, due to the absence of large heat distribution grids. Superheated high quality steam around 600°C is demanded by many industrial processes. Standardized modular HTR can adopt most advances of the conventional power plant technology because both are operating with hot gas as the working fluid and in comparable power ranges. Therefore, modular HTR creates a bridge between conventional and nuclear technologies and may open a new market segment for nuclear energy beyond electricity. An example for integration of nuclear process steam in a refinery is shown in the following figure. Even if only process steam and electricity is provided by HTR to a typical refinery (~10 million tons of crude oil per year) this will result in a reduction of ~ 1.4 million tons of CO2 and 450 000 tons of crude oil for the feed of the refinery.



Effect of HTR process steam supply on a refinery [J. Ruer, SAIPEM, 2009]

A single HTR module of 500 MW_{th} replacing fossil fuel boilers saves 1 million t/yr CO_2 (if replacing natural gas) and even 1.8 million tons CO_2 per year (if replacing coal). For redundancy reasons, several HTR modules are expected to be built on a specific site.

In the medium term, if materials for higher temperature (> 850°C) are available, HTR can be coupled to steam reformers which are currently mainly used for hydrogen production. As a long-term perspective, direct water splitting technologies may be developed for hydrogen production. Then, an even more CO_2-efficient use of HTR can be considered without waiting for the possible direct use of hydrogen as a fuel for transport: Recycling of smoke stack CO_2 from fossil-fired plants, transforming it with hydrogen into synthetic transport fuel, e.g. methanol. In this way, the energy usage per emitted CO_2 more than doubles and a single HTR could save 4.9 million t/yr CO_2. Moreover the use of hydrogen in many other industrial applications, already representing a large and quickly growing market, is expected to grow massively once a competitive CO_2 free hydrogen production process is available.

Feeding the reflections on cogeneration in the SNETP process

The ARCHER project worked towards establishing the technical basis for nuclear cogeneration demonstration. By generic but targeted R&D in the field of gas-cooled high temperature nuclear systems, with the specifications, boundary conditions and requirements of end users available from EUROPAIRS on one side, and the V/HTR knowledge base established in preceding framework programs on the other, many advancement towards demonstration were performed.

The results and insights of the project feed directly into SNETP, more specifically the 'other applications of nuclear energy' pillar represented by the Nuclear Cogeneration Working Group. For the R&D addressed in ARCHER, the project specifically targeted to provide the Nuclear Cogeneration Industrial Initiative (NC2I) with the information required for establishing a realistic and high potential demonstration initiative in the SET-Plan framework.

ARCHER was a technical building block, which via SNETP, and NC2I, intended to support the SET-Plan goals of reduction of greenhouse emissions and fossil fuel dependency.

ARCHER was a technical project, which intended to provide generic technical support to establish this new perspective for both the nuclear and conventional industry communities.
The 'end-users' of the results from ARCHER are not only potential operators of nuclear cogeneration plants but also a multiplicity of manufacturers of materials, systems and components for HTR to be built in and outside of Europe.

1.5 List of Beneficiaries

Website address: www.archer-project.eu

Project type: Collaborative Project
Project start date: 01/02/2011
Duration: 48 months
Total budget: EUR 9,772,783
EC contribution: EUR 5,400,000

Partners:

No	Name	Short name	Country	Project entry month[10]	Project exit month
1	NUCLEAR RESEARCH AND CONSULTANCY GROUP	NRG	Netherlands	1	48
2	AKADEMIA GORNICZO-HUTNICZA IM. STANISLAWA STASZICA W KRAKOWIE	AGH	Poland	1	48
3	ALFA LAVAL PACKINOX SAS	ALP	France	1	48
4	ALFA LAVAL VICARB SAS	ALV	France	1	48
5	ALSTOM POWER LTD	Alstom	United Kingdom	1	48
6	AMEC NUCLEAR UK LIMITED	AMEC	United Kingdom	1	48
7	AREVA NP SAS	Areva	France	1	48
8	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	CEA	France	1	48
9	Furnaces Nuclear Applications Grenoble	FNAG	France	1	48
10	FORSCHUNGSZENTRUM JUELICH GMBH	FZJ	Germany	1	48
11	UCAR SNC - GROUPE GRAFTECH INTERNATIONAL LTD	GTI	France	1	48
12	BRITISH ENERGY GENERATION LTD	BE	United Kingdom	1	48
13	INSTITUT DE RADIOPROTECTION ET DE SURETE NUCLEAIRE	IRSN	France	1	48
14	JRC -JOINT RESEARCH CENTRE- EUROPEAN COMMISSION	JRC	Belgium	1	48
15	WESTINGHOUSE ELECTRIC GERMANY GMBH	Westinghouse	Germany	1	48
16	LGI CONSULTING	LGI	France	1	48
17	PAUL SCHERRER INSTITUT	PSI	Switzerland	1	48
18	SAIPEM SA	Saipem	France	1	48
19	EMPRESARIOS AGRUPADOS INTERNACIONA L SA	EA	Spain	1	48
20	SGL CARBON GMBH	SGL	Germany	1	48
21	ASSOCIATION POUR LA RECHERCHE ET LE DEVELOPPEMENT DES METHODES ET PROCESSUS INDUSTRIELS - ARMINES	ARMINES	France	1	48

22	THYSSENKRUPP VDM GMBH	ThyssenKrupp	Germany	1	48
23	TECHNISCHE UNIVERSITEIT DELFT	TUDelft	Netherlands	1	48
24	TECHNISCHE UNIVERSITAET DRESDEN	TUDresden	Germany	1	48
25	TUV Rheinland Industrie Service GmbH	TÜV	Germany	1	48
26	CENTRUM VYZKUMU REZ S.R.O.	CVRez	Czech Republic	1	48
27	THE UNIVERSITY OF MANCHESTER	UNIMAN	United Kingdom	1	48
28	RHEINISCH-WESTFAELISCHE TECHNISCHE HOCHSCHULE AACHEN	RWTH	Germany	1	48
29	UNIVERSITAET STUTTGART	USTUTT	Germany	1	48
31	Karlsruher Institut fuer Technologie	KIT	Germany	1	48
32	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	FZD	Germany	1	48
33	THE CHANCELLOR, MASTERS AND SCHOLARS OF THE UNIVERSITY OF CAMBRIDGE	UCAM	United Kingdom	1	48
34	CENTRE NATIONAL DE LA RECHERCHE SCIENTIFIQUE	CNRS	France	1	48
35	NOORDWES-UNIVERSITEIT	NWU	South Africa	12	48

2. Use and dissemination of foreground

Section A (public)

This section includes two tables