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Status Report on HTR Research, Development, and Deployment in the USA

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Status Report on HTR Research, Development, and Deployment in the USA

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Abstract – *The Gas-Cooled Reactor deployment effort in the United States is a collaboration between the US government and private vendors. The US Department of Energy sponsors the qualification of fuels, materials, and analysis methods that exploit the capabilities of the national laboratories (mainly Idaho, Oak Ridge, and Argonne) to generate the fundamental fuel, material, and core behavior data needed to support design and licensing efforts by vendors in general. Private HTR vendors provide input to the R&D program and engage in the design of the systems, structures, and components that are unique to their design concepts. Their plant design all exploit the properties and behavior of the TRISO fuel, graphite, and alloys being tested in the DOE qualification programs. This paper will summarize the goals and recent progress of the US Gas-Cooled Reactor Program.*

I. INTRODUCTION

The United States Department of Energy's Office of Nuclear Energy sponsors fundamental research and development of advanced reactor systems through the Advanced Reactor Technologies (ART) Program. The goal of the program is to identify and reduce the technical risks confronting the deployment of non-water-cooled reactors.

The Gas-Cooled Reactor (ART-GCR) Campaign defines, executes and manages research and development (R&D) activities required to design and deploy modular high temperature gas-cooled reactors (HTGR). The Campaign conducts these activities in cooperation with the Nuclear Regulatory Commission (NRC), industrial partners, and universities to support vendor design and deployment of high temperature gas-cooled reactors.

The current ART-GCR campaign is a continuation of the Next Generation Nuclear Plant (NGNP) demonstration project authorized by the Energy Policy Act of 2005. Research and development was on component of NGNP in which a Very High Temperature Reactor (VHTR) was to be built to demonstrate high efficiency hydrogen production. Although the demonstration phase was suspended, R&D continues under ART with activities in Fuel Development & Qualification, Graphite Characterization and Qualification, High Temperature Alloys qualification, and Core Analysis Methods Development and Validation. About US$400M has been spent to date since 2003 on R&D.

The campaign works with industrial vendors to help them achieve their deployment goals. In addition to the basic qualification programs, the ART program solicits input from reactor and fuel vendors on technical issues and research priorities. Through the ART programs, the federal government also provides direct funding for advanced reactor development in a competitive solicitation process.

Recent progress in these areas is summarized in the following sections.

II. FUEL DEVELOPMENT AND QUALIFICATION

II. A. The Advanced Gas Reactor Fuel Program

The objectives of the Advanced Gas Reactor fuel qualification program are to design, fabricate, and test high performance coated particle fuel for use in a graphite-moderated, helium-cooled HTR. The fabrication process was a joint activity between the INL, Oak Ridge National Laboratory, and an industrial fuel vendor, BWXT, Inc. and had the over-arching goal of re-establishing a production-scale TRISO fuel manufacturing capability in the United States. A 7-part program (AGR) of irradiation, safety testing, and post-irradiation examination (PIE) is underway to determine the performance of the fuel under expected operating and accident conditions. New fabrication techniques, special furnaces for heating the irradiated fuel to accident temperatures, and sophisticated analytic microscopy techniques are developed and deployed in support of this effort.

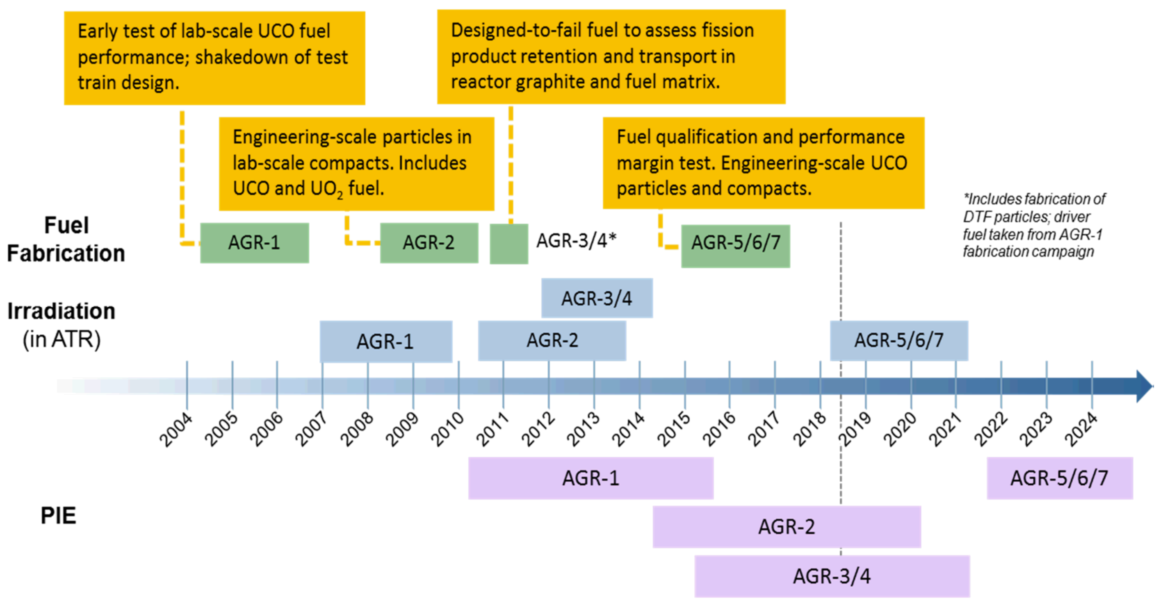

Figure 1: Overview of Fuel Qualification Schedule

The AGR program comprises seven irradiation campaigns with associated fuel fabrication and post-irradiation examination (PIE). **Error! Reference source not found.** shows the schedule for completing t elements of the overall campaign. Each irradiation achieves a specific objective leading toward fuel qualification. The regulator may still require a vendor to perform an additional 'proof test' in which their own fuel design (pebble or prismatic) is tested under expected operating conditions but the generic AGR campaign nonetheless will have generated extensive data on TRISO fuel performance under a broad range of operating conditions.

Irradiation tests 3 and 4 were performed simultaneously in the Advanced Test Reactor (ATR) at Idaho National Laboratory. Tests 5, 6, and 7 were inserted into the core earlier this and are currently generating in-core performance data. PIE continues on tests 2 through 4. All AGR tests contribute to our understanding of fuel behavior and failure mechanisms.

II. B. Accomplishments to date

AGR-1 was designed to establish and demonstrate the methods employed in the full campaign. 300,000 UCO-fueled particles were fabricated under lab-scale conditions and irradiated for close to 3 years, with some particles achieving a burnup of 19.6% FIMA over a range of temperatures. No particle failures were recorded although damage to individual layers was observed (Figure 2). PIE included heating of compacts that simulated the conditions to which they would be subjected during an unmitigated loss of forced cooling event. No TRISO failures were observed in the 1600-1700°C temperature range while failure rates between 1700 and 1800°C were far lower than vendor performance requirements.

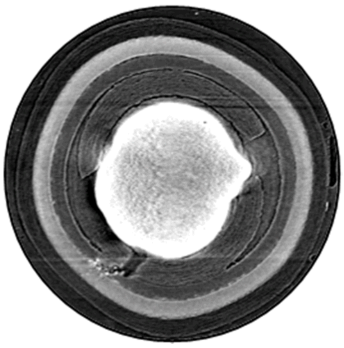

Figure 2: X-ray tomogram of an irradiated AGR-1 particle showing fission product attack of the SiC layer

AGR-2 used particles fabricated at a production scale (by BWXT) and compacted at Oak Ridge National Lab. Both UCO (US) and UO_2 particles (from France and South Africa) were tested, in some cases at temperatures well in excess of expected service conditions (≤1360 °C). Once again, online monitoring detected no failure during irradiation and excellent fission product retention although higher in-pile temperatures resulted in higher release rates of both europium and strontium. During post-irradiation accident testing, AGR-2 UCO particle performance appeared to be superior to both AGR-1 performance and UO_2 performance. A new (but low probability) failure mechanism was detected in which buffer-IPyC adhesion during irradiation can lead to IPyC degradation and SiC failure. This is being investigated further in current PIE work.

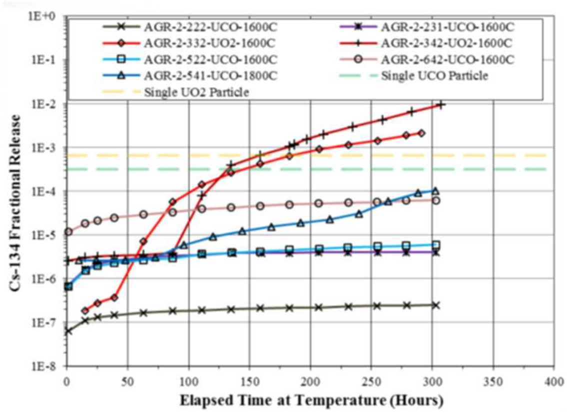

Figure 3: AGR-2 UCO and UO_2 Cs-134 release from compacts

Particles for the AGR-5,6, and 7 tests were fabricated and compacted by industrial fuel vendor partner, BWXT. As one can see in Figure 1, there was a hiatus of a few years between the fabrication of AGR-3/4 fuel that of AGR-5/6/7 fuel. This period of inactivity resulted in a temporary loss of expertise as the highly skilled technical staff were re-assigned to other parts of the company or left it altogether. AGR-5/6/7 fabrication commenced with new staff that lacked some of the knowhow that one acquires through continuous execution of, and feedback from, a complex process. As a result, pre-irradiation testing revealed some parameters, such as exposed kernel fractions, that did not meet program specifications. In the end, overall fuel performance under irradiation is still expected to meet designer requirements and the decision was made to proceed with the fuel as fabricated. The experience, however, underscored the difficulty in developing and maintaining the skills and resources needed for TRISO fuel manufacture.

These struggles notwithstanding, the AGR Fuel Qualification Program is regarded as one of the more successful DOE –NE advanced reactor-focused efforts in recent history. To document and apply the gains made thus far, the ART-GCR campaign is collaborating with the Electric Power Research Institute and HTR vendors to issue a Limited Scope Topical Report on TRISO fuel performance to the Nuclear Regulatory Commission in 2019.

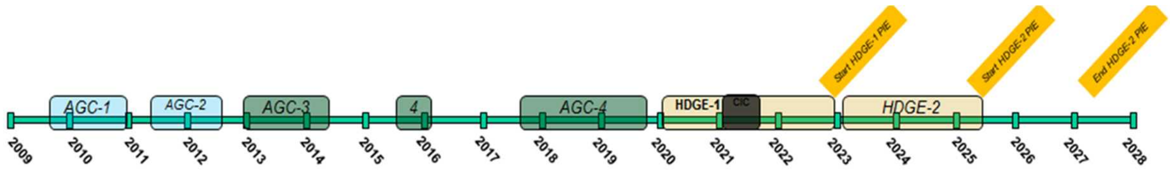

Figure 4: Timeline for AGC Irradiations

The report will focus on the demonstration of intact coated particle performance based upon the AGR-1 and AGR-2 experience. It will establish that the current UCO TRISO particle design meets expectations for in-service and accident condition performance as intended under the AGR program.

II. GRAPHITE CHARACTERIZATION AND QUALIFICATION

The objective of the Advanced Gas Reactor fuel qualification program is to characterize and qualify commercial grades of nuclear graphite for use in high temperature reactors. The scientific and engineering techniques employed encompass all the anticipated tests required to validate and qualify nuclear grade graphite for use in a graphite-moderated HTR. The main elements of the program are: non-irradiated material testing, high dose irradiation experiments, material characterization, and model development.

II. A. The Graphite Program

Commercial grades of graphite exhibit properties that differ from those used in the 1st generation of high temperature reactors. Qualified nuclear-grade graphite is no longer available due to the depletion of the feedstock used in its manufacture. The first step in the qualification process is the baseline characterization of unirradiated grades. Billets of available commercial grades are sectioned into numerous samples on which thermal and mechanical property measurements are performed.

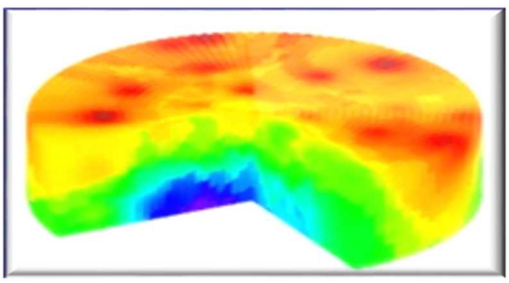

Figure 6: Color map showing property variations in a graphite billet

In the second step, *irradiation performance* data on new nuclear graphite grades at different temperatures, compressive loads, and fluences is taken to support HTR design. The program currently sponsored by DOE is centered on six capsule irradiations in INL's Advanced Test Reactor, which are designated as AGC-1 through AGC-6, followed by PIE of the graphite capsule is designed to measure the thermal and irradiated creep properties of the samples under various compressive loads. The 6 irradiation span the anticipated ranges of dose and temperature. The irradiation schedule is shown in Figure 4. AGC-4 is currently in ATR. Originally, AGC-5 and 6 would have been operated at the very high temperature anticipated in a VHTR. Recently, however, a decision was made to replace those tests with to other tests in which AGC-2 and AGC-3 samples will be re-inserted into the core to achieve very high dose rates at lower temperatures. These operating conditions are more typical of the high temperature reactors that are under development today. The so-called High Dose Graphite experiments will commence in 2020 and will subject some of the samples to sufficient dose to achieve 'turnaround', the dose at which graphite ceases to shrink under irradiation and begins to expand due to excessive displacements of carbon atoms from their nominal crystalline positions. The modified dose/temperature envelope is shown in Figure 5. A small number of samples were irradiated at very high temperature (the HTV test) in the High Flux Isotope Reactor at Oak Ridge National Laboratory (ORNL).

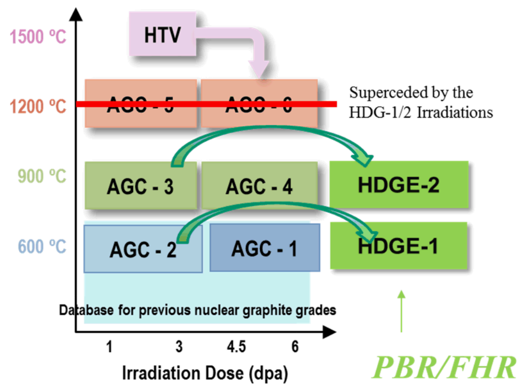

Figure 5: Dose/temperature envelope of AGC Irradiations

Thermal and mechanical properties of the samples will again be measured during the PIE phase; results are being compared to unirradiated graphite properties.

Models of graphite properties, behavior undergoing thermal and radiation creep, and oxidation are being developed at the INL and ORNL. These models will help in the development of American Society of Mechanical Engineers (ASME) and American Society for Testing and Materials (ASTM) codes and standards for graphite. Incorporation of this data into the codes and standards is essential to the design and licensing of the HTR.

II. B. Other accomplishments

In addition to the ongoing irradiations and measurements of irradiated and unirradiated samples, ART scientists and engineers contributed significantly to an (IAEA) Technical Document Discussion of Nuclear Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors. A new moisture-induced chronic oxidation model was developed based upon the experimental data generated in the program.

III. HIGH TEMPERATURE ALLOYS

Alloy Testing and Qualification focuses on two specific areas in support of the near-to-mid-term deployment of high temperature gas-cooled reactors. The first addresses the cross-cutting issues of materials behavior during extended service beyond the scope of the current ASME Code, and the high temperature design and materials issues identified by the U.S. Nuclear Regulatory Commission (NRC) and its Advisory Committee on Reactor Safeguards (ACRS) in support of the licensing process. This entails some additional testing of currently codified alloys (SA508/533 stainless steel at INC800H) to support use at elevated temperatures. The second area corresponds to the incorporation of Alloy 617 into ASME Section III, Division 5 of the Boiler and Pressure Vessel Code.

III. A. Alloy 617 Testing

The ASME Code Case qualifying Alloy 617 for nuclear construction up to 425°C has been fully approved. Balloting of the Code Case for qualifying the material for use up to 950°C and 100,000 hours is proceeding according to a plan that has been developed by the Alloy 617 Code Qualification Task Group. Final approval is expected by 2021. Since a new material has not been approved for nuclear construction in many years the technical committees are not generally familiar with the detailed analysis and background required to develop the sections of the Code Case for elevated temperature construction. As a result, the preferred path forward is to ballot individual topical sections of the Code Case with the appropriate technical groups that have specific expertise in that area, while providing the remainder of the Code Case for review and comment. Sections dealing with time dependent allowable stresses, physical properties, fatigue and creep-fatigue, and calculation of strain limits have all been passed at the Working Group level and are now being balloted at the Subgroup or Committee level.

Experiments have been carried out on the behavior of creep specimens containing two notches with U-shaped profiles such that one notch ruptures and the other remains intact for metallographic analysis (Figure 7). Tests have been completed on specimens with a pair of either small or large radius notches over a range of temperatures from 750 to 900°C where data from smooth bar specimens are available. No detrimental effects from the multi-axial stress states (imposed by the notches) have been observed. To the contrary, the small radius U-notch tests exhibit notch strengthening with a rupture life ~3 times longer than straight gauge specimen tests for similar conditions. The large radius U-notch tests have rupture lives of similar magnitude as the straight gauge tests at similar conditions. Metallographic examination using electron back-scatter diffraction indicates that in the small radius notched specimens the strain is localized very close to the root of the notch, while for the large radius notches the deformation is uniformly distributed across the gage section (similar to smooth bar test specimen).

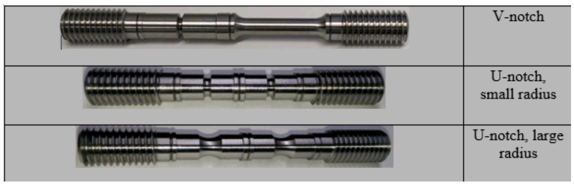

Figure 7: Notch specimens employed in the base metal test program

III. B. Model Development

Along with the testing described above, the ART-GCR Materials staff are developing an integrated Elastic Perfectly-Plastic (EPP) and Simplified Model Test (SMT) methodology to incorporate a SMT data-based approach for creep-fatigue damage evaluation into the EPP methodology. This will eliminate the separate evaluation of creep and fatigue damage and the requirement for stress classification in current methods; thus greatly simplifying evaluation of elevated temperature cyclic service. The purpose of this methodology is to minimize over-conservatism while properly accounting for localized defects and stress risers. Computational analyses and experimental studies have been conducted to support the development of this integrated methodology. Results from the verification and validation simulations of two-bar ratcheting experiments and scaled nozzle-to-sphere tests, conducted previously at ORNL, have provided added confidence that the EPP methodology can be successfully used as a screening tool to evaluate designs for compliance against the ASME Section III, Division 5 strain limits and creep-fatigue criteria. Results from cyclic tests of Alloy 617 SMT specimens conducted at 950°C with long hold times have demonstrated the enhanced creep-fatigue damage resulting from strain redistribution and slowed stress relaxation due to elastic follow up effects. This confirms that the integrated EPP and SMT methodology is required to capture the enhanced creep-fatigue damage in an actual structural component.

IV. CORE DESIGN METHODS AND EXPERIMENTAL VALIDATION

As high temperature reactor development in the US slowed significantly in the 1990's., simulation tools and the data needed to validate models are outdated and do not meet modern design and licensing standards. The HTGR Methods program seeks to rebuild a basic HTGR core simulation capability and validate those tools using data from separate effects and integral test facilities at the national laboratories and universities. Analysis techniques for both pebble bed and prismatic HTGRs are being developed.

IV. A. Experimental validation

In 2005, the Nuclear Regulatory Commission published a Phenomenon Identification and Ranking Table (NUREG/CR-6944) for the VHTR as part of the NGNP project. Volume 2 of the report focused upon accident and thermal fluids analysis. A number of phenomena were identified as being 'Important' with regard to plant safety or not well understood in terms of the data need to characterize them. An experimental plan was developed to address the most important and least understood phenomena through a combination of integral and separate effects tests. Experiments at the INL, ANL, and a number of universities were designed and partially executed to provide data on the following:

- Bypass flow
- Lower plenum flow
- Core heat transfer
- Plenum-to-plenum flow under natural circulation
- Lock exchange flow and air ingress
- Reactor vessel cooling

Later, when the program shifted to support steam cycle power conversion, a moisture ingress 'mini-PIRT' was conducted to identify research priorities needed to address this scenario.

The Nuclear Regulatory Commission also sponsored the construction of an integral experiment at Oregon State University. The High Temperature Test Facility (HTTF) comprises a ¼-scale electrically heated 'reactor vessel' connected to a large tank that simulates the surrounding reactor cavity. Studies of the effects of breaks in the primary coolant boundary and subsequent air ingress will be investigated.

Another integral test, the Natural Circulation Shutdown Heat Removal System (NSTF), was rebuilt at Argonne National Laboratory to investigate the performance of reactor vessel cooling systems.

In 2013 and about the time the demonstration phase of the NGNP project was suspended, the focus of plant designs shifted to near-term deployment of HTRs with lower outlet temperatures (~750 °C). This reduced some of the risks associated with phenomena listed above and the lab-based experiments in bypass flow, lower plenum flow, and other phenomena were cancelled or shifted to universities with sponsorship from DOE's Nuclear Energy University Program. All experiments except the NSTF and HTTF are conducted at universities.

NSTF

The NSTF was initiated to perform large scale experimental testing and generate validation data for passive decay heat removal concepts. Electrically-heated panels simulating the hot pressure vessel.

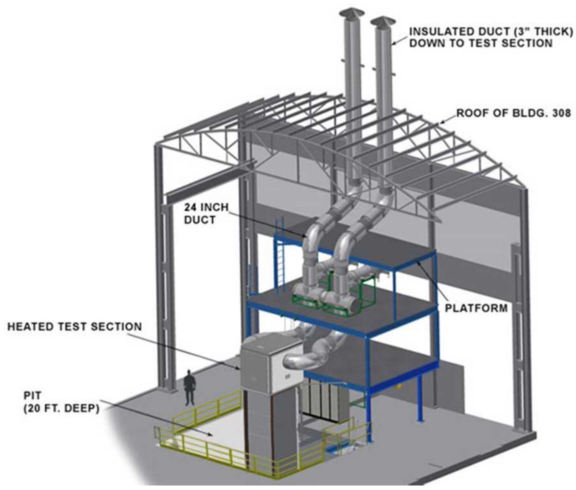

Figure 8: Overview cartoon of NSTF

Radiate heat to riser panels filled with a working fluid that transports the heat to a heat exchanger after which it is rejected to the atmosphere (Figure 8). The original working fluid was air and the facility was scaled (1/2) to the General Atomics MHTGR-350 vessel.

In early 2017, after 33-months of active air-based testing and completion of stated air-based program requirements, the project saw a formal conclusion to the successful air-based testing campaign. In total, 27 tests were conducted to NQA-1 standards, 16 of which were accepted, and included studies of multiple baseline repeats, prototypic accident scenarios, blocked risers channels, power variations, azimuthal and cosine skews, chimney roles, and meteorological variations. The NSTF is now being reconfigured to use water as the working fluid. Figure 9 shows the bottom of a water-cooled riser channel.

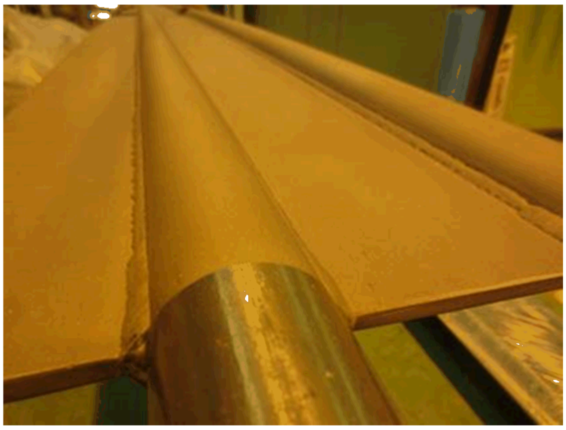

Figure 9: Bottom of the riser channel showing water-filled pipes connected by fins.

HTTF

Construction of the HTTF was competed in 2014 and was followed by shakedown testing. The main vessel contains ceramic blocks with coolant channels and holes to contain the heating elements, as in a prismatic HTR (Figure 10).

During the first set of matrix experiments, problems were encountered with asymmetrical heating and poor connections in the graphite rodlets that compose the core heating elements. This necessitates a re-design of the heater rods and replacement of some blocks and instrumentation. The rebuild is close to completion and testing is expected to resume early in 2019.

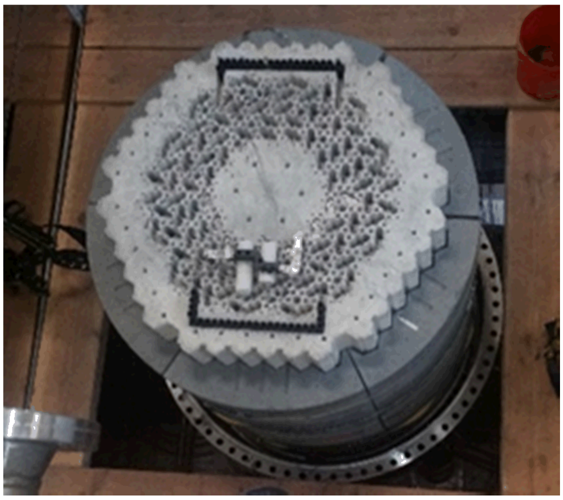

Figure 10: Top view of HTTF core internals

Meanwhile, a number of separate and mixed effects tests have been and continue to be conducted a US universities in support of the gas-cooled reactor campaign. For example, to investigate hot helium flow inside core cooling channels and emerging as plumes or jets into the upper plenum (Figure 11 and Figure 12Figure 11), experiments have been performed at City College of New York and Texas A&M University. These provide high fidelity data for validating computational fluid dynamics models of plenum-to-plenum heat transfer during a loss-of-forced flow event.

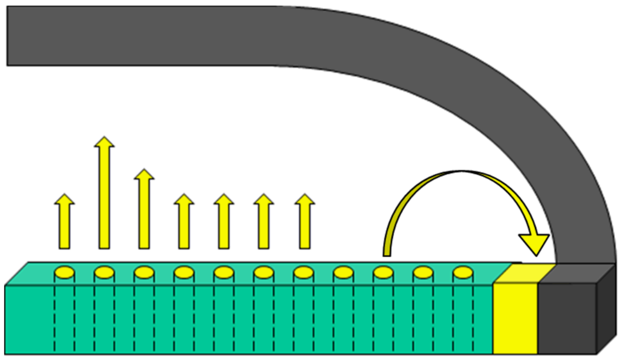

Figure 11: Cartoon illustrating helium emerging into the upper plenum and possibly returning through a riser channel

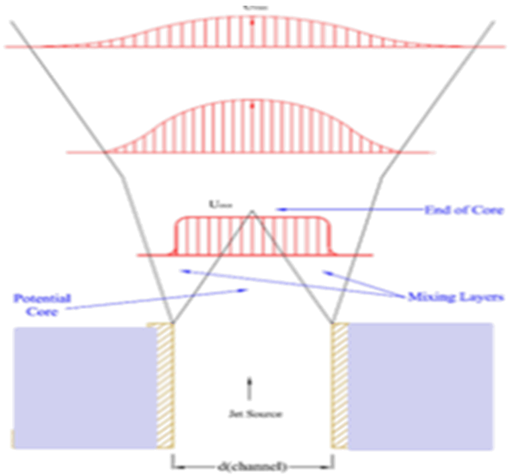

Figure 12: Evolution of a coolant jet into an upper plenum plume

Understanding of such phenomena can help designers prevent local heating of upper plenum structures during accident scenarios.

III. B. Core Simulation Methods Development and Demonstration

This effort focuses on developing and testing engineering-scale analysis tools for prismatic and pebble bed design. The emphasis is upon functionality and system behavior, not fidelity. The goal is to simulate with confidence and efficiency typical burnup and transient calculations such as Anticipated Operating Occurrences (AOOs) and Design Basis Accident sequences initiated from different core burnup states are used. The ability to quantify uncertainties in these calculations is of particular importance.

To this end, the ART-GCR campaign supports involvement in various international benchmark projects (OECD [1] , IAEA [2] , and Generation IV VHTR) and bilateral cooperation with the Japan Atomic Energy Agency. These project provide a focus for the modeling efforts, illuminate gaps in capabilities, and provide valuable feedback on user and modeling effects through comparisons to results from the other participants.

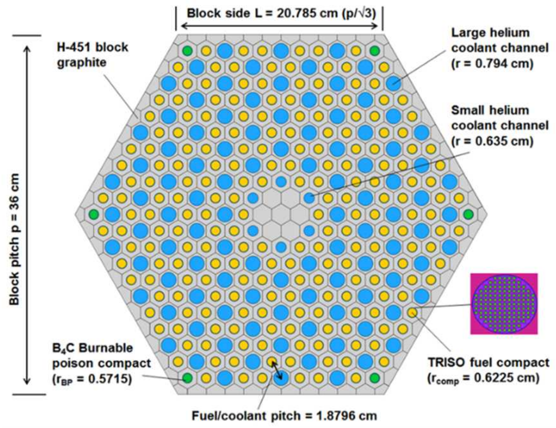

Figure 13: Model of a fuel block used in the IAEA Coordinated Research Project on Uncertainty Analysis in HTRs, Exercise I-2A

For example, the (currently suspended) OECD HTTR LOFC project, in which three loss-of-forced-cooling (LOFC) experiments will be conducted on Japan’s High Temperature Engineering Test Reactor, provides a formidable test case for modeling a severe core transient in a partially burned core as the time constants of the neutronic and thermal fluid transient phenomena are widely different and the time and magnitude of the re-criticality are sensitive to a range of phenomena. The PHISICS-RELAP5-3D core analysis code has been applied to this problem and, after a few years of development and testing, plausible results are being obtained. Design and safety analysis of generic HTR operating scenarios can now be attempted with some confidence and with limited uncertainties.

V. LICENSING

ART Regulatory Affairs staff have engaged the regulator on licensing gaps and issues since the early days of the NGNP Project. Under NGNP, a series of White Papers addressing key HTR safety issues was submitted to the NRC. NRC staff have reviewed and commented on many of them including the issuance of draft guidance late in 2017 on the requirements for functional containment.. More recently, the INL worked with the NRC to develop General Safety Design Criteria for advanced reactors as well as Safety Design Criteria for HTRs. Earlier this year, the NRC issued Regulatory guide 1.232, "Guidance for Developing Principal Design Criteria for non-Light-Water Reactors."

VI. INDUSTRIAL ENGAGEMENT

Direct involvement by commercial HTR vendors has been an important part of the gas-cooled reactor R&D campaign since the beginning of the NGNP Project. Vendor input has been crucial to identifying design data needs, setting research priorities, assisting in experiment planning, and evaluating results in the context of licensing and cost-effective design. Industrial input was important in the decisions to:

- use UCO fuel instead of UO2, to reduce the outlet temperature from 1000 ºC to ~750 ºC,
- focus on the ASME codification of Alloy 617 for high temperature structural applications,
- Pursue high dose, lower-temperature irradiation of graphite useful to designs currently being pursues, and
- Address moisture ingress phenomena in the experimental validation task

among other research directions taken. Recently, vendors have been asked to provide input and feedback on Technology Roadmaps commissioned by the Department of Energy [3]. The HTR Roadmap lays out the important tasks and logical pathway to demonstration of a high temperature reactor by 2030.

Five reactor vendors (Framatome, X-Energy, U-Battery, StarCore Nuclear, and UltraSafe Nuclear) all intend to use data and knowledge generated by the ART-GCR Campaign in their design and licensing activities. This is important in developing HTRs for an uncertain market. The designs being pursued are diverse and include: a 625 MWt prismatic (Framatome), a 200 MWt pebble bed (X-Energy), and very small reactors for special markets (StarCore, USNC, and U-Battery). Indeed, one molten-salt-cooled concept being developed by Kairos Power will rely on the TRISO fuel and graphite grades being tested in the program. The diversity in power conversion (electricity, gas turbines, industrial steam, etc.) also reflects both the versatility of both the concept in general and the specific designs being pursued.

VII. SUMMARY

The Department of Energy's Gas-Cooled Reactor Campaign seeks to resolve the primary technical issues confronting the deployment of the high temperature gas-cooled reactor. The campaign focuses on TRISO fuel qualification, graphite characterization and qualification. Codification of new alloys for use in primary loop structures, and the development of codes and the validation of core neutronic and thermal fluid models.

In the past 2 years, the AGR Fuel Program has made considerable progress in understanding TRISO fuel performance and failure mechanisms through a state-of-the-art PIE campaign. The fuel for the final AGR Irradiation tests was fabricated and delivered by our industrial fuel partner BWXT and is now being subjected to a radiation field in ATR. The 4^{th} of 6 graphite irradiations tests is underway and an extensive set of graphite thermos-mechanical properties has been catalogued. A code case for using Alloy 617 in primary loop structures has been submitted to the ASME for approval. Balloting is underway in the first such effort since the 1990s. A series of reactor cavity cooling experiments using air has been completed and the facility is being reconfigured for water-cooled testing. Integral testing of vessel behavior during a major primary coolant leak is slated to commence in 2019. Coupled core models of steady state and transient behavior are being built and executed using modern computational tools.

These efforts and others are laying the groundwork so that a strong and competitive high temperature reactor industry can emerge in the coming decades.

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