AREVA-16-03616
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Subject: AREVA NP Advanced Reactor R&D needs

Dear Dr. Corradini,

I apologize that AREVA could not attend the recent NEAC meeting on Industry Needs for Advanced Reactor R&D. AREVA is pleased to take this opportunity to provide input to NEAC.

While there is not a pressing need for a new test reactor to support AREVA's current SC-HTGR technology, we do have some suggestions for what such a reactor should be able to do to support more long-term developments.

For example, we believe a test reactor should be able to handle large composite structures, which will be helpful down the road as we consider very high temperature reactors (VHTRs). This would require the following characteristics:

- Large helium loop
- Minimum test article size on the order of 0.5m x 1.0m x 0.5m (this large size is necessary to support irradiation testing of future composite structure segments that are not amenable to scaling from small material samples)
- Thermal neutron spectrum (slightly harder than LWR spectrum due to graphite moderator)
- Irradiation temperature range 400 C to 1000 C

It is conceivable that these VHTR data needs could be satisfied by operating steam cycle HTGRs in the 2030 timeframe, if the temperature requirement can be dispositioned.

Another example would be the capability to test more aggressive LWR accident tolerant fuels which might not be amenable to testing as lead test assemblies in the current LWR fleet. This would require the following characteristics:

- Large high pressure water loop
- Minimum test article size on order of 0.05m x 0.05m x 1.0m
- Thermal neutron spectrum (LWR spectrum)
- Irradiation temperature range 250 C to 350 C

More specifically, the needs for our next generation of HTGR reactors (the VHTGR) were published as design data needs developed as part of our contribution to the NGNP pre-conceptual project (Attachment 1). In addition, under the DOE Advanced Test and Demonstration Reactor (AT/DR) project AREVA provided the design attributes and considerations for a gas-cooled graphite moderated irradiation test reactor (Attachment 2)

The time line for our VHTR material R&D and Data needs is 15-to-30 years following a successful commercialization of our near term Steam Cycle HTGR.

I hope this information proves useful to you in your studies.

Warmest regards,

M V Parece

Martin V. Parece
Vice President & Chief Technology Officer
AREVA Inc.

Attachment 1 – NGNP PCDR Chapter 19 – Research and Development
Attachment 2 – Design Attributes and Considerations for a Gas-cooled Graphite Moderated Irradiation Test Reactor

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NGNP with Hydrogen Production

Preconceptual Design Studies Report

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19.0 RESEARCH AND DEVELOPMENT (R&D)

The Very High Temperature Reactor (VHTR) is uniquely suited for producing hydrogen without consuming fossil fuels or emitting greenhouse gases. Successful deployment of an advanced VHTR will depend to a large extent on the research done to anticipate and address the large number of technical risks and, at the same time, to demonstrate the advantages of the VHTR.

Although the VHTR is an unprecedented first of a kind (FOAK) system, the basic technology for the next generation nuclear plant (NGNP) has been established in former high temperature gas-cooled reactor plants (DRAGON [England], Peach Bottom Unit 1 [U.S.], Arbeitsgemeinschaft Versuchsreaktor (AVR) [Germany], Thorium Hochtemperatur Reaktor (THTR) [Germany], Fort St. Vrain (FSV) [Colorado]). In addition, the technologies for the NGNP are being advanced in the Gas Turbine-Modular Helium Reactor (GT-MHR) project, and the Japanese HTTR and Chinese HTR-10 projects are scaled reactors demonstrating the feasibility of some of the planned NGNP technology and materials. Further research and development (R&D) is needed to increase coolant temperature beyond 850 ºC and to develop the interface between the Nuclear Heat Source (NHS) and the heat utilization systems.

The objectives for R&D needs are to identify and characterize the needs for R&D work to mitigate technical risk and to resolve critical issues affecting design, fabrication, testing and operation of the VHTR plant. The systems of principal interest in this effort are the nuclear heat source (NHS) / nuclear island and the power conversion system (PCS). Creation of an R&D development plan is beyond the scope of this effort.

19.1 Approach to Define R&D Needs

The general approach for defining applicable R&D needs for the NGNP is shown pictorially in Figure 19-1. First, the objectives and scope of VHTR hardware and analytical computer codes were determined from the work breakdown structure (WBS). Next, Subject Matter Experts (SMEs) were surveyed to determine current technology maturity, R&D needs to mitigate technical risk and/or resolve critical issues, prioritize R&D needs, estimate cost and schedule, and identify facilities to perform the R&D. An adaptation of the aerospace Technology Readiness Level (TRL) approach was used to define technological maturity. The "Importance" and "Knowledge" parameters of the Phenomena Identification and Ranking Technique (PIRT) were used to prioritize R&D needs. The surveys were then compiled, compared to previous applicable VHTR work, and iterated for completeness and consistency. The risks and risk mitigation approaches identified in Section 18.0 were then used to confirm that all R&D needs have been identified. A similar approach, using a modified survey, was taken to identify R&D needs for computer codes and models.

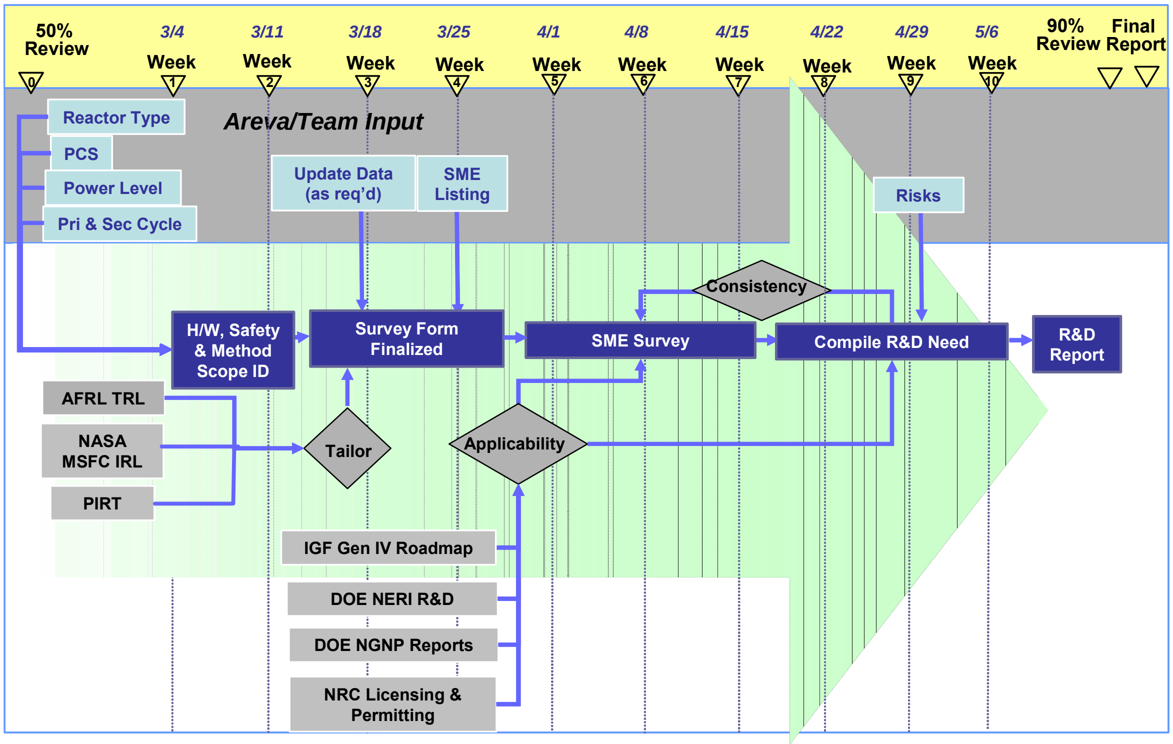


Figure 19-1: Work Flow to Identify R&D Needs.

19.1.1 Technology Readiness Level (TRL)

A technology readiness level (TRL) scale, such as that used in the aerospace industry, was adapted for use in assessing VHTR technical maturity [27], [28] .The original NASA TRL approach offers a method of subjectively quantifying the maturity of certain technologies for use in the space program. It provides a "snap shot" of program maturity at a given time. TRLs range from Level 1: Concept Conceptualized to Level 9: Mission Proven. Efforts to correlate TRL to the Nuclear Technical Maturity Assessment have been adapted [29]. This is illustrated in Table 19-1, where the TRL for the NHS and PCS are mapped to the traditional aerospace TRL. It was further tailored for identification of maturity of computer codes and design methods for the VHTR Table 19-2.

Table 19-1: VHTR NHS and PCS TRL Mapped to Aerospace TRL

TRL	Aerospace Technology
9	Has an identical unit been successful on an operational mission (space or launch) in an identical configuration?
8	Has an identical unit been demonstrated on an operational mission, but in a different configuration/system architecture?
	Has an identical unit been mission (flight) qualified but not operationally demonstrated (space or launch)?
7	Has a prototype unit been demonstrated in the operational environment (space or launch)?
6	Has a prototype been demonstrated in a relevant environment, on the target or surrogate platform?
5	Has a breadboard unit been demonstrated in a relevant (typical; not necessarily stressing) environment?
4	Has a breadboard unit been demonstrated in a laboratory (controlled) environment?
3	Has analytical and experimental proof-of-concept been demonstrated?
2	Has a concept or application been formulated?
1	Have basic principles been observed and reported?

TRL	VHTR PCS Technology
9	Has an identical unit been successful on a commercial operation in an identical configuration?
8	Has an identical unit been demonstrated on a commercial operation, but in a different system/configuration architecture?
	Has an identical unit been successful on a pilot plant?
7	Has a prototype unit been demonstrated in the operational environment with demonstration of safety features?
6	Has a prototype unit been demonstrated in a relevant environment, on the target or surrogate platform?
5	Has component/breadboard been demonstrated in a relevant (typical; not necessarily stressing) environment?
4	Has component/breadboard been demonstrated in a laboratory (controlled) environment?
3	Has analytical and experimental proof-of-concept been demonstrated?
2	Has a concept or application been formulated?
1	Have basic principles been observed and reported?

TRL	VHTR NHS Technology (Process & H/W Equipment)
9	Process integrated into operations /Equipment is commercially available or proven and in use
8	Hot Prototype off-design (for safety) demonstrated
7	Hot Prototype design basis demonstrated
6	Cold Prototype demonstrated
5	End-to-end design (flowsheet / equipment) completed
4	Hot feasibility demonstrated
3	Cold feasibility demonstrated
2	Design concept or technology application formulated
1	Identification of new design

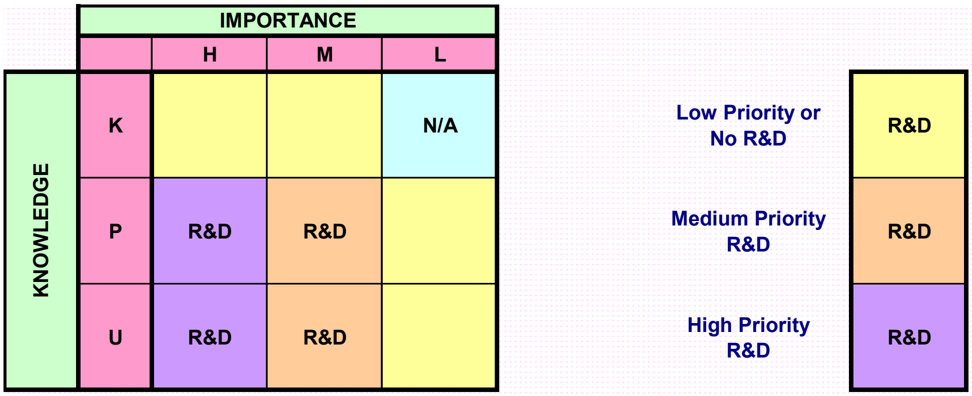
Table 19-2: VHTR Computer Code Maturity TRL Definitions

TRL	VHTR NHS Technology (Computer Code / Modeling)
9	Computer code / model proven & in use for identical applications
8	Computer code / model fully verified and validated for applications; Existing model used for different, but in scope application
7	Computer code / model verified and validated only for AOO and DBA scenarios
6	Integrated modeling (in Prototype) completed
5	Individual module modeling completed and validated at simulated operating environment
4	Individual module modeling at laboratory environment completed
3	Proof-of-concept demonstrated at laboratory environment
2	Modeling concept formulated
1	Basic principles know

19.1.2 Priority of R&D Needs

Ranking of R&D needs ultimately must consider many factors including: required time to complete the R&D, when the results are needed, cost and other resource requirements to perform the R&D, and probability of success. Detailed ranking, considering all these factors, must be performed within the context of developing the overall R&D plan. This is beyond the scope of the current effort. A knowledge-based priority ranking has been developed based on the matrix in Figure 19-2. An additional ranking reflecting R&D urgency should be developed as a next step based on when the R&D results are needed and the estimated duration of the R&D.

The Phenomena Identification and Ranking Technique (PIRT) is a systematic way of gathering information from Subject Matter Experts (SMEs) and ranking the importance of that information, in order to meet some decision-making objective. Among the various PIRT parameters, "Importance" and "Knowledge" were adapted to prioritize R&D needs. This approach is shown in Figure 19-2. From the identified "Importance" and "Knowledge" levels, one can prioritize R&D needs from low to high.



IMPORTANCE

High (H)	**Implies that the phenomenon/model/parameter has a controlling impact on Figure of Merit (FOM). Simulation of experiments and/or analytic modeling with a high degree of accuracy is critical.**
Medium (M)	**Implies that the phenomenon has a moderate impact on the FOM. Only a moderate degree of accuracy is required for analytic modeling or measurements.**
Low (L)	**Implies that the phenomenon has a minimal or zero impact on the FOM.**

KNOWLEDGE

Known (K)	**Implies fully or almost fully known (more than 75% of what we could expect to know).**
Partially Known (P)	**Implies the knowledge base is moderate (25-75% of the knowledge base is established).**
Unknown (U)	**Implies that the knowledge base is low (less than 25% of the knowledge base is established).**

Note: No assessment of knowledge is given if the importance is low (NA, not applicable, is given).

Figure 19-2: PIRT Approach to Determine R&D Needs and Priorities

19.1.3 Survey Forms

The hardware survey form and the software survey form are shown in Table 19-3 and Table 19-4 respectively. The hardware survey form is applicable to VHTR hardware such as fuel, materials, component, and power conversion system development and qualification. The software survey form is applicable to VHTR computer codes and methods development and validation.

Table 19-3: VHTR Hardware Survey Form

VHTGR H/W Survey Form					
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*					
WBS # (Reference)		WBS Title			
Subject Matter Expert Name		Email		Phone	
Organization					

Current Technology Readiness Level Identify TRL, including rationale		
TRL		*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant	
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*	
1	

Which Program Phase Will This R&D Support? (please check one)			
Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Table 19-4: VHTR Computer Codes and Models Survey Form

VHTGR Computer Code / Modeling Survey Form					
NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*					
WBS # *(References)*		WBS Title			
Subject Matter Expert Name		Email		Phone	
Organization					

Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version			
Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Status of Computer Code Please check applicable				
Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code

Current Technology Readiness Level Identify TRL, including rationale		
TRL		*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach) Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant						
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*						
1						
Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

19.2 R&D Needs

Application of the approach described in Section 19.1 resulted in the identification of the R&D needs that are discussed in this Section. Individual survey forms appear in Appendix C; and a summary of TRL, importance, knowledge, and estimated cost and schedule for each R&D item is listed in Table C.2 of the Appendix.

The R&D needs that are key to the success of the project include:

- Fuel development and qualification, particularly irradiation and testing of compacts and mass production processes. R&D costs in this area are about $210 million.
- Materials development and qualification. This covers certain high-temperature steels, composites, and graphite selection/qualification. The associated R&D costs are estimated at $33 million.

- Components testing. A large (10 MW) helium test loop is required for prototype tests of components. This loop could cost as much as $110 million. An additional $50 million would be needed for actual hardware tests (includes a smaller 1 MW test facility).
- Computer codes & methods development/qualification. Included here are neutronics, fuel performance, heat transfer, and mechanical analysis codes. The total R&D expense is estimated at over $26 million with $8 million associated with neutronics code benchmarking to critical experiment data.
- Power Conversion System. This work covers nitriding tests and improvement of blade performance. The associated R&D costs are estimated at $10M.

In total, the R&D program is expected to cost about $440 million and span 60 months

19.2.1 VHTR Fuel Development and Qualification

Historical programs have demonstrated the feasibility of TRISO fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Irradiation testing is currently ongoing with preliminary UCO kernels (contained in particles) produced in a pilot-scaled facility. However, the production capabilities do not exist in the U.S. infrastructure. Two risks (Risk D-005 & D-006) have been identified for fuel development. These are insufficient funding and unavailability of a test reactor and fuel inspection facilities to support fuel development on a schedule required to meet the NGNP operational date of 2018.

Fuel Kernel

Currently the VHTR TRISO kernel TRL is 4, and three “High” to “Medium” priority R&D needs for kernel materials and manufacturing have been identified.

1. The first is development of an advanced carbon source for UCO kernel production. It is estimated that it will require $5M to $10M and 12 months to produce a carbon source and test materials in a UCO kernel fabricating pilot-facility.
2. Second is development of an advanced kernel wash and dry system to cost effectively increase throughput of the kernel line without degradation in kernel quality.
3. Third is development of enhanced fuel sintering for either UCO (large fluidized bed sintering) or UO2 (static bed sintering) focusing on increased throughput and reduced cost. Costs and schedule are estimated at $15-$20M and 24 months for this development effort and the preceding effort (2).

Facilities at BWXT have been identified for performing fuel kernel R&D.

Coating

Extrapolation of fuel coating parameters and performance results for the existing six inch coater to production scale has been identified as Risk D-009. Currently the VHTR TRISO coating manufacturing process is at TRL 4 and the R&D priority is ranked as “High”. Coating materials qualification R&D need is included in the following section on compact materials.

This R&D item will examine whether the maximum coating batch capacity of the six inch coating retort that currently exists at BWXT, will economically support VHTR fuel production. Should a larger coater be required, R&D should be performed on the new coater and this would require a facility expansion. Cost estimates range from $5M, if the existing facility will support production, to $20M, if a new facility is required. It may take up to 24 months to complete the new facility.

Compact

Compact fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale. However, currently-available materials have not been irradiated and performance under relevant environment has not been demonstrated. As a result the compact TRL is now at 3. Further, the priority of this R&D need was evaluated as “High”.

It has been estimated that it would take 36 months and $60M to select a graphitic matrix, resin, etc. to produce thermosetting compacts and to demonstrate the performance under normal and off-normal accident conditions.

In addition, it is essential to establish compact manufacturing capabilities in the U.S. and it is recommended that it be based on the AREVA process. The other three compact R&D needs include:

1. Testing to confirm compact pressures and temperatures in order to minimize fuel damage.
2. Development of the heat treating process to ensure complete graphitization of the matrix material.
3. Perform irradiation tests on compacts to demonstrate performance for nominal and off-nominal operating conditions.

These four R&D needs will cost $40M and 36 months schedule. Expansion of BWXT fuel line for compacts is recommended.

<u>Inspection and quality control methods</u>

Several inspection techniques are available for fuel kernels, particles, and compacts. However, a strong correlation between as-fabricated and inspected particles and compacts and irradiation performance has not been shown in all cases. It has been identified as an ”Avoid” risk with “Likely” occurrence and “Crisis” consequence for unavailability of fuel inspection facilities. Consequently, the first three fuel inspection and quality control R&D needs below have been categorized as “High Priority” while the last is “Low Priority.” These R&D needs are:

1. Development of QC inspection techniques that directly relate to irradiation performance.
2. Development of techniques for large-scale production capabilities that minimize the quantity of materials that require destructive evaluation to ensure statistically acceptable fuel is produced. Techniques to be investigated could be: microfocus x-ray of particles (dimensional inspection of particle layers), mercury porosymetry (buffer density), sink-float (IPyC, SiC, and OPyC density), anisotropy measurements of the IPyC and OPyC layers, etc.
3. Irradiation testing of the compacts to ensure that as-measured attributes actually correlate to performance. This would be necessary to ensure the correct attributes are being measured and characterized.
4. Development of highly reliable instrumentation and data acquisition software will be needed to ensure fuel particle quality is built into the fuel.

These R&D needs are estimated to cost $27M and require 36 months.

As for quality control methods, most are currently available for pilot-scale fuel production with a higher TRL of 6.

<u>Fuel mass production</u>

Many areas of the fuel fabrication process have been demonstrated on a pilot-scale. However, some chemical processing areas or the process will require significant scale-up to meet production demands. This scale-up is not expected to be linear and product quality must be demonstrated on the larger scale. As a result, the TRL for fuel production is as low as 3.

Three "High Priority" R&D needs have been identified. These are:

1. The "scale-up" R&D should focus on kernel wash and dry, sintering, coating (assuming larger than 6" coater is required), compact matrix formulation, and compact fabrication.
2. QC techniques need to be developed with mass production in mind (please see the previous paragraph).
3. Irradiation testing will be required to confirm that fuel performance matches performance from the laboratory/pilot facilities.

It is estimated these R&D needs will cost $30M and require 30 months. The existing facilities at BWXT can be modified to develop larger-scaled production.

19.2.2 Materials Development and Qualification

The materials R&D needs will focus on testing and qualification of the key materials commonly used in very high-temperature designs. The materials R&D will address the materials needed for the VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant.

There are five risks identified in the area of materials, though none has score greater than 6 (Risk Critical). There is a general understanding that the materials R&D is essential for VHTR success.

In addition, the VHTR design relies on contact conditions between different materials (metal to metal, graphite to ceramics, ceramics to metal, etc.) and R&D actions have to be performed to assess the contact conditions to avoid unexpected situations (bonding, wear, etc). As an example, the core support to reactor vessel interface is currently assumed to be a sliding interface. R&D actions are required to make sure that the helium environment (together with the contact pressure) is not likely to create a bonding effect between the alloy 800H and the 9CR1Mo materials. Tribology tests are needed on expected couples of materials in representative VHTR conditions. Dedicated facilities, for example facilities at AREVA NP and CEA, will be required. These tests were estimated for $0.5M and 18 months.

The materials development and qualification R&D needs discussion is grouped into three areas: metallic, ceramic, graphite materials.

19.2.2.1 Metallic Materials

The primary candidate for vessel materials is Modified 9Cr1Mo steel. Alloy 800H is considered for internal materials. Modified 9Cr1Mo is also a candidate only if the temperature is kept well below 750°C. As for the IHX, superalloys such as In617 or Haynes 230 are candidate materials.

Mod 9Cr1Mo has already been used in conventional power plants and is also supported by significant R&D test results from past Fast Reactor R&D programs. An R&D program has already been launched in the context of HTR ANTARES activities to complete the required input data for the final selection and the qualification program. The TRL for Mod 9Cr1Mo is 6.

For Mod 9Cr1Mo steel the R&D needs, of "High Priority," include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability (Risk C-001), emissivity, negligible creep conditions and creep fatigue. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification. It has been estimated that the qualification of Mod 9Cr1Mo will take approximately 72 months and $4M due to the need of procuring a large forging. Due to the long lead procurement time of Mod 9Cr1Mo forgings a risk (Risk P-002) has been identified.

Mod 9Cr1Mo is covered by the ASME code up to 371ºC in Subsection NB and beyond 371ºC in Subsection NH. Subsection NH does not currently cover heavy section products (Risk L-006) and needs to be updated to cover specific aspects of Mod 9Cr1Mo. Actions have already been launched in the context of the DOE/ASME Gen IV material project to provide basis for code development. R&D efforts to support this codification should be continued.

In view of past experience in gas cooled reactor, alloy 800H is a prime candidate for metallic internals operating in cold helium. Moreover, efforts are in progress to extend its coverage up to 850°C in ASME III-NH (Risk L-005 The TRL for alloy 800H is 8.

For 800H alloy the R&D needs include:

1. Emissivity measurement under likely representative state of surface (as machined and oxidized after machining) and
2. Corrosion behavior under representative primary helium environment.

For extension of alloy 800H coverage in ASME III-NH the following items are needed:

1. Long term tests at temperature higher than 760°C,
2. Tensile tests at temperature higher than 870°C and
3. Extension to cover 60 years lifetime.

Two available nickel-based super alloys (In617 and Haynes 230) have been selected as structural materials for the IHX: In617 (NiCr22Co12Mo), which has been widely studied in the early 80's for HTR application and Haynes 230 (NiCr22W14), which has been developed more recently but it exhibits better corrosion resistance. An extensive research program has been launched in France within the framework of the ANTARES program to evaluate mechanical properties, thermal stability, and corrosion resistance in the temperature range of 700 °C to 1000 °C for extended periods. Currently the TRL of In617 is 6.

In617 and Haynes 230 R&D needs, of "Medium Priority," have been identified to address the following issues:

1. baseline mechanical property data, including creep-fatigue data,
2. long-term thermal stability,
3. effects of helium coolant chemistry on material degradation,
4. effects of 80% nitrogen-20% helium mixture on material degradation and
5. corrosion effects on mechanical properties.

The In617 and In230 R&D efforts will cost $4M and require 30 months to complete.

19.2.2.2 Ceramics

No nuclear components or structures made of composites were used for the past HTRs or for other reactor concepts. The use of composites is driven by their high resistance to high or very high temperatures (Risk D-002). An R&D program has been launched in the frame of ANTARES to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures. For the aforementioned applications, the ceramic TRL is currently at 7.

The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of material properties such as:

1. thermal-physical properties (thermal conductivity (K), coefficient of thermal expansion (CTE), heat capacity (Cp)),

2. mechanical properties including multiaxial strength,
3. fracture properties,
4. fatigue properties and
5. behavior in an oxidizing atmosphere and oxidation effects on properties.

In addition, for thermal insulation, ceramic materials qualification should be for:

1. thermal-physical properties (K, CTE, Cp) and
2. behavior under oxidation.

No control rods made of composites were used for past HTRs, or for other reactor concepts. The use of composite C/C for control rods has a low TRL of 2. Other composites such as C/SiC are also envisioned. An R&D program has been launched in the frame of ANTARES to explore the possibility of employing such composites for the control rods. SiC/SiC composites are not considered mature enough to meet the NGNP 2018 schedule.

Additional tests for control rod ceramic materials include:

1. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep and
2. tribology.

The R&D needs for ceramics are of "Medium Priority." Total cost was estimated at $4M and the schedule was 54 months.

19.2.2.3 Graphite Materials

Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures (Risk D-013). Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available (Risk D-016). An R&D program has been launched within ANTARES program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. The TRL of graphite materials is 7.

Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes:

1. thermal-physical properties (K, CTE, Cp, emissivity),
2. mechanical properties including multiaxial strength,
3. fracture properties,
4. fatigue properties,
5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep,
6. behavior under oxidized atmosphere including oxidation effects on properties (Risk D-014) and
7. tribology.

Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed. The R&D needs for graphite materials are of "High Priority." Total cost was estimated at $20M ($6M for the preliminary phase and $14M for the detailed phase) and the schedule is greater than 54 months.

Development of ASME and ASTM codes and standards for graphite is essential for timely application graphite for NGNP reactor (Risk L-007).

19.2.3 Components

R&D needs have been identified for the following nuclear heat source / nuclear island subsystems / components: Circulators, IHX (Tube), IHX (Plate), Isolation Valves, Fuel Handling System, Neutron Control System Drive Mechanism, RCCS, Plant and Safety Protection, Hot Gas Duct and Instrumentation.

The components of the Helium Purification System and the Shutdown Cooling System have been evaluated and no R&D needs have been identified due to similar subsystems currently in use, or were used, in various other helium cooled reactors. Qualification of the helium purification charcoal can be performed during the commissioning phase.

Circulators

Circulators up to 4 MWe have already operated in HTR reactors. The test program is dedicated to component qualification during the commissioning phase rather than as an R&D task. Planned tests ("Low Priority") include:

1. Air tests of the impeller (at scale 0.2 to 0.4).
2. Helium tests of magnetic and catcher bearings.
3. Tests of the circulator shutoff valve.
4. Full scale integrated tests.

IHXs

The R&D inputs are based on two IHX concepts: Tubular IHX for 193 MWt power conversion and Plate IHX for 60 MWt loads for hydrogen plant loop.

Small test facilities up to 1 MWt are available. Large test facilities of about 10 MWt will need to be designed and built (Risk D-015). It is estimated that it will require $20M and 30 months to build a 1 MWt test loop and $80 to $120M to build and test a 10 MWt test loop: $72M to $112M for the facility, $1M for the test article, and $7M for the test.

Tubular IHX

The Tubular IHX design is based on the extrapolation of past German experience. NGNP requirements lead to high temperature operation with an innovative secondary fluid mixture of helium and nitrogen. Risk D-012 identifies feasibility concerns on module size, temperature level, corrosion/nitriding, manufacturing and assembly (which are not state of the art).

Tubular IHX R&D needs of "High Priority" include:

1. Tests to confirm fabrication feasibility (tube bending, tube welding, nozzles on hot header, ISIR and assembly, etc).
2. Corrosion and nitriding (Risk D-003) tests on base and coated materials in a representative environment.
3. Fabrication of representative IHX mock-ups from thermo-hydraulic and manufacturing point of views.
4. Testing in representative helium and helium-nitrogen environments is recommended.

The current plan is to use a full scale mock-up for component qualification. The need for intermediate testing on sub-scale mock-ups is deemed unnecessary provided that manufacturing issues are sufficiently addressed.

Plate IHX

The feasibility of the plate IHX is a concern and a reduced lifetime is expected (Risk D-011). Primary concerns are temperature level, corrosion, manufacturing, and thermal mechanical resistance. The current TRL of plate IHX is at 2. The plate IHX R&D needs, which are "Medium Priority"," include:

1. Development of visco-plastic model (material data-base to be completed).
2. Corrosion tests on base and coated materials in a representative environment.
3. Development of manufacturing techniques (fusion welding, diffusion bonding, brazing and forming).
4. Tests on representative IHX mock-ups from both thermo-hydraulic and manufacturing point of views (diffusion bonding, brazing, ISIR).

A three step approach is recommended for component qualification, these are:

1. tests in air with sub-scale mock-ups,
2. tests in helium with sub-scale mock-ups (about 1 MWt test loop). These tests will provide a basis for recommendations on which type of concept should be used for the NGNP, and
3. final qualification on a full scale mock-up (at least for the channels and the plates) on a large test facility (around 10 MWt).

Isolation Valves

A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR.

Qualification of the isolation valve has a priority of "High." The two qualification steps are:

1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.
2. Full scale mock-up tests in a relevant helium-nitrogen environment.

These tests should cover:

1. manufacturing parameters,
2. depressurization tests,
3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment,
4. leak tightness tests of the valve,
5. closing and opening and
6. fatigue and creep-fatigue of specific areas.

Fuel Handling System

Currently the Fuel Server portion of the Fuel Handling System requires the most development and is judged to have TRL of 6. Risk D-008 describes the Fuel Server risks and mitigation approaches. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate and Fuel Handling Machine, has been demonstrated at the Fort St. Vrain reactor. In addition, the HTTR reactor utilized a similar set of components. These portions of the system should be considered TRL 8.

Due to its "Low" priority, the Fuel Server system will be designed during the program. Testing of the Fuel Server system, beyond initial component testing, will be incorporated into the Fuel Handling System development testing program.

Reactor Cavity Cooling System (RCCS)

Use of an un-insulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated (Risk D-017). The basic components of the system are fairly common and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces. Consequently the RCCS has a TRL of 5.

Determination of the heat transfer characteristics of the proposed surfaces for the reactor vessel and the panel heat exchanger will need to be accomplished. A large scale demonstration of the capability of the RCCS to remove reactor decay heat is recommended. The cost for these tasks ("High Priority") is approximately $1.0M and the schedule is 24 months. Currently there is facility available at ANL which can accommodate a large scale demonstration of the RCCS.

Hot gas duct

The reference design (TRL of 6) for the primary and secondary hot gas duct is the Vee-shaped metallic concept. This design appears to be compatible with the core expected outlet temperature, subject to demonstrating that no significant hot streaks occur. The ceramic concept will be retained as a fall back option.

The hot gas duct qualification should be performed in three steps:

1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.
2. Sub-scale mock-up tests, about 1 MWt in helium if possible, to validate fiber specification and ceramic spacer specification.
3. Full scale mock-up tests, around 10 MWt.

These tests should at least cover

1. depressurization tests,
2. pressure loss, heat loss, temperature of the support tube (in helium),
3. leak tightness tests of connections
4. fatigue and creep-fatigue tests (e.g., bellows, Vee-shape spacers, etc).

The cost for these tasks is approximately $4.5M, not including cost of 10 MWt test facility which is currently not available, and the schedule is 24 months. In the first stages of the design, tests should cover both the metallic and ceramic concepts. Priority is "High."

Instrumentation

NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy.

For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime.

For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired. It will cost $2M to develop detector technology.

19.2.4 Computer Codes and Methods Development and Validation

A summary of computer codes that may potentially be used for VHTR analysis is provided in Table 19-5. The applicability and status of the subject codes are also shown in the table. In the following sections R&D needs for development and qualification of each code are presented.

Table 19-5: Summary of Potential Computer Codes for VHTR Applications

Codes	Categories	Objectives	Code Status
MANTA	Reactor System Analysis	Calculation of main system parameters (temperature, pressure, flow rate) of the HTR plant during all transient (normal, abnormal) when the primary coolant flows in forced convection, in order to define plant operation and control and to provide load data for primary components. Possibility to calculate generalized natural convection.	Fully Applicable Needs Validation
RELAP5-3D	Reactor Systems Analysis	Best-estimate system analysis coupled to CFD models for Generation IV, including gas reactor concepts. RELAP currently is a principal tool for LWR safety analyses.	Needs modification Needs validation
MCNP	Neutronics	Reference steady state core calculation for all type of cores.	Fully Applicable Needs Validation
NEPHTYS	Neutronics	1) Reactivity effects (first criticality, moderator and doppler coefficients, control rod worth, reactivity loss versus depletion) 2) 3D neutron flux and nuclear power distribution within the reactor core 3) 3D burnup distribution and nuclide inventory for back-end cycle and decay heat issues.	Needs Modification Needs Validation
MONTEBU RNS	Neutronics	Reference depletion and decay heat core calculations for all types of cores.	Fully Applicable Needs Validation
CABERNET	Coupled Neutronics/ Thermal	Reactivity, power and temperature, burnup and fluence distribution calculation in steady state and transient conditions for block type cores (input to fuel performance assessment).	Fully Applicable Needs Modification Needs Validation
STAR-CD	Thermal-Hydraulic	Determination of: 1) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, 2) the thermal behavior of the core, 3) the mixing inside the primary system, 4) heat losses and performances of components, 5) flow repartition across the components and 6) pressure shock waves.	Fully Applicable Needs Validation
FP Transport	FP Transport	Transport of radio contaminant species from the fuel block graphite walls into the primary coolant in normal operation and up to the environment in case of accidents.	Needs Major Modification

Codes	Categories	Objectives	Code Status
Structural Mechanics	Structural	Assessment of component behavior under normal operation and accident mechanical and thermal loadings.	Needs Modification
ATLAS	Fuel Performance	Assessment of coated particles performance during normal operation and accidental conditions. Calculation of the failure fraction and fission product release rate from a fuel load in normal operation or accidental conditions	Needs Modification Needs Validation

19.2.4.1 Reactor System Analysis Code

MANTA

Global validation of MANTA currently consists of code-to-code benchmarking: comparisons with CATHARE from CEA (France), LEDA from EDF (France), ASURA from MHI (Japan), REALY2 from GA (USA) and RELAP5-3D from INL (USA) have already shown good agreement. Qualification against experimental data is also progressing (EVO loop, HE-FUS3 loop and PBMM). Nevertheless additional benchmarks against experimental data are required. Some facilities that could provide valuable data have been identified: namely, HTTR reactor in Japan, HTR10 reactor in China, SBL-30 loop in the USA (SNL). The qualification of component models will follow from the qualification tests of the components. The core model qualification follows from comparison with other codes and with experimental results. Further, experimental data from HTTR and HTR-10 safety tests and from SBL-30 loop is required.

RELAP

The U.S. DOE sponsors RELAP5 code development at the INL. It is expected that this support will continue. Development needs are highlighted in the report INEEL/EXT-04-02993. Validation beyond that identified in this report and consistent with that planned for MANTA should be pursued.

19.2.4.2 Neutronics Codes

All neutronics codes identified for VHTR application have a TRL of 6. The R&D needs are mainly for code qualification against experimental data.

MCNP and NEPHTYS

The R&D needs for both MCNP and NEPHTYS are of “High Priority.”

1. The approach for qualification consists of comparing results against Monte-Carlo reference calculations and benchmarking against the few available experimental data (FSV, HTTR). Thus new dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and control rod and burnable poisons worths are needed.
2. Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly.

Data from FSV and HTTR first criticality testing can be applicable to MCNP and NEPHTYS code qualification.

The cost for neutronics code qualification has been estimated at $8M and schedule of 36 months, which includes the procurement of the fuel assemblies for the critical mock-up. The modification of the existing neutronics code for VHTR will cost $0.5M and 12 months.

MONTEBURNS

The R&D needs for MONTEBURNS are "High Priority."

1. Experimental results of fuel irradiation experiments (compacts or pebbles) at representative burnups, temperatures and fluences.
2. Experimental results of decay heat at short term (<100 hours) for representative fuel composition and burnup.

The cost to qualify MONTEBURNS is estimated at $2M with a schedule of 24 months.

CABERNET (=NEPHTYS / STAR-CD)

The TRL of the CABERNET code is 6 and the R&D needs are

1. Enhancement of capabilities for the calculation of transient analyses and
2. Experimental data of coupled power and temperature distributions obtained on representative fuel assembly geometry. If not achievable before NGNP: (a) Partial qualification data (e.g., burn-up measurements on fuel columns after irradiation in HTTR, which can provide a code/experiment comparison on the axial power distribution of a cycle); (b) Additional power margins will be necessary for initial operation of NGNP, to account for the uncertainty on the coupled neutronics-thermo-fluid dynamics calculation; (c) Need to provide in-core measurements of power and temperature distributions in NGNP for qualification of coupled calculations; (d) R&D needs for developing appropriate sensors for in-core measurements (never performed in HTRs). This code qualification can be performed during commissioning phase.

19.2.4.3 Thermal/ Hydraulics/ Pneumatics Codes

STAR-CD

The current TRL of the STAR-CD code is 6. Code development and qualification R&D needs are evaluated at a "High" Priority.

- Development of graphite oxidation model for air ingress transients on reactor internal structures.
- Qualification of:
  - conduction cooldown models on representative geometry, materials and temperature,
  - turbulence and mixing on representative mock-ups in critical areas (lower and upper reactor plena, hot gas duct, core bypass, IHX collectors) and
  - graphite oxidation models with selected graphite grades in representative operating conditions.

Several predecessor tests performed with different graphite grades at CEA and FZJ. NACOK experiments within the European RAPHAEL project (coupling of graphite models with thermo-fluid dynamic behavior) can be applied for STAR-CD qualification.

The qualification of STAR-CD code will cost $1.8M and 18 months and for development is $0.2M and 12 months.

19.2.4.4 Fuel Performance Models and Codes

ATLAS

The R&D need for ATLAS development/modification is to improve the diffusion and the coatings corrosion modeling. For the ATLAS code it has been estimated to cost $1.5M and 24 months for code development/modification.

For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, and fluence) are required to anchor the developed code. The estimated cost for qualification of ATLAS is $5M and schedule of 30 months, which includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models. Both development and qualification efforts for ATLAS have a "High" priority.

19.2.4.5 Other Codes

Fission Product (FP) Transport

The FP code has a TRL of 4. The R&D needs of the FP Transport code include development of models for:

- assessment of product activation in the primary circuit (in particular tritium and 14C),
- radio-contamination distribution in the primary circuit, making distinction between circulating activity, plated out / deposited activity and purification system, for both normal operation and accidental situations,
- radio-contamination releases outside the primary pressure boundary and
- radio-contamination releases in the environment during accident scenarios.

This FP transport code development along with the experimental work will cost $6M and 60 months. The experimental aspects are not included in this estimate.

It is also recommended to develop a mechanical analysis code for the NHS.

Structural Mechanics

Among all applicable computer codes the structural mechanics code has the lowest TRL of 3. The main tools for structural analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work (priority of "High") includes:

1) incorporation of constitutive laws for materials and developing numerical models
2) seismic modeling of a block-type core
3) fluid structure interaction and flow-induced-vibration methodology, and
4) leak-before-break methodology.

This effort is estimated to cost $1M and take 18 months to complete.

19.2.5 Power Conversion System

The major components of the PCS, including He/N_2 Cycle Control and Ducting, Heat Recovery Steam Generator, Steam Cycle and Generator and Electrical Equipments have a very high TRL (8 to 9). No R&D needs have been identified. However the turbo-machinery in the Brayton Cycle has been evaluated at TRL of 7.

Nitriding of metals will occur when exposed to hot nitrogen (Risk D-001). This nitriding process tends to embrittle metals which could lead to failures of turbine blades and pressure boundaries such as boiler tubes, gas shells, etc. The need to experimentally determine the degree of nitriding that occurs in potential PCS materials, and to quantify the effects of temperature on nitriding, has been identified. This R&D need is not only for turbo-machinery, but also for IHX (Tube) and Brayton cycle gas duct.

In addition, R&D is also needed for compressor blade performance in order to ensure high efficiency, mitigating the risk of lower than expected PCS efficiency (Risk O-001).

The total R&D efforts for PCS will cost $10M and take 18 months.

19.3 Possible Sources for R&D Tests

Using the R&D needs identified in Section 19.2, existing sources for performing some of the test programs were identified. These sources appear in Table 19-6. This table should not be regarded as an exhaustive list of test facilities, but as a starting point for detailed R&D planning.

Table 19-6: NGNP Test Facilities

R&D need	Description of the experimental means required	Sources of R&D	Comments
FUEL			
Fabrication			
Laboratory scale fabrication	To test the industrial fabrication process with limited flow of fuel material in order to be able to sreen a large number of fabrication parameters in order to improve the understanding of phenomena and optimize the process		
Coated particles		CEA Cadarache BWXT, ORNL	
Compacts		CERCA	
Industrial pilot line	Main components at the industrial scale in order to • finalize the definition of the industrial process • qualify a fuel representative of industrial production		
Coated particles		BWXT	6¨ diameter coater. Industrial coater might have larger diameter
Compacts			
Characterisation	Measurement of the key geometrical, physical, mechanical, thermal, chemical & micro-structural parameters of the fuel & of its materials & of the evolution of these parameters under irradiation		
Before irradiation		BWXT, Under development in CEA & AREVA	
After irradiation			
Fuel qualification			
Irradiation	• Online monitoring of fission gas releases in order to be able to count the number or failed particles • Test rig with sufficient capacity to have a statistically sufficient number of particles for proving the target performance of the fuel	SIROCCO (OSIRIS, CEA Saclay)	Cannot receive a sufficient number of compacts for acceptable statistics for qualification. Sufficient for lab. scale process validation
		HFR (Petten, Netherlands) ATR (INL), HFIR (ORNL)	
Safety testing	Out-of-pile heat-up of an irradiated fuel element with online fission gas monitoring & periodic solide fission product measurement	KÜFA (ITU, Karlsruhe), HFEF (INL), ORNL	

R&D need	Description of the experimental means required	Sources of R&D	Comments
MATERIALS			
Apart from standard laboratory means for material characterisation, the following specific means are necessary for studying the impact of HTR environment on materials:			
Corrosion studies			
In impure He atmosphere	Vessel material, high temperature metallic materials, graphite, composites		
Atmospheric pressure	High temperature (up to 1000°C), precise control of impurity content		Taking into account the number of materials to be examined (including the variability of composition of materials within the range of their specifications), the number of temperatures & atmosphere compositions to be screened & the length of each test, several loops must be operated in parallel, in order to be able to finalize the selection of the most appropriate materials for HTR operating conditions & to qualify the selected material within a reasonable period of time.
Uniform corrosion		CORALLINE, CORINTH (CEA Saclay) ESTEREL (EDF Les Renardières) AREVA Le Creusot	
Corrosion + creep		CORSAIRE, FLAMENCO (CEA)	
Corrosion + low cycle fatigue		CEA Pierrelatte	
Normal operating pressure			Verification of the absence of influence of the absolute pressure performed in the component qualification loops
In air ingress situations	Graphite & composite oxidation	OXYGRAPH (CEA Cadarache) THERA, INDEX (FZ Julich)	
Tribology	Friction and wear in high temperature representative He atmosphere	He tribometers in AREVA Le Creusot & CEA Cadarache	
Irradiation • Vessel material • Graphite • Composites (control rod cladding, possibly internals)	Irradiation test rigs • In a reactor providing a sufficient level of fast flux in order to obtain a representative neutron damage in a reasonable period of time • Maintaining high temperature conditions (between 400 & 1000°C) on material samples • Sufficiently large to accept a large number of samples in order to take into account the variability of the material, possibly also to irradiate together different material grades and to accept samples which are large enough to satisfy ASTM requirements for mechanical testing	An irradiation of vessel material have been performed and an irradiation of graphite is ongoing in HFR (Petten, Netherlands). Graphite irradiation is planned in OSIRIS (CEA, Saclay), including a test with in-situ irradiation creep measurement. Also, testing could be done in ATR (INL), HFIR (ORNL), and/or the MIT test reactor..	

R&D need	Description of the experimental means required	Sources of R&D	Comments
COMPONENTS AND SYSTEMS			
Plate IHX: selection of design concept, validation of the design and qualification by a step by step approach			
1st step: separate effect tests:impact of different design options on the performance			
Thermo-mechanical behavior	Representative operating temperature of the IHX and representative temperature transient. Air atmosphere is acceptable	CLAIRE loop being presently upgraded in CEA Grenoble (900°C, cool-down transients of 300°C in 5 sec., and heat-up transients of 300°C in 120 sec).	
Heat transfer performance	Representative fluids Representative geometry Representative temperature	• Tube geometry, He and He and He+N_2, 900°C (AREVA Le Creusot) • Representative geometry, air (CLAIRETTE loop, CEA Grenoble) • Representative geometry, He, 500°C (HE-FUS3 loop, ENEA Brasimone)	
Homogeneity of flow distribution	Large flow, room temperature & air acceptable	PAT loop available in EdF Chatou	
Corrosion	Representative atmosphere with controled impurities	See material corrosion loops	
2nd step: validation of the selected design	He loop providing at the same time most of the representative operating conditions		Detailed design finalized for a 1 MW He loop, 950°C, full pressure, controlled impurities, HELITE, to be built in CEA Cadarache. The flow is too small to have a representative flow distribution in the headers
3rd step: IHX qualification	Large He loop providing at the same time all the representative operating conditions allowing the qualification of an IHX module (10-20 MW)		
Tube IHX	Only final qualification necessary on the large He loop		
Valves, hot gas duct	Qualification on He loops		
Circulator	Air tests of the impeller (at scale 0.2 to 0.4)	Loops available in manufacturer facilities	
	He high temperature test of magnetic and catcher bearings	FLP 500 bench, Zittau University (Germany)	
	Integrated scale 1 test		Either dedicated loop at operating temperature (400°C) with very large He flow or during NGNP commissioning tests

R&D need	Description of the experimental means required	Sources of R&D	Comments
COMPUTER CODE QUALIFICATION			
Neutronics	Critical experiment with the possibility of getting an asymptotic neutron spectrum as well as transition spectrum at the interface of the core and the graphite reflector	Feasibility of a representative critical experiment in the MASURCA zero power reactor in CEA Cadarache assessed	
	Isotopic analysis of fuel irradiated to very high burn-up	Undertaken by ITU (JRC Karlsruhe) for a pebble irradiated to 15%FIMA in HFR, also INL	
Thermo-fluid dynamics	Representative mock-ups for critical areas, for instance		
	• Mixing of cold and hot streaks in core outlet plenum	MIR facility (INL)	Validation of turbulent mixing but not of the thermal diffusion between hot and cold streaks
	• Conduction cool-down: validation of modelling		
	- Decay heat release through internal structures to the vessel outer wall		
	- Radiative heat transfer in the reactor cavity and natural circulation in the RCCS	NSTF (ANL)	
	• Flow in IHX headers (see IHX)		
Neutronics / Thermo-fluid dynamics coupling	In core flux and temperature instrumentation in NGNP		To be developed
System analysis	Comparison with data on gas loop transient operation	Many data exist on transient operation of gas systems in which same type of phenomena as in NGNP occur (EVO loop (Germany), HE-FUS3 (ENEA), Micro-model (South Africa), HTR-10, HTTR.	

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Technical Data Record

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Design Attributes and Considerations for a Gas-Cooled Graphite Moderated Irradiation Test Reactor

Design Attributes and Considerations for a Gas-Cooled Graphite Moderated Irradiation Test Reactor

Safety Related? ☐ YES ☒ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

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1.0 INTRODUCTION

1.1 Background

The US Department of Energy (DOE) is conducting an evaluation of candidate advanced reactor concepts that could serve as either a test reactor to provide irradiation services or as a demonstration reactor which might demonstrate an advanced power reactor concept for subsequent commercial deployment. AREVA is providing support to Idaho National Laboratory (INL) to assist in developing point design reactor concepts for a gas-cooled, graphite-moderated test reactor and for a larger gas-cooled, graphite moderated demonstration reactor.

This report is focused on the irradiation test reactor concept and related design considerations.

AREVA has reviewed draft goals, criteria, and metrics from the DOE study. These are being developed within the DOE study for eventual use in the evaluation of candidate advanced test reactor concepts. Based on these criteria and AREVA's familiarity with gas-cooled, graphite-moderated reactor technology, AREVA has considered the benefits, issues, and challenges of such a test reactor concept. The resulting insights are the primary focus of this report.

In addition, INL has already performed a variety of neutronic and thermal-hydraulic scoping analyses for a possible gas-cooled, graphite-moderated test reactor configuration. These initial scoping calculations provide insights into the feasibility and performance capabilities of such a concept. AREVA has reviewed preliminary results of these calculations. Those results have helped to inform observations offered by AREVA later in this report.

In addition, comments and suggestions pertinent to the preliminary INL analyses are offered based on AREVA's reactor design experience. Those comments are intended to provide insights for INL analysts who are working on additional analyses of the gas-cooled, graphite moderated test reactor concept.

1.2 Report Structure

In meeting its purpose to support INL work, this report contains several key sections which are designed to provide a vendor's perspective on the use of a high temperature gas-cooled reactor (HTGR) as a test reactor. These sections are:

- A review of the required characteristics that have been developed for the Advanced Gas-cooled Test Reactor (AGTR) with a short vendor perspective on those requirements. The AGTR name is used in this report to describe a generic material test reactor based on gas-cooled, graphite moderated reactor technology.,
- Two sections which examine the advantages and challenges associated with the use of HTGR technology for the AGTR,
- A review of suggested attributes that an HTGR-based AGTR should exhibit to maximize the benefit of this technology,
- A high-level description of a notional design for such a test reactor,
- A list and brief discussion of suggested additional analyses that INL should consider in its future work,
- A list of specific questions and comments generated by AREVA during its review of INL documents provided as part of this effort, and
- Overall conclusions regarding the sections listed above.

2.0 REQUIRED CHARACTERISTICS

2.1 Mission

The mission of the AGTR is to provide an irradiation test bed to support development and qualification of fuels, materials and other important components/items (e.g. control rods, instrumentation) of both thermal and fast spectrum Generation IV advanced reactor systems. To support this mission, the reactor chosen to be the AGTR must be based on a mature reactor concept, capable of providing reliable irradiation services supporting development of other advanced nuclear technologies without requiring significant testing and development itself. Put another way, the purpose of the AGTR is to provide a platform to perform testing supporting other reactor designs, not testing for its own design.

2.2 Fundamental Requirements

In defining the desired characteristics of an advanced technology test reactor concept for the US DOE Advanced Test/Demonstration Reactor Planning Study, some universal characteristics were defined which would be expected to be attributes of any candidate reactor. Therefore, these characteristics are not included in the criteria intended to be used to quantitatively evaluate each candidate reactor technology or to differentiate between those reactor concepts. These fundamental characteristics are identified as Desirable Outcomes and Requirements in the draft evaluation criteria (see Appendix A).

However, while all candidate reactor concepts are required to satisfy these fundamental requirements, such compliance is not an inherent characteristic of all reactor technologies or of all specific reactor concepts. The ability of a specific reactor concept to satisfy these requirements is a direct function of the details of the reactor design concept and the accommodation of the strengths and weaknesses of the underlying reactor technology. Therefore, while it is anticipated that a reactor concept based on any of the candidate technology options could be designed to satisfy these fundamental requirements, it will still be necessary to evaluate the individual concepts to ensure that they each meet these requirements.

The following paragraphs provide an initial assessment of an HTGR-based test reactor concept's ability to satisfy these requirements.

<u>Desirable Outcome 1: Provides a focal point for nuclear energy R&D activities</u>. The development, construction, and operation of the AGTR would provide many diverse opportunities to engage and support a variety of stakeholders in the United States nuclear R&D arena.

The AGTR project would engage all sectors in the development of the reactor including R&D support from national laboratories and university programs, reactor vendors and supporting elements of industry, the anticipated reactor owner/operator, and regulators. While the reactor concept would make the maximum use of existing technology in order to minimize risk, current HTGR technology development must be completed. In addition, subsequent materials development work would continue to benefit the reactor during its lifetime, providing a variety of performance enhancements for increased irradiation capabilities. Licensing of the reactor will allow the US NRC to address many of the issues which are generic to HTGR licensing. This will help to advance the corresponding expertise within both the NRC and in industry, and it will help to advance the general licensing framework for gas-cooled reactors, benefiting future commercial HTGR projects.

Operation of the AGTR would also provide significant opportunities for training the next generation of engineers and scientists, providing familiarization with key HTGR technologies which will be a key component in the future of extending the benefits of nuclear power to process heat industries.

<u>Requirement 1: Robust Safety Design Basis</u>. The AGTR concept shares the same fundamental safety characteristics as larger commercial modular HTGR concepts. The inherent safety characteristics of the modular HTGR far exceed the basic requirements which must be satisfied by all reactor concepts. This concept offers true walk-away safety.

The concept does not require electrical power, operator action, active shutdown systems or even reactor coolant to meet safety requirements. Like other modular HTGR concepts, the system has large thermal inertia, so that temperatures do not change rapidly, even with a complete loss of cooling. The use of TRISO particle fuel provides excellent radionuclide retention for all operating and accident scenarios. The use of inert coolant and extremely stable graphite structural components prevents adverse interactions between reactor materials.

In any test reactor, the design and safety evaluation of individual irradiation experiments to be inserted into the reactor is an important consideration in the ongoing safety of the facility. However, the large operating and accident margins provided by the AGTR concept provide increased confidence that the facility will be able to accommodate a wide variety of experiments without adversely impacting the safety of the reactor or that of the individual experiments to be inserted into the reactor.

<u>Requirement 2: Safeguards and security</u>. The AGTR facility will fully satisfy all relevant IAEA and US NRC safeguards and security requirements.

The inherent safeguards characteristic of the prismatic block test reactor are robust. The reactor is based on a low-enriched uranium (LEU) fuel cycle. Access to the fuel materials is limited since the fuel cannot be removed from the reactor in mid-cycle without shutting down the reactor and opening the pressure boundary. The individual fuel elements are not easily diverted or transported, and all fuel elements are uniquely identified for accountability. The same fuel handling and accountability system would be used for the advanced gas-cooled test reactor as would be used for the commercial prismatic block core HTGR.

Security of the AGTR facility is provided both by the inherent invulnerability of the system to malicious acts and by the optimization of the facility to prevent unintentional access. The fundamental safety characteristics of the system make it resistant to inappropriate operator actions of omission or commission. This also minimizes the vulnerability of the system to deliberate malicious acts (such as deactivating cooling systems or removing the primary coolant). In addition, the minimal reliance of the system on safety cooling or protection systems minimizes the vulnerability to potential sabotage involving those systems.

2.3 Criteria

Appendix A provides a listing of anticipated criteria to be considered when assessing different proposed test reactor design concepts. These criteria are taken from a working draft of the Advanced Test and Demonstration Reactor study criteria provided for AREVA use by INL. These criteria were used to shape the discussion presented in the following sections; that is, the information presented is intended to provide background and interpretation of AREVA's HTGR test reactor concept in the context of helping to demonstrate compliance with the stated criteria.

In order to facilitate the discussion, each of the metrics, by which the criteria are assessed, is presented below with a statement clarifying AREVA's interpretation of the intent of the metric.

<u>Metric 1.1.1 – Flux Conditions</u> – This metric assesses the maximum achievable fast and thermal flux in the test reactor, with the caveat that they do not have to occur in the same place. It is assumed that the flux levels reported are averaged over the duration of a particular irradiation cycle, or at least over the duration claimed in response to Metric 1.1.3. In addition, though not discussed in the criteria, the ability to control the ratio of fast-to-thermal flux at various locations could be valuable to potential experimenters.

<u>Metric 1.1.2 – Irradiation Volumes and Length</u> – This metric appears to be a fairly straightforward assessment of geometric conditions. It is assumed that the reported volume is the total volume of all test locations, while the length is the free length of the longest available test location. Though not considered under this metric, the diameters of the various test locations are also considered important to facilitate various types of test articles.

<u>Metric 1.1.3 – Maximum Sustainable Time at Power</u> – As noted for Metric 1.1.1, this sustainable time at power is assumed to be that time for which the claimed neutron flux values are available during a single operating cycle.

Metric 1.1.4 – Provisions for Testing Prototypic and Bounding Conditions – There are two potential interpretations for this metric: 1) the reactor has the capability to accommodate test loops in which prototypic and bounding test conditions for a particular reactor technology under development can be simulated, and 2) the reactor operates at conditions prototypic of a full-scale version of its reactor type. Both interpretations are considered in this document.

Metric 1.2.1 – Number of Test Zones – It is assumed that the number of test zones refers to the number of individual experiments that can be inserted into the core, using typical experiment sizes for existing test reactors as a guide. That is, a reactor with a core significantly taller than existing test reactors, or with test holes significantly wider, may take credit for more than one “location” per physical test hole.

Metric 1.2.2 – Number and Type of Distinct Test Irradiation Test Loops – This metric is taken to refer to the number of individual spaces available in the core for irradiation test loops, regardless of how those loops are used in any given irradiation cycle.

Metric 1.2.3 – Ability to Insert/Retrieve Irradiation Specimen at Power – It is assumed that this ability is not required of all test locations in the core, but rather for some locations anticipated to handle fairly small samples. In addition, it is not interpreted to refer to the removal of loop-type experiments while at power.

Metric 2.1.1 Project Cost – This metric is taken to refer to the projected overnight cost of the test reactor, including supporting R&D, design, licensing, and construction.

Metric 2.1.3 Schedule – This metric is assumed to measure the expected duration of construction and initial startup of the test reactor. Design, licensing, and site preparation work are not assumed to be included in this estimate.

Metric 2.2.1 Annual Operating Cost – This metric is assumed to include only operating costs, that is, it does not take into account projected additional capital costs. It is also assumed to not include the costs of the individual experiments and associated specialized supporting equipment. It is noted that the caveat, “including contingency that reflects technical maturity of the concept”, is not clear, including how such a contingency is determined and how it is applied to the expected operating cost.

Metric 2.3.1 Availability Factor – This metric is interpreted to be the number of days in a year minus the number of days required to refuel or maintain the reactor over that year. Included in this time would be a representative estimate of the time needed to insert experiments into the core during each shutdown.

Metric 3.1.1 Number of Secondary Missions – This metric is assumed to include secondary activities which provide an additional revenue stream to the facility, beyond irradiation of test samples. It is noted that there are limitations on the application of test reactors for revenue generating secondary missions under current regulations. These limits would have to be addressed in any future test reactor project.

3.0 ADVANTAGES OF HTGR TEST REACTOR

The following sub-sections examine the characteristics of an HTGR-based material test reactor that are deemed to be advantageous in light of the requirements and metrics described in Section 2.0. These discussions focus on the main characteristics of an HTGR concept only. A more detailed discussion of AREVA’s view on a possible advanced gas-cooled test reactor concept, the Material Test HTGR (MT-HTGR), is presented in Section 6.0.

3.1 Size/space for experiments

The core of an HTGR is physically large compared to many of the other potential material test reactor concepts. The diameter of the annular core plus one row of outer reflectors, that is, the region of interest with regard to space available for the placement of test samples, is approximately 2.5 meters. The height of the active core is approximately 6 meters.

This large physical size affords the opportunity to incorporate many and varied test locations, which together result in a very large available total volume for experiments. This large test volume, and resulting large number of potential test locations, should alleviate issues related to long wait times for an available test spot.

The height of the active core will support placement of experiments requiring a long length under irradiation or facilitate a significant number of individual test capsules in a single test string.

The availability of locations able to support relatively large diameter experiments will support the capability to irradiate fairly large diameter samples or samples requiring more extensive supporting hardware and systems, such as fuel assembly structures, flow loop experiments simulating various reactor primary coolants and conditions or experiments requiring extensive in-core instrumentation.

3.2 Accommodate a wide range of experiments

In addition to the large physical size, the nuclear characteristics of the reactor provide significant flexibility to support a wide range of experiments. The three dimensional nature of the core, that is, stacked blocks rather than full length fuel elements, provide an additional degree of freedom for core designers to accommodate a wide range of types of experiments. The basic neutronic characteristics of the core provide a wide range of available neutron energy spectra, which can be further tailored through the use of alternate fuel materials and additives to the fuel element in select compact locations and through selective use of materials in the test assembly.

An HTGR is able to accept a wide variety of materials for irradiation, including fissile materials, absorber materials, moderator materials, and structural materials. The space available for deployment of sample cooling strategies, using both reactor coolant and/or external cooling loops, allow control of sample irradiation temperatures during the course of the irradiation.

The ability to accept significant sample support infrastructure, including externally cooled loops and various neutron spectrum tailoring solutions, allow an experiment to be conducted under conditions that are prototypic of, and support, all current advanced reactor concepts.

3.3 Significant secondary mission potential

An HTGR, through its use of high temperature helium as a primary coolant in a technically mature heat transport and power conversion system, has significant potential to support secondary missions without impact on its primary mission as a test reactor. Using the hot reactor outlet helium to supply energy to a gas-to-water steam generator facilitates high efficiency electricity production via a conventional Rankine turbine-generator set. Steam conditions are comparable to modern fossil-fired steam systems. This leads to high efficiency (anticipated to be 40% for the test reactor), and it gives compatibility with conventional “off-the-shelf” steam turbine and generator equipment.

Anticipated revenue could be a significant fraction of the operating cost of the plant. The high temperature steam can also be used to support various process heat needs, including process heat research and new technology demonstration.

The very large irradiation test volumes available in an HTGR make possible commercial-scale production of key medical and industrial isotopes without significant impact on the space available for testing. This capability has the potential to provide a very lucrative revenue stream to support ongoing test reactor operations.

The HTGR’s ability to readily support the secondary missions of electricity and process heat generation as well as radioisotope production will provide a consistent revenue stream for the plant that will offset reactor operating costs and allow for significantly lower fees to be charged for irradiation services.

3.4 Potential for extended cycle lengths (much longer than 90 days)

It is anticipated that the normal cycle length for the HTGR will be significantly longer than the benchmark 90 days, perhaps up to six months. The actual cycle length would depend on the core management strategies employed to maximize benefit for the contained experiments. This would allow samples to experience uninterrupted irradiation cycles of up to that long without the intermittent power changes and cooldowns associated with shorter cycles.

As an additional consideration, the HTGR is designed to use fuel with uranium enrichments below 20 w/o, compared to most existing test reactors which utilize fuel with high enrichments to achieve even one to two month cycles. The configuration of a gas-cooled reactor provides significant core design options that can be utilized to meet operational goals. The main parameters are the packing fraction (which affects the neutron energy moderation), the number of surrounding moderator blocks (mainly because HTR fuel blocks are generally designed as under-moderated), the enrichment (the average affects core lifetime and local enrichment roughly affects assembly power), and the reflector thickness (which affects leakage and therefore lifetime). The exact core design envelope will need to be determined as part of the overall test reactor design and optimization process once design requirements of all potential stakeholders are considered.

3.5 Increases diversity in research reactor fleet

Once operational, an HTGR-based system would be the only high temperature gas-cooled reactor specifically available for material sample irradiation. This would not only provide more diversity to the available suite of irradiation services in terms of temperatures and neutron spectra, it would be the only source of prototypic irradiation conditions available for development of advanced gas-cooled reactors.

3.6 Inherent safety characteristics and large margins

The HTGR test reactor shares the robust inherent safety characteristics of all modular high temperature gas-cooled reactors, which provide a robust platform upon which the test reactor has been developed.

The basic materials from which the reactor is constructed are well understood. The primary coolant helium is easy to handle, is chemically inert, and does not experience phase change during any operational condition, including accident conditions. Core structures are graphite, a well understood high temperature structural material. The vessel system is based on use of conventional light water reactor (LWR) pressure vessel steel.

The large thermal mass of the reactor core, reflectors and structural materials make the reactor very stable thermally. Postulated events, including accidents, occur over a very long time scale, allowing time for planning and executing appropriate mitigating measures. In addition, the refractory nature of the core materials provides large margins to thermal limits. In all postulated normal operation and accident conditions, the fuel temperature is expected to remain well below allowable limits.

Neutronically, the HTGR core design is well-behaved under all operating and accident conditions. The core has negative temperature feedback over its whole operating range, and it will naturally shut down in the event of a temperature excursion, even without a reactor trip. Therefore, the core design is expected to allow significant flexibility and to be very tolerant of the placement of a variety of test specimens. That is, it is fully expected to be feasible to construct a suitable core design and a corresponding bounding safety analysis that would facilitate a wide envelope of potential sample types and configurations that could be irradiated without need for extensive re-analysis and update. These characteristics will also ensure success in special test cases where more specific analysis is required for conditions outside the standard test envelope.

3.7 Cost, Schedule and Reliability

Of all choices for a Generation IV-based material test reactor, the HTGR provides the surest ability to predict design, construction, and operation costs and schedules based on its high level of technical maturity and basis in operating HTGRs. All of the key technologies upon which the HTGR relies have been demonstrated in past HTGRs or industrial applications. In a similar fashion, this design and experience base can be used to provide the most reasonable estimates of operational reliability amongst potential Generation IV concepts.

4.0 CHALLENGES

While the HTGR has several characteristics that would be beneficial for an advanced gas-cooled test reactor, there are two primary areas that will be challenging for the designers of this reactor concept. The first challenge is achieving the desired irradiation test conditions using a graphite-moderated reactor core. The second relates to the basic reactor design challenges of providing accessible test locations in close proximity to the HTGR core.

4.1 Test environment challenges

For an ideal irradiation test reactor, it is desirable to have as high a neutron flux as possible. This implies a high power density core, and therefore preferably a small core, in order to keep the overall power level as small as possible. However, the modular HTGR typically has a relatively large core. This suggests that modifications will be needed to maximize the flux.

There are three main challenges to be addressed regarding the neutron flux distribution:

- Desired flux magnitude
- Flux spectrum (fast/thermal)
- Reactivity swings and spatial flux distribution (due to the different neutronic impact of various irradiation experiments)

If one considers the combination of total test irradiation volume and the flux, one gets an overall indication of the total test irradiation throughput of the reactor concept. So even though the flux may not be as large as for some reactor alternatives, the large available test volume may compensate for this in terms of overall irradiation capability. In that case, even though required irradiation times for some experiments might be longer, the wait time to get into the reactor might be significantly less.

Nonetheless, a high flux is still desirable to support individual tests requiring rapid fluence accumulation. So, a key challenge for the advanced gas-cooled rest reactor designer is to optimize the design to maximize the flux and achieve the necessary fast flux and thermal flux.

4.1.1 Achieving desired flux magnitude

The designer will obviously move to a smaller higher power density core, such as those which have been evaluated in preliminary work performed by INL [1]. The challenge is to accomplish this within the design constraints of the TRISO particle fuel performance envelope on the one hand and while maintaining adequate overall thermal margins in the core for both normal operation and accident conditions on the other.

4.1.2 Flux spectrum (fast/thermal)

Beyond the overall flux level, the ratio between the fast flux and the thermal flux is an additional challenge. The HTGR core is normally fabricated entirely from structural graphite. While it is typically slightly undermoderated, it is still a relatively thermal spectrum. The spectrum is relatively hard compared to a fully-moderated water reactor, but it is much softer than a fast reactor. Therefore, the ratio of fast to thermal neutrons is lower than that of a fast reactor spectrum. This will adversely affect tests which specifically require a fast neutron spectrum, and it may increase the required irradiation time for some materials tests which are primarily sensitive to fast fluence.

Therefore designers may have to consider special features which can locally enhance the fast flux or which can shift the ratio between fast and thermal neutrons. And the impact on total flux of such factors must also be taken into account.

4.1.3 Reactivity swings (due to variations in experiments, etc.)

Beyond the neutron spectrum provided by the test reactor core, the designers must also take into account the effects of various possible experiments which might be placed in the reactor. This includes both the effects on overall core reactivity as well as the impact on the local flux and flux ratio. Any special measures included to tailor the spectrum (e.g., preferential moderation of fast or absorption of thermal neutrons) must be taken into account as part of this evaluation. The overall excess reactivity of the core as well as the compensating total available control rod worth must provide the necessary flexibility to address this wide range of potential conditions.

4.2 Reactor Design Challenges

The nature of the HTGR is quite different than most current test reactors. It is a high temperature, high pressure system compared to a water-cooled pool type reactor that operates near atmospheric pressure and at relatively low temperature. While the HTGR does offer significant advantages, it also imposes some engineering challenges in terms of inserting and accessing irradiation test locations inside the reactor core. These reactor design engineering challenges include:

- Crowding caused by an increased number of pressure boundary penetrations
- Maintaining the pressure boundary for test wells that extend into the reactor
- High core outlet temperature
- Accident temperatures (for metallic structures adjacent to core)
- Support of test wells in graphite core structures

Compared to other reactor types, the gas-cooled test reactor offers a very large irradiation test volume at locations in and adjacent to the core. However, access to these locations must pass through the reactor pressure boundary. Some samples may be placed inside the reactor during shutdown or refueling when the pressure boundary has been opened. However, samples that require an alternate controlled environment during irradiation and samples that require online monitoring instrumentation or gas sampling during irradiation will require dedicated access through the pressure boundary during reactor operation.

4.2.1 Crowding for access on top head

The most obvious arrangement for inserting tests into the reactor is from the top, since this provides good alignment of test well structures with the basic geometry of the reactor core. There is also adequate open space above the reactor to allow the removal of test specimens and related support structures from test wells in the reactor. Such access would require penetrations for test well access in the top head of the reactor vessel.

However, the top head of the reactor vessel already includes large penetrations that house the control rod drive motors. These penetrations also provide access for the refueling machine to remove and insert fuel blocks and reflector blocks during refueling operations. The distance between penetrations in the top head is governed by fundamental structural design considerations. Adding additional penetrations for test well access will increase crowding on the top head of the vessel. Figure 4-1 illustrates this challenge.

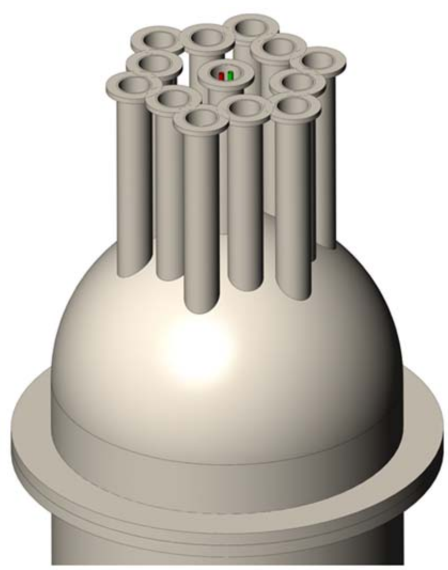

Figure 4-1: Top Head Crowding

Initial evaluation indicates that there is room to add some penetrations for test well access, but the total number of penetrations that can be added is clearly limited. In addition, the allowable locations for the top head penetrations may not all be ideal for test well access. Clearly careful consideration will have to be given to the penetration arrangement on the reactor vessel top head as the arrangement of control rods and test wells is finalized.

4.2.2 Accommodating high pressure boundary

The issues associated with the primary coolant pressure boundary extend into the heart of the AGTR. For test wells and test loops which may be inserted into the reactor core, these test wells may represent an extension of the pressure boundary down to the irradiation test location in the core. If the test well can be sealed during reactor operation, then the pressure boundary may be closed at the vessel penetration where the test well enters the reactor. But if the test well must remain open to the outside for irradiation sample access during operation or for supply of an external cooling fluid to maintain an alternate environment for the irradiation sample, then the primary coolant pressure boundary must be at the test well boundary within the reactor core.

In situations where the test well forms part of the pressure boundary, it makes design of the test well more complicated. The test well will have to satisfy additional structural requirements. The differential pressure across the wall of the test well could span a wide range depending on the anticipated pressure within the test well.

4.2.3 High core outlet temperature

The reactor operating temperatures are significantly higher than those in a low pressure water-cooled reactor. The reactor inlet temperature is relatively modest, being similar to commercial LWR operating temperatures. The reactor inlet temperature is fully compatible with conventional reactor structural materials which might be used for irradiation test wells.

However, the reactor outlet temperature is significantly higher than typical LWR temperatures. For metallic components which would be exposed to these temperatures, the material options are more limited. Some alloys are available for use at these temperatures, but compatibility with other design constraints would have to be evaluated. An equally important consideration would be the high temperature capability of irradiation specimens to be placed in test wells exposed to the core outlet temperature. These specimens would also have to be designed for the higher temperatures.

Of course, a key advantage of the AGTR is that it can readily provide such a high temperature irradiation environment for tests that require it. Therefore, the reactor designers will want to include such test locations in the detailed concept. But in the process, they will have to address the design challenges entailed in such a test well location.

4.2.4 Accident temperatures

A key feature of the AGTR core is its ability to tolerate a complete loss of cooling and to accommodate the resulting core temperatures without fuel damage or structural damage. If all forced cooling is lost or the reactor coolant is lost due to a coolant leak, the core temperatures will stay within design limits. The resulting temperatures are well within the capability of the TRISO fuel particles and the graphite structures. However, those temperatures can be challenging for metallic components.

It is anticipated that during normal operation most test wells will be cooled to reactor inlet temperature (or to the temperature set by externally supplied cooling within the test well to maintain an alternate environment for the sample). But all test wells within the core or immediately adjacent to the core will be exposed to the local core temperatures during a loss of forced cooling event.

Since the AGTR core is much smaller than the large core in a commercial scale HTGR, it is anticipated that the peak core temperatures during a loss of cooling event will be significantly lower than in the larger modular HTGR. Nonetheless, at a minimum, the local temperatures will approach the normal fuel element operating temperature. Therefore, designers will have to design the metallic structures adjacent to the core to be able to tolerate these temperatures under accident conditions.

4.2.5 Support of test well(s) in graphite core structures

The physical support of the test wells also poses a new design challenge. To reach test locations adjacent to the active core, test wells will have to extend a significant distance from the reactor vessel penetration location, through surrounding structures, through the graphite reflector structures, and to the active core locations. This will likely result in a very long, slender structure for each test well.

The expected entry path for the test wells is for the metallic structure to pass through circular openings in the graphite reflector blocks and fuel elements. Once the test well passes into the graphite core structure, the opportunities to provide rigid support for the long slender structure are limited. Significant relative motion of adjacent graphite blocks is possible over time. The metallic structures and adjacent supports will have to accommodate these relative motions.

Designers will have to develop a design strategy that accommodates the relative motions of core graphite components, provides the required lateral support for the test wells, and is compatible with both normal operating and accident temperatures. It is certainly possible to define such a strategy. Nonetheless, this will be a challenging aspect for reactor designers. It is noteworthy that in general, the test wells can have a design lifetime that is very limited relative to the lifetime of the reactor. (Graphite reflector elements also have limited lifetimes due to irradiation damage.) This shorter lifetime provides much better design flexibility in dealing with fluence, creep, and thermal damage considerations.

5.0 ATTRIBUTES RECOMMENDED FOR DESIGN CHARACTERISTICS

Considering the fundamental characteristics of an HTGR and the strengths and challenges anticipated in adapting an HTGR core to an irradiation test reactor mission, several attributes are recommended in order to best fulfill the test reactor mission.

5.1 Access Locations

As has already been discussed, crowding on the top head of the reactor vessel is expected to be a challenge. Therefore, designers of the gas-cooled test reactor should consider all potential alternatives for test well penetrations. For example, test well entry from the bottom and sidewall of the reactor vessel should also be considered.

Several different types of access are anticipated for the gas-cooled test reactor:

- Direct placement of capsules or test articles within the core structure.
- Test wells accepting large test articles (e.g., component mockups, partial fuel assemblies, etc.)
- Test wells accepting externally controlled test loops (with external coolant and instrumentation connections)
- Test wells containing multiple small test capsules (immersed in air)
- Test wells containing multiple capsules in helium coolant
- Test wells containing multiple capsules in helium (actively cooled)
- Test wells for rabbit or other small samples to be rapidly inserted/removed

5.1.1 Access for test articles placed directly within the core

Some tests involving large volume but not requiring a special (non-HTGR) test environment would be placed directly in the graphite core structure. Such potential tests might include irradiation of HTGR materials and structural components, particularly composite structures. The performance of these structures is very dependent on the final geometry form, and qualification benefits significantly from using larger irradiation specimens. Such potential tests might also include bulk fuel samples intended to provide large volume qualification data. In this case entire fuel blocks containing the special particle design could be loaded in the reactor. Component and material testing for other high temperature reactor concepts could also be performed in cases where the irradiation behavior of interest was not dependent on the local coolant environment. Performing this testing in a helium environment would simplify test article loading, removal, and post irradiation examination compared to performing similar irradiations in alternate coolants (e.g., sodium or molten salt). (For irradiations requiring an alternate coolant environment, the gas-cooled reactor can accommodate multiple external test loops.)

Access for these test articles is relatively simple. They are inserted and removed when the reactor is shut down for refueling. This is appropriate, since the irradiation cycle for these components will typically be the same or longer than the normal refueling interval. Test articles requiring cooling (typically alternate fuel designs) would replace one or more fuel blocks in the active core. Structural articles would be placed in reflector locations in either the central reflector or the outer side reflector. (Structural test articles requiring exposure to core outlet temperature would likely be placed in the bottom reflector, where they would be exposed to core outlet flow.)

5.1.2 Access for large test articles or externally controlled test loops

For test wells accepting very large articles or external test loops, access from the top of the reactor is probably required. The anticipated approach is that a test well will be provided from the vessel penetration extending into the graphite core structure. This test well would be part of the primary coolant boundary. Then a separate test loop assembly (or other large test article) would be inserted into this test well. The test loop structure would be self-contained, relying on the surrounding test well only for maintaining alignment with the reactor geometry. This allows various test loops to be inserted and removed without directly impacting the reactor coolant boundary. This arrangement is illustrated in Figure 6-2. (For long-term use, dedicated test loops could be installed directly, without the surrounding test well, so that the test loop outer shell provides both the test loop structural support and the reactor coolant boundary. This increases neutron flux at the expense of slightly reduced operational flexibility.)

These test wells will generally require essentially a straight line access from the vessel penetration to the test locations adjacent to the active core. Additionally, they will require pull space beyond the vessel penetration for removal of the test article, test loop, and associated structures. Clearly a vertical alignment of these long, straight test wells is most compatible with the graphite block core structure, and the space above the reactor provides readily available pull space.

Straight access from the bottom of the reactor could also be possible for these long test wells. However, direct access from the bottom of the vessel would have to pass through the hot core outlet plenum region. This is certainly possible, but it imposes additional engineering challenges. More importantly, bottom access would require substantial pull space below the reactor, significantly increasing the building cost.

Side access could also be considered for these larger straight test wells. For side access, the path to the core is shorter, so the total length of the test assembly and associated instrumentation and/or cooling connections would be shorter. This means that less pull space would be required. Moreover, pull space at the side of the reactor is not as expensive as space beneath the reactor. The biggest challenge for side access would be the additional complication that this brings to the design of the side reflector structures. These challenges are not insurmountable, but they will add to the design cost, they will complicate the reflector replacement process, and they will complicate the bypass cross flow analysis.

If it is anticipated that some externally controlled test loops are narrow enough or sufficiently flexible to allow some bending, then other options become possible. This allows the access penetrations for some test wells to be offset from their locations directly above the test well. This provides additional flexibility in laying out all the penetrations on the reactor vessel top head in such a way that meets all structural and code requirements.

5.1.3 Access for small capsules in air

There is much more flexibility for smaller capsules which can be immersed in air. These would typically be irradiation samples which do not have significant heat generation and/or do not require precise temperature control. Test wells for these specimens could curve much more easily than those required for large articles or major external test loops. Therefore, there is much more flexibility in locating the penetrations through the reactor vessel for these test wells. They could be on the periphery of the top head, the sidewall or the bottom of the vessel. Since they can follow a curved path, they could more easily traverse the reactor internals structures located at the bottom of the reactor.

The amount of pull space required for these test wells should be modest. The space required for the test capsules themselves would be minimal. The real pull space requirement depends on the support structure used for the capsules. Nonetheless, there is significant flexibility in designing these support, insertion, and retrieval structures. It is anticipated that the required pull space could be accommodated on any reactor boundary with minimal impact on the surrounding building structure.

With the increased flexibility of the test well design for smaller capsules, it is also possible to extend the test well all the way through the reactor, providing vessel penetrations at both ends. This would allow additional flexibility in the insertion and removal of irradiation specimens, with the potential to significantly increase throughput.

For these test wells, the test well forms a part of the primary coolant boundary. Nonetheless, it is possible to also seal the test well at the vessel penetration, if that is necessary to provide redundancy of the coolant boundary. More importantly, the test well connection at the vessel penetration could be connected to a controlled air supply in order to supply cooling air, if necessary. This would allow excess heat generated in the test capsule(s) to be removed, and it would allow the test capsules to operate at temperatures below the surrounding primary coolant temperature, even though precise temperature control was not necessary. (If precise temperature control was required, then an externally controlled test loop would be used.)

5.1.4 Access for capsules in helium coolant

Access for capsules to be immersed in helium instead of air would be similar to the test wells for static capsules immersed in air. There would be significant flexibility in the location of the vessel wall penetrations for these test wells.

The main difference for the helium test well would be that the primary coolant boundary would be formally placed at the vessel penetration. This would necessitate a leak-tight pressure retaining closure at that location. The test well within the reactor would be vented to the primary coolant environment within the reactor. If desired, limited cooling could be provided in principle by venting the test well both near the core inlet and near the core outlet. This would provide a modest cooling flow similar to the bypass flow which cools the control rods.

The main advantage of this is that the test well wall thickness is greatly reduced, since the pressure differential across the wall would be negligible. This reduces any attenuation of neutrons in the test well wall material. The overall flux in the vicinity of the test well is increased slightly, and the flux reaching the specimen within the test well is improved significantly.

5.1.5 Access for capsules in helium (actively cooled)

Test wells which immerse capsules in helium could optionally be actively cooled. Instead of venting the test well directly to the primary coolant within the reactor, purge helium would be provided at one end of the test well. The cooling flow would then exit to the main coolant volume through a vent hole at the other end of the test well. The cooling flow rate and temperature would both be controlled. This would allow more direct control of the sample environment for cases were moderate cooling was required but the precise control provided by an external cooling loop was unnecessary. Depending on the cooling requirements, cooling flow could be supplied near ambient temperature or at the test reactor core inlet temperature.

This would utilize the same system as the purge helium which is used to control the control rod drive and circulator motor environments. Clean helium from the helium purification system is provided to all purge locations, ultimately discharging to the main coolant flow stream.

5.1.6 Access for small samples to be rapidly inserted/removed

Finally, direct access would be provided for small samples that were to be inserted or removed rapidly while at power. This would typically involve very small samples, so the test well path could be quite curved. This provides significant design flexibility in where the required penetration(s) are located on the reactor vessel.

It is anticipated that samples would be transported pneumatically, although other schemes could be considered.

This scheme is most easily implemented with test capsules immersed in air. This allows ready access for insertion and removal. Cooling air flow could be provided if necessary for basic thermal control. Of course, the use of air would require a thicker test well wall with the resulting neutron flux penalty. As an alternative, the samples could be immersed in helium in order to allow a thinner test well wall. This approach would require a double valve mechanism for inserting and removing specimens. While this is more complicated, it is certainly possible. Cooling would still be possible using helium purge flow.

In principle, these approaches could be applied to most of the previously discussed access schemes. The primary limitation on the insertion and removal of samples at power is the relative impact on reactivity, and the resulting changes in local and global power levels and neutron fluxes. If the effects are significant, then they must be within the capability of the reactor controls to compensate for the resulting changes.

5.2 Central Access Channel Structure

A central access channel is a major design feature that could offer several benefits for test well access to the core. This channel is a metallic structure which extends down through the center of the reactor, replacing a significant fraction of the central reflector (slightly more than the central column of reflector blocks). Ideally, it would pass all the way from the central penetration on the top head of the reactor down to the bottom head of the reactor. The general arrangement of this concept is illustrated in Figure 6-2.

The central access channel provides a conduit for test wells and instrumentation to reach the central core area. It provides improved access to these structures for maintenance and reconfiguration activities. It also provides rigid support for these structures and a large volume allowing unusual test articles to be placed in the central core region.

The central access channel provides space for several test wells to pass down to the center of the reactor. These test wells would share a common top head penetration at the entrance to the access channel. For test wells requiring straight access, they could simply pass straight down to the center of the reactor, or they could be canted slightly in order to pass into the remaining central reflector to reach the active core. A preferable arrangement would be to introduce a modest bend in the test well near the middle of the core. This would allow the test location to be adjacent to the active core, while simplifying the arrangement within the central access channel and minimizing the impact on the central reflector graphite structures.

The central access channel has a cylindrical shell structure which is designed to provide rigid support without relying on adjacent graphite structures. It provides more rigid structural support for test wells over the long distance from the reactor vessel top head penetration down to the core region of interest.

The central access channel is not part of the reactor pressure boundary. The pressure boundary is formed by the penetration closure at the top of the access channel and by those test wells extending from that penetration closure down into the reactor via the access channel. The main structure of the access channel is vented to the primary coolant flow in order to balance the pressure across the channel wall. The venting openings are sized to provide moderate bypass flow through the access channel for cooling of internal structures.

Graphite and boron carbide structures and other materials would be inserted at appropriate locations to provide shielding. The quantity and placement of this shielding is a detail to be worked out in the design process. It is required, but the detailed configuration is not critical to the concept.

5.3 Core Design Flexibility

A key aspect of the prismatic block HTGR is its core design flexibility. Each fuel element can be optimized with a unique enrichment and packing fraction loading pattern. In addition, during refueling, fuel elements can be shuffled axially as well as laterally, since the core contains several layers of fuel elements. This flexibility is used to optimize the core performance and the cycle length for the full-sized commercial HTGR. But for the advanced gas-cooled test reactor, this flexibility takes on much more significance.

For the test reactor, the inherent design flexibility of the core is needed to accommodate the wide range of test specimens which could be irradiated in the reactor. In principle, each test location could contain test articles which either strongly increase reactivity or decrease reactivity. And given the large core height and the fact that most test wells will contain various different experiments at different positions along their length, this local reactivity contribution could vary significantly along the height of the core. Therefore, the ability to change the axial zoning of local parts of the core by rearranging existing fuel blocks is a strong advantage for the prismatic block test reactor.

For most test campaigns, the goal will be to envelope the minor reactivity contributions of the individual tests within the substantial performance margins of the standard core design. One strategy might be to optimize the core to routinely put more reactive tests in certain locations and less reactive tests in other locations in coordination with the detailed core design. But if the overall test loading strategy needs to be updated or if some unusual test with a larger impact on reactivity must be accommodated, then the detailed core design can be adjusted in a variety of ways to accommodate this. It is anticipated that over the lifetime of the test reactor, a variety of partially irradiated fuel elements will be accumulated which will be available to fill out a variety of optimized core designs.

Given that the individual test wells in the reflectors will be removable, the prismatic block core structure provides an even greater degree of freedom. If necessary for certain core configurations, the test wells in a particular column can be removed, and all the reflector or fuel elements in that column can be replaced with fully fueled elements to increase local core reactivity. Similarly, if it was necessary to reduce the local reactivity in the vicinity of a particular test well located within fueled elements, those fuel elements could be replaced with reflector blocks. If other elements in that column were still fueled, then the reflector block would still include the normal fuel element coolant holes to provide cooling for the remaining fuel blocks.

Of course, this broader design flexibility must be taken into account when designing the overall control rod layout for the reactor. And it would also have to be enveloped by the plant safety analysis, particularly the conduction cooldown calculations for passive heat removal.

5.4 Multiple Test Wells in Single Column

It is recommended that the option be maintained to locate multiple test well positions within a single reflector or fuel element column. Figure 5-1 illustrates prismatic blocks with one, two and three holes for test wells. This increases the overall test volume in the core. Importantly, it does this with minimal impact on the pressure vessel penetration situation, since these test wells would all enter through a

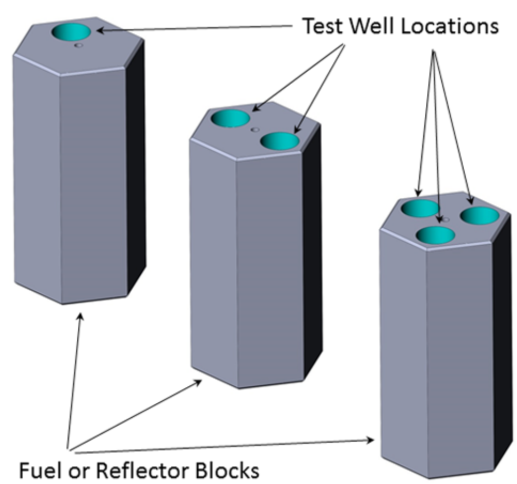

Figure 5-1: Multiple Test Well Configurations

shared vessel penetration. The penetration closure would have to accommodate each of the three test wells, but this is much more manageable than having three independent vessel penetrations.

Obviously, the flux in each of the adjacent test wells would not be the same. Particularly for locations within the outer reflector, there will be significant gradients in both fast flux and thermal flux (not necessarily the same). For locations within fueled blocks, the gradients should be smaller.

The challenge for the experiment and core designers will be to assign individual tests to the channels with the most appropriate flux for that test. This optimization process would also consider the impact of the individual tests on the core flux distribution. But in any case, the benefit of the large test volume offered by placing multiple test wells in a single column/penetration location should not be ignored.

5.5 Cooling of Metallic Incore Structures

Normal HTGR design practice is to exclude metallic materials from the reactor core to the extent possible. Normally the only metallic structures within or immediately adjacent to the active core are the control rods. This is done to prevent damage to any components due to both normal operating temperatures and accident temperatures. There are other metallic components in the vicinity of the core, such as the upper core restraint elements above the top reflector and the hot duct liner downstream of the reactor outlet plenum. However, these components are not exposed to the same temperature extremes as at the active core, particularly during accidents. Also, an incore movable flux detector is inserted into the central reflector periodically, but it does not experience active core temperatures.

For metallic structures in the vicinity of the core (i.e., control rods and movable detectors), cooling is provided to maintain the temperature of these components within a reasonable range. A controlled amount of bypass flow from the reactor inlet plenum is diverted down each control rod channel to provide the necessary cooling. This coolant flow exits through an orifice to the reactor outlet plenum.

The AGTR will have numerous additional metallic structures in the vicinity of the active core to accommodate all the test wells and the test loops, capsules, and instrumentation within those test wells. It is strongly suggested that the normal HTGR design practice be followed, and that all of these metallic structures be provided with superficial cooling using controlled bypass flow in the gap between the surrounding graphite structures and the test well boundary. The blocks containing the test wells would be designed with small raised tabs to maintain the required minimum gap distance between the graphite channel wall and the test well outer surface.

This bypass flow would prevent excessive metallic temperatures during normal operation. As discussed previously, individual test wells could have separate dedicated cooling capabilities depending on the specific requirements of the test articles in those test wells. Those dedicated cooling capabilities could be in the form of externally controlled cooling loops with alternate fluids, etc., or they might be supplemental helium purge flow within the test well. In any event, providing basic cooling of the metallic structure with core inlet bypass flow will ensure that the metallic structure is protected from extreme temperatures during normal operation. The design of graphite core structures is such that eliminating this bypass flow would be very difficult in any event. The design is best optimized by making intentional use of it.

The temperature of metallic structures within the core is also of interest during accident conditions when active cooling is unavailable. In a full-sized HTGR core, these temperatures could exceed 1600°C at some locations near the inner radius of the active core. Fortunately, for the smaller test reactor core, the accident temperatures will be much lower. This is due to the much smaller ratios between the decay heat generation in the active core, the total heat capacity of the core including reflectors, and the vessel surface area. AREVA has not performed detailed passive cooling calculations for the test reactor core geometry, but based on past experience these temperatures would not be expected to significantly exceed peak core operating temperatures (e.g., 800-850°C). Reflector temperatures would be lower. This conclusion is consistent with preliminary analyses performed by INL [1].

Therefore, several acceptable material candidates should be available for test well structures. AREVA briefly considered a few material options as part of this test reactor evaluation. Some of the options considered were Alloy 800H, various stainless steels, titanium, aluminum-titanium alloys, and carbon-carbon composites. The conclusion of that outcome confirmed that reasonable materials are available. Selection of the best material would be the focus of a more detailed study considering the specific neutronic, thermal, and structural characteristics of each material relative to the local structural demands. Such a study would be part of the conceptual design phase of the test reactor.

For now, the most obvious choice is Alloy 800H which is the current material reference for metallic control rod parts, core support structures, upper core restraint elements, and hot duct liner components in the full size steam cycle modular HTGR. It is included in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for use in HTGRs. Alloy 800H is compatible with long-term operation at the reactor inlet temperature, and it can accommodate accidents with peak temperatures in the necessary range. Therefore, Alloy 800H provides a suitable basis for initial gas-cooled test reactor scoping studies.

5.6 Impact of Increased Bypass Flow

Significant bypass flow is a fundamental part of HTGR core design. For a typical full-sized modular HTGR, core bypass is expected to vary in a range from 5% to 30% over the reactor lifetime. The total bypass flow would be expected to be higher for a smaller test reactor. This is partly because the fraction of reflector columns (and gaps) in the overall core is higher for a smaller active core. In addition, the test reactor has several metallic structures which pass through the graphite reflector and core structures. The passages through the graphite for these metallic structures will further increase the bypass flow.

Bypass flow has various impacts on the core design, but a key consideration is the increase in local operating fuel temperatures which results from the diversion of coolant from the fuel elements. The core design and the selection of target operating temperatures must take this into account. Generally speaking, for the same core power distribution, if the bypass fraction is increased, then the nominal core temperature rise must be decreased in order to maintain the same peak operating fuel temperature.

Therefore, initial designer guidance would be that the nominal reactor outlet temperature should be reduced for the test reactor compared to a similar full-sized modular HTGR. The reference Steam Cycle HTGR (SC-HTGR) core outlet temperature is 750°C, so an initial target core outlet temperature for the gas-cooled test reactor would be in the range of 650°C-700°C. This value is based purely on designer judgment, since detailed bypass and core design calculations have not been performed. But in the absence of such calculations, it represents a prudent starting point for subsequent analysis.

5.7 Regions for Increased Fast Flux Ratio

One of the challenges identified was to achieve the desired ratio of fast to thermal neutrons for some irradiation experiments. One advantage of the large core size of the HTGR is that it allows modifications to locally enhance the fast flux ratio.

One approach would be to modify the ratio for a specific sample by tailoring the flux using appropriate shielding materials. While this would reduce the overall flux, it may be the best approach for some tests.

Another approach would be to create a zone in the reactor with less local moderation and a locally hardened spectrum. Given the neutron migration distances within the HTGR core, such a zone would have to be larger than a typical irradiation sample size. It is envisioned that such a zone could be created by introducing a “moderation window” within a part of the reflector. This could be done either by replacing graphite reflector blocks with blocks fabricated from an alternate ceramic material that is not an effective moderator or simply by replacing the standard graphite reflector block with an alternate graphite blocks containing large cavities to provide an extremely high void fraction (and minimal moderation). This concept is noted in Section 6.0, which describes a potential design configuration of an HTGR test reactor.

5.8 Potential Benefits of a Larger Core

Generally speaking, the smaller the reactor core, the more severe the reactor vessel top head crowding issue will be for the control rod drive and test well penetrations. Therefore, AREVA considered the 12 column core configuration evaluated by INL [1] as the minimum practical reactor size. Even for that configuration, the crowding on the top head is apparent Figure 4-1.

As will be discussed in the next section, AREVA actively evaluated a slightly extended version of the core arrangement which added fuel to the corner columns containing the outer test wells. This is identified as the "12 + 6" column configuration. This configuration may provide some neutronic benefit in providing better flux profiles for the corner test wells. But it does not change the test well or control rod locations, and it does not have any impact on the top head congestion issue.

The next logical step would be to add fuel to the two blocks between each fuel corner block and move the control rods out one row (illustrated loosely in Figure 5-2). This becomes a "12+18" or a 30 column active core. This configuration has not been evaluated in detail, but it is expected to offer greater design flexibility for the top head.

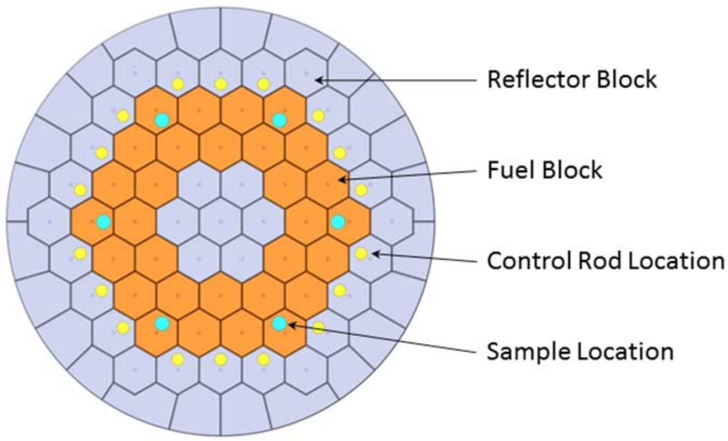

Figure 5-2 : 30 Column Core Layout

Of course, to maintain the desired flux levels, it probably would be necessary to increase the reactor power level for the enlarged core. This would increase plant cost slightly. If the power level were doubled from 100 MWt to 200 MWt, the reactor cost would increase by a few percent, since there would probably be one additional ring of reflector blocks. The cost of the steam generator and the steam turbine would increase more significantly, since their capacity would double. However, the building sizes would not change significantly. Moreover, the potential power generation revenue from a secondary mission would double as a result.

6.0 POTENTIAL DESIGN CONFIGURATION OPTIONS/RATIONALE

This section provides an overview of several features of a specific AGTR design, the MT-HTGR, to illustrate possible solutions for meeting some of the benefits of, and challenges associated with, the use of a high temperature gas-cooled reactor as a material test reactor. These ideas are offered as an example of the suggested approaches, and should not be considered as specific design recommendations.

6.1 Description/Rationale

For this material test reactor application, a small version of AREVA's SC-HTGR is proposed. Like the larger SC-HTGR, the MT-HTGR is based on established technology applied to be consistent with the 12 column core design presented in Reference [1]. The use of technology, which has already been demonstrated in previous operating HTGRs, provides the lowest development risk with demonstrated concept feasibility. The use of a steam cycle system to facilitate energy removal from the reactor system also provides the opportunity for both electricity generation and/or heating applications as secondary missions using proven technologies. Perhaps most importantly, the HTGR has unmatched safety characteristics. Those safety characteristics provide an acceptable risk profile, which eliminates the potential of offsite consequences for all postulated accident scenarios. This is especially important in multi-use sites, such as laboratory locations, where evacuation would be problematic.

Key MT-HTGR parameters are defined in the table below.

Table 6-1: Key MT-HGTR Parameters

Reactor type	Helium-cooled graphite-moderated
Fuel type	TRISO coated particle fuel
Reactor power level	100 MWth
Reactor inlet temperature	325°C
Reactor outlet temperature	700°C
Primary pressure	7 MPa
Core configuration	Annular prismatic block core 12 columns (base design) 8 blocks high
Main steam temperature	550°C
Maximum electric output	40 MWe
Refueling interval (at maximum power)	6 Months

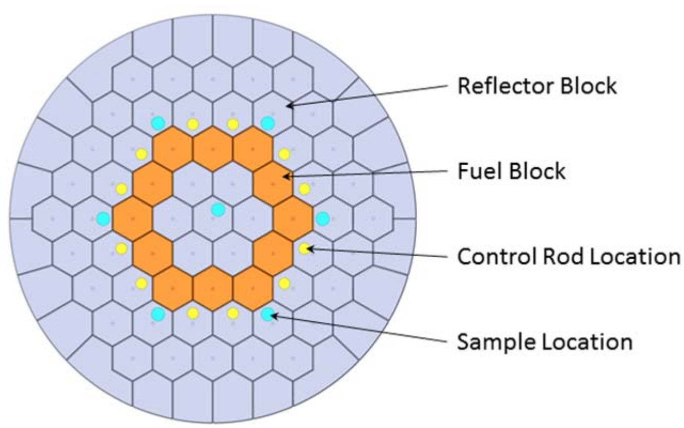

Figure 6-1: Reference MT-HTGR Core Layout

The MT-HTGR concept is designed for a nominal 100 MWth capacity. Its reference design uses an annular core of 12 columns of standard prismatic blocks, each 8 blocks high, as depicted in Figure 6-1. Control rod and sample locations are in the reflector blocks adjacent to the fuel. This configuration results in a fairly high specific power per fuel block, which provides maximum neutron fluence at the sample locations. This configuration retains very large thermal performance margins for the fuel, resulting in enhanced design flexibility for both normal operation and safety. Acceptable fuel temperatures are maintained for all conditions, even without active cooling. Essentially all fission products are retained within the coated fuel particles for all normal operating and accident conditions. Operating fuel temperatures are well below the design limits generally imposed for normal operation of HTGR particle fuel and accident fuel temperatures should not rise significantly above normal operating temperatures.

6.1.1 Sample Location and Configuration Options

Three different types of test locations were considered when developing the strategy for placement of samples within the MT-HTGR reactor. These were:

- Static sample – This type of location serves to house samples that are placed into the reactor and remain in the reactor until a particular fluence, either thermal or fast, is reached or exceeded. They are then removed from the reactor. This type of sample is typified by a material test coupon in which the critical attribute is reaching a particular material damage accumulation (displacements per atom). In many, if not all, cases, insertion and removal can be timed to correspond to planned refueling or shutdown dates.

- Loop Sample – Similar to a static sample, in that a particular overall fluence is required, but with the addition of a controlled sample environment. This environment, which may include specific controlled temperatures and chemicals in which the sample is immersed, is established through the use of a test loop. The test loop is expected to require significant external support infrastructure.
- Rabbit Sample – This type of sample is envisioned as a relatively small size item that requires precise control of the accumulated neutron fluence and rapid placement and removal from the core, for example to accommodate short half-lives of produced radionuclides. This sample is expected to be handled via a pneumatic transport system, a "rabbit tube" system.

Two basic types of sample configurations are envisioned for the MT-HTGR. These configurations provide significant flexibility in that each of the three sample types can be introduced into each configuration, with appropriate consideration during detailed design.

The first sample location configuration is the Test Well. Test wells are located in the first row of reflector elements surrounding the core, on the side of the element closest to the core, as depicted in Figure 6-1, or in multiple locations within the element if desired. The well consists of an upper flange that interfaces with a support and access structure attached to the reactor vessel head. The flange supports a tube that extends through the upper plenum and into the reflector region, terminating with a sealed end at the bottom of the active core as shown in Figure 6-2. The outside diameter of these Test Wells is estimated to be about 11 centimeters. The wall thickness and composition are dependent on the pressure boundary strategy selected.

There are two potential strategies for establishing the pressure boundary for the Test Well configuration. The entire Test Well could be considered a part of the reactor pressure boundary. As such, the wall thickness and composition will be fairly robust resulting in reduced neutron flux at the sample locations. This strategy will allow much easier access to the interior of the test well while the reactor is pressurized, potentially supporting placement and removal of samples during operation.

A second strategy would be to establish the reactor pressure boundary at the flange at the top end of the Test Well and operate with the interior of the Test Well at reactor pressure. This would allow significantly thinner walls on the test well or walls of structurally weaker materials. This would result in higher flux at the sample locations at the cost of more difficult access to the test well, particularly at operational pressures. Access while pressurized would still be possible, but would require a more complex air-lock type device.

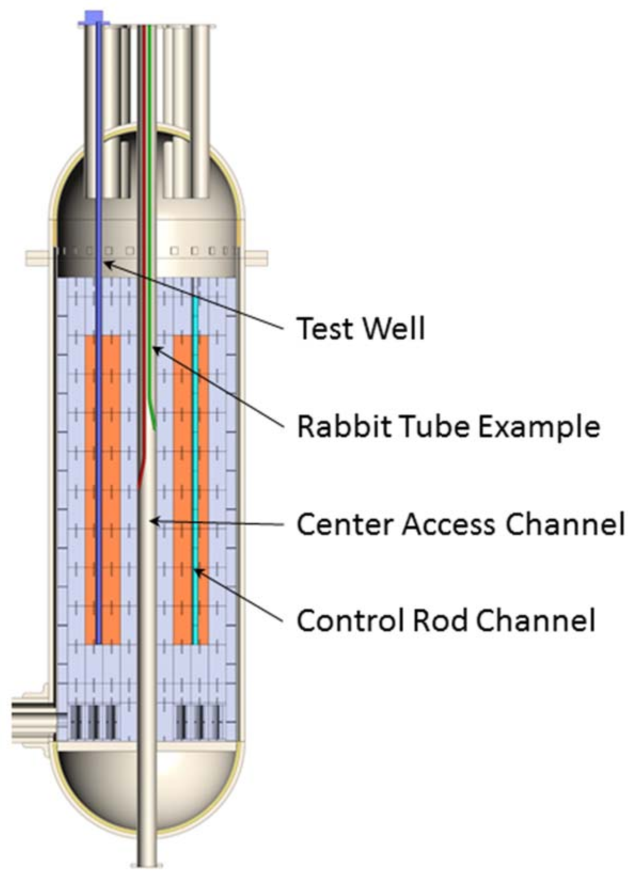

Figure 6-2: MT-HTGR Elevation

The reference design would have six Test Wells, one in each of the reflector blocks at the "corners" of the core. The addition of multiple Test Wells at these locations would increase the volume available for placement of test materials. The wells further from the core would have lower overall flux levels, with different fast-to-thermal flux ratios, which may be advantageous for certain tests. Multiple Test Wells per block would complicate the design of the support structures on the reactor vessel head. These potential configurations are depicted in Figure 6-3.

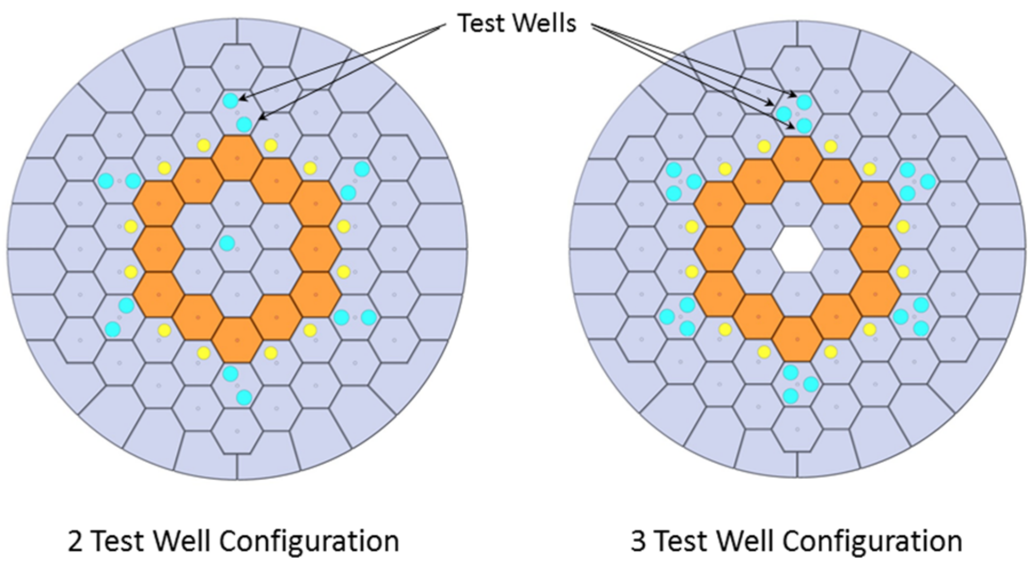

Figure 6-3: Multiple Test Well Configurations

The second sample access configuration is the Center Access Channel, also shown in Figure 6-2 and Figure 6-4. This configuration consists of a cylindrical structure running from above the reactor vessel upper head to below the reactor vessel lower head. This structure is estimated to be up to 45 centimeters in diameter. The large volume available will allow for placement of either large numbers of individual test objects or placement of very large or complex test samples. Access to the Center Access Channel could be from either the top or bottom, adding further flexibility to this concept.

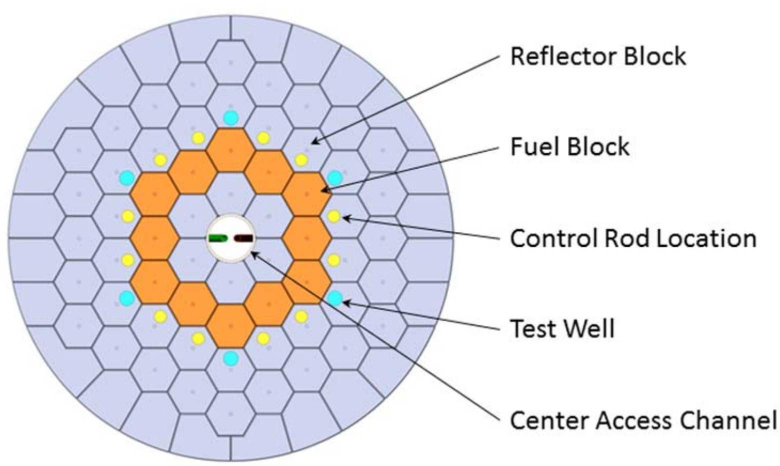

Figure 6-4: Center Access Channel

This is one potential area that permanently installed rabbit tube systems would access the core area. They would traverse through the interior of the column to the core region, through the wall of the column into the adjacent reflector material then terminate close to the surface of the fueled blocks. In this design, the rabbit tubes would also be part of the reactor pressure boundary.

Either the Test Wells or the Center Access Channel could be configured to host test loop systems and their associated support equipment. Figure 6-5 schematically depicts such a configuration housed in a Test Well.

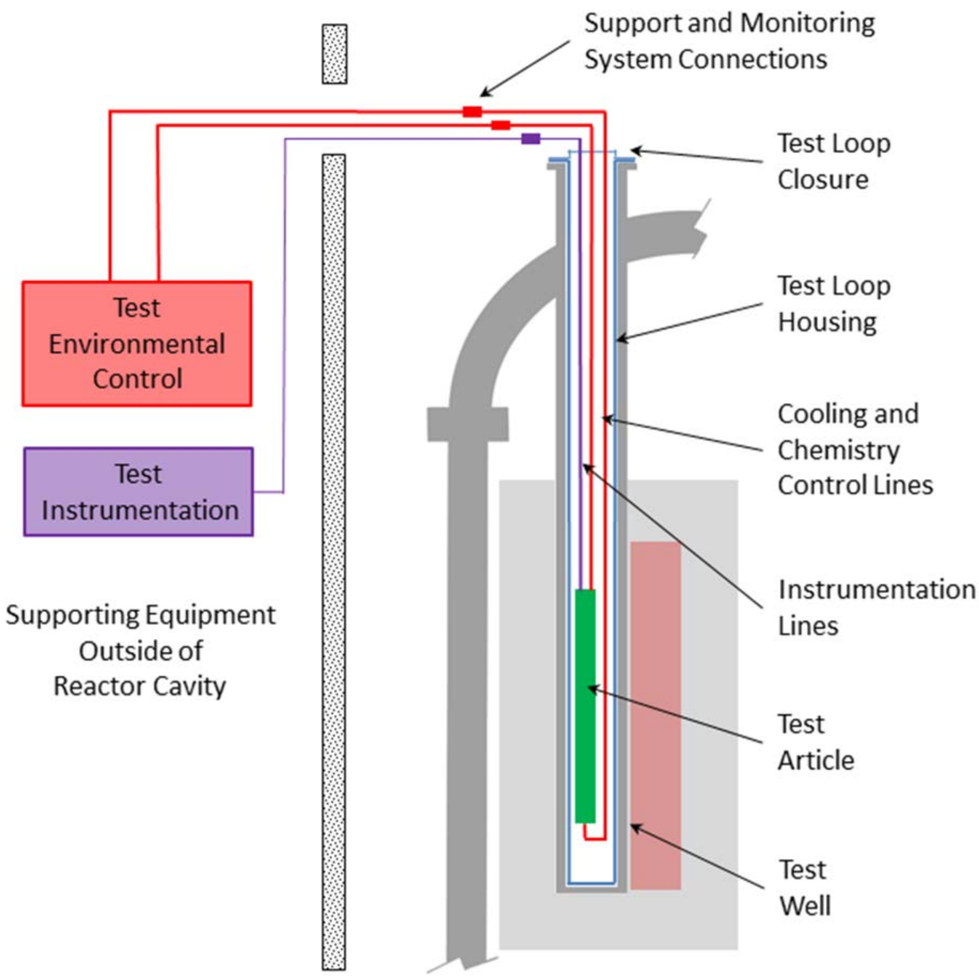

Figure 6-5: Externally Supported Test Loop Configuration

6.1.2 Options for Maximizing Neutron Flux

It is recognized that the MT-HTGR, based on gas reactor technology, does not provide fast neutron flux levels attainable with other reactor technologies. There are several design solutions which can be used, however, to maximize the fast flux at the sample locations and make the MT-HTGR competitive in this area.

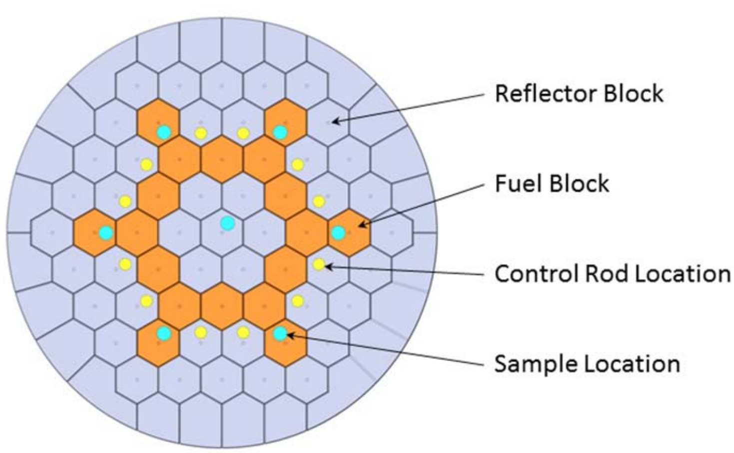

Figure 6-6: 12 + 6 Column Configuration

Figure 6-6 depicts one potential solution whereby the reflector blocks on the “corner” of the core, those containing the sample location Test Wells, are replaced with fueled blocks. The fuel content of these blocks can be tailored to maximize desired neutron flux characteristics. Employing this solution would involve a trade-off with overall core flux levels. By adding more fuel to the core, while keeping the reactor power level constant, the average neutron flux would decrease. It may be possible to partially compensate for this effect by employing various control schemes, such as control drums similar to the existing Advanced Test Reactor at INL which depress or elevate local flux, creating areas of high and low flux within the core.

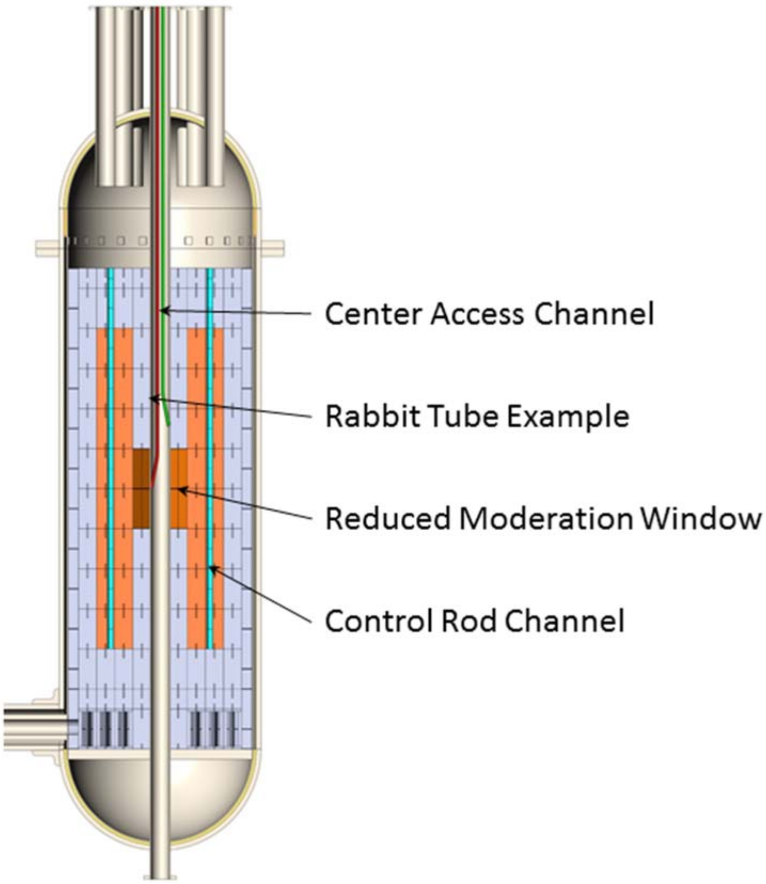

Figure 6-7: Reduced Moderation Window

Another fuel-related solution would be to include a very high power block in one of the fuel columns near the top (inlet) of the core, with most of the remaining blocks in the stack at below average power. This would provide very high local flux, but should still give acceptable fuel temperatures and compatible core outlet temperature in the column of interest. This solution exploits the benefits the 8 block high core in the MT-HTGR compared to other concepts with full length fuel assemblies.

An additional solution would be to locally reduce or eliminate a portion of the reflector in the sample region. This would reduce moderation locally, thus increasing local fast flux. It is estimated that this flux window would need to be several feet in length to achieve the desired fast flux increase. This configuration, applied around the middle section of the Center Access Channel is depicted in Figure 6-7.

Though these solutions will raise local fast flux, perhaps the biggest advantage the MT-HTGR has in this area is its very large available sample volume. While local fast fluxes are lower than in some reactor concepts, and thus the accumulation of fast fluence for an individual sample will be slower, the ability to irradiate a significantly greater number of samples at one time may result in the ability to accumulate a desired fast fluence for an entire set of samples in comparable or shorter times.

6.2 Summary of MT-HTGR potential capabilities

As described in Section 6.1, the MT-HTGR will have the capabilities necessary to meet the mission of the next material test reactor. Key capabilities of the reactor are:

- Ability to handle multiple test loop coolant types representing the major advanced reactor systems. By using the Test Well and Center Access Channel, experimental loops can be irradiated which utilize externally supplied sources of cooling and environmental control materials prototypic of water, sodium, molten salt, or other reactor coolant systems.
- Ability to irradiate multiple fissile material types representing the major advanced reactor systems. The ability to tailor fast and thermal neutron flux levels in different test locations provides the opportunity to mimic most advanced reactor system currently under consideration.

- Thermal compatibility with external test loops. The MT-HTGR inherently provides a significant range of potential test temperatures, from the reactor inlet temperature of 325°C to the core outlet temperature over 750°C. (Temperatures slightly above core outlet temperatures are possible due to the bypass flow impacts, which result in local fuel element temperatures higher than the bulk core outlet). Externally supplied heating or cooling loops can modify this temperature range to accommodate most expected needs.
- Placement of test samples and test loops without reactor depressurization. The Test Well and Center Access Channel concepts foster the ability to access sample locations without reactor system depressurization.
- Placement of test samples at power. The Rabbit Tube system provides a means to quickly and precisely place samples when the reactor is at power. Under other particular conditions, larger samples may also be able to be placed at power in the Test Well and Center Access Channel spaces.
- Irradiation of substantial HTGR samples in an actual HTGR. The MT-HTGR would be the only HTGR-based material test reactor in operation. As such it would provide a unique platform to support testing of materials and components for this major advanced reactor type. Such tests would not be limited to the sample locations described in this report, but would also be able to be integrated into the design of special fuel blocks and irradiated in truly prototypic HTGR conditions.
- Potential to irradiate large test articles. The very large available test volumes, particularly in the Center Access Channel, would allow placement of test articles of significantly greater size than competing test reactor concepts. In fact, in the extreme, one could imagine placement of samples up to a full boiling water reactor (BWR) assembly or quarter sized pressurized water reactor (PWR) assembly in size.
- Very large overall sample volume for higher overall irradiation throughput than smaller high-flux test reactor systems. This higher throughput would significantly reduce wait times for reactor time while also providing space supporting beneficial reactor secondary missions, such as radioisotope production.

7.0 SUGGESTED ANALYSES/INVESTIGATIONS

7.1 Fundamental Requirements

There are many criteria that affect the design of a gas-cooled test reactor. The most basic are the volume of test space desired, the neutron flux levels desired, and the minimum refueling intervals desired. This chapter describes some of impacts of these requirements and what analyses/investigations will be necessary to demonstrate that these can be met.

7.1.1 Experiment Volume

One of the most fundamental requirements that affect the reactor core design is the volume required for experiments. A large enough reactor core is required to accommodate the size and number of experiments to be irradiated. Another factor affecting the core size is that it must be large enough radially to accommodate the number of access ports, since access port density on the vessel head is limited by mechanical design constraints.

Once the volume requirement is met, increasing the core size beyond that is detrimental since it merely increases the amount of heat generated and the amount of fuel consumed with no improvement in flux levels. Therefore, it is recommended that the volume requirements be clearly established up-front and that the core be sized to meet that requirement.

7.1.2 Neutron Flux Level

The neutron flux level is fundamental to the design because it affects the throughput of the test reactor. A lower flux level will require longer irradiations to accumulate the fluence desired. Therefore, the higher the flux level, generally, the more desirable the design. The flux level is proportional to the fission density of the fuel near the experiment. The core average flux level is maximized by minimizing the amount of fuel to produce a given core power level. This also has to include power distribution effects to ensure local fuel power density limits are maintained. After that, other design optimizations should be pursued to maximize the local flux level for some experiments. These include local enrichment increases (within the bounds of the supporting fuel qualification database), burnable poison placed in other regions to shape the flux in the core, and local moderator increases to increase local fuel reactivity. All of these are suggested for inclusion in further investigations.

7.1.3 Minimum Refueling Intervals

The capacity factor is mainly affected by the refueling time. The longer the operational interval between refuelings, the less time is spent in the shutdown condition, given a fixed time for shutdown, refueling, and startup. The minimum refueling interval is set to achieve the target capacity factor. The refueling interval and the core size together affect the fuel energy extraction, or burnup. The burnup is limited by fuel design and licensing considerations and may be a limiting factor in accommodating a desired refueling interval. It is suggested that the maximum allowable fuel burnup be set, based on the TRISO particle burnup limits, and then the core lifetime will be set based on that limit.

7.2 Additional Optimizations

A number of additional optimizations are possible. These are described below.

7.2.1 Control Rods

Reactivity control rods will play a major role in the operation of the gas-cooled test reactor. These will perform the functions of startup/shutdown and burnup compensation. It is suggested that the rod locations be identified and their sufficiency be included in future design evaluations. This should address shutdown, startup temperature compensation, fission product compensation such as xenon and samarium, and radial and axial power shaping capabilities.

In order to accommodate refueling operations, the total control rod worth requirements must be satisfied with one pair of control rods completely removed (plus any additional stuck rod requirements).

In addition, shim rods may be of use in local power distribution control and should also be investigated as a further possible refinement.

This evaluation should be a high priority, since the control rod worth (and the resulting total number of rods required) is the key interface parameter between the neutronic design of the core and the reactor vessel top head crowding issue. This evaluation also directly affects the evaluation of potential cycle length, because the total control rod worth determines how much excess reactivity can be designed into the core.

7.2.2 Test Well Approach

There are two general approaches to inserting experiments into the flux of a gas-cooled reactor. The first approach is to provide robust test wells that are constructed of a suitable metal alloy and serve as a pressure boundary. These types of wells provide both the benefit of supporting easy experiment access, since they are at atmospheric pressures and therefore require no primary gas lock for access, and also provide more flexible experiment cooling than only using primary inlet coolant. The negative aspect of this approach is that it requires a thick metal wall between the experiment and the fuel.

The other approach is to place the experiments inside the primary pressure boundary and use primary gas coolant to cool the experiment. This has the benefit of eliminating much or all of the material between the experiments and the fuel. The negative aspect is that it requires a double valve system or a pressurized transport cask to load and unload the experiments into the reactor. It should be noted that this approach has been a proven commercial approach for gas-cooled reactor refueling [2][3].

It is suggested that the impact of steel in the test well walls of the first approach be evaluated against the impact of a non-steel metallic or composite structure that could be used in the second approach relative to flux levels, flux spectrum, and core reactivity and power distribution.

7.2.3 Large Central Access Channel

Another option for experimental access to the neutron flux is by means of a large (approximately the size of a fuel block) access port in the center of the core. This area is free of control rods and would be an excellent location for access through the pressure vessel head. It is suggested that the effect of one, and multiple, experiments on the flux level and spectrum, as well as core reactivity and power distribution, be evaluated.

7.2.4 Test Location Assembly Optimization

In general, commercial gas-cooled reactor fuel is slightly over-moderated. Because of this, the flux in an assembly can be increased by placing it adjacent to more than one reflector block. This opens up the possibility of placing a fuel block containing a test well in a location that has 3, 4, or 5 adjacent reflector blocks. One approach to doing this is described in Section 6.1.2. It is suggested that the effects of this strategy be investigated as a means of increasing the neutron flux level in the experiment.

7.2.5 Neutron Flux Windows

Once the fundamental parameters are used to determine core size and the Test Well approach and Central Access Channel concepts are evaluated, it is suggested that the effectiveness of the application of "neutron flux windows", as described in Section 6.1.2, should be evaluated. This may provide a way to increase the flux levels for some experimental locations.

7.2.6 Effects of Experiments

The presence or absence of experiments in the test wells will affect the core reactivity, power distribution, and, to some extent, control rod worth. It is suggested that this be investigated using experiments of fissile materials (such as fuel pellets), absorbing materials (such as burnable poisons and actinide targets), and structural materials (such as steels and carbon-carbon composites).

7.2.7 Reflector Thickness

Often the reflector thickness is minimized in order to minimize the size, and therefore the cost, of the reactor vessel. However, for this particular size vessel and pressure, this should not be assumed. It is suggested that the relative cost as a function of vessel diameter should be obtained from a vendor. This cost should be evaluated against the benefits of a thicker reflector, which could include longer cycle life, lower enrichments and fuel costs, and lower vessel fluence.

The selection of the optimum reflector thickness for a modular HTGR core must also take into account the impact on control rod worth (generally enhanced by thicker reflector) and the impacts on conduction cooldown behavior (a thicker reflector adds more thermal inertia to the system and is generally beneficial).

8.0 QUESTIONS AND COMMENTS

In the course of reviewing the preliminary test reactor analyses performed by INL [1], several observations were noted. Key observations are noted below for the benefit of the analysts in current and future work. No response to these comments is necessary.

1) Draft test reactor metrics and criteria were provided as an input to this work (see Appendix A). However, weighting factors for the criteria were not initially indicated. Obviously, different relative weightings on the individual criteria will suggest different design solutions.

Late in the preparation of this report, tentative draft weighting factors were indicated. Those weighting factors imply that the relative importance of secondary missions may be higher than originally anticipated. This underscores the importance of the substantial power generation capability of the MT-HTGR concept, though existing regulations regarding secondary missions for test reactors will have to be addressed in order to fully utilize this benefit.

2) All test well locations should be offset from the center of reflector or fuel elements in order to leave the central grapple attachment point for the fuel handling machine. This is a minor point, but it adds to design credibility.

3) In optimizing the core design to control local power peaking, the initial INL analyses have considered burnable poison and reduced particle fuel packing fractions. Reduced enrichment is another key tool in controlling power that will certainly have to be used. Using only reduced packing fraction in high flux locations puts more burden on peak particle powers. Better results can be achieved by also using reduced enrichment, resulting in more uniform individual particle powers.

   This is AREVA's approach for SC-HTGR. Two or three different enrichments should be sufficient. We do not anticipate using the multitude of different enrichments that are typically used in an LWR fuel assembly.

   Traditionally, fuel particle size is optimized to match the enrichment. However, to keep things simpler, one can just use the standard particle currently undergoing qualification in the Advanced Gas Reactor (AGR) particle fuel development program, but with reduced enrichment. (So, at this time, AREVA would NOT increase the particle size for the lower enrichment particles.)

4) For normal HTGR core design, the minimum gap between blocks is set to be 2mm. This is the required tolerance for fuel handling machine operation, taking into account block bow, etc.

   Sensitivity calculations considering a range of gaps sizes should consider from 2mm to 4mm (or larger).

5) The most important calculation that must be done is an estimate of control rod worth. Until this is done, there is no assurance that the core layouts, the top head arrangement, the estimated fuel burnup/cycle length, or the flux shaping capability will work. For the initial control rod design, an assumption can be made that the control rods are just cylinders of boron carbide. (These would be placed within Alloy 800H cans roughly three inches in diameter.)

6) In doing rod worth evaluations, note that the rod worth must be sufficient to take the whole reactor down to cold conditions, with one full set of rods within a one-sixth core segment removed (for refueling access) while still maintaining required shutdown margin.

7) Some of the indicated metallic operating temperatures in Section 3 of Reference [1] seem high. Allowance for bypass flow cooling of all metallic materials should be the normal design approach.

8) Somewhere it should be noted that the irradiation throughput of test reactor is not simply determined by flux. In simplistic terms it is the product of the flux times the available test volume. Assuming a reactor with lower flux but very high volume, it might take a given test longer during the actual irradiation, but the wait time to get into the reactor might be significantly less. This could reduce total time interval from initial irradiation request to obtaining final irradiation results.

9) A peak power limitation of 400 mW/particle and a guideline of 200 mW/particle are both significantly above normal design practice for commercial HTGR design, even if they are within AGR limits.

10) For a detailed prediction of steady state core temperatures, cross flows must be taken into account. Because of cross flows, net flow in fuel coolant channels will vary by several percent between the top of the core, middle of the core, and bottom of the core. Coolant tends to flow out of fuel block

channels at the top of core and back into the fuel block channels near the bottom of the core, bypassing the middle of the core. (This is not counting the normally reported bypass flow which goes straight to the reactor outlet plenum.)

11) The axial temperature profile suggested by Figure 3-3 of Reference [1] is not consistent with typical depressurized conduction cooldown (DCC) behavior. Normally, there is much more variation between the center of the core and the periphery. Perhaps this is due to the assumed axial power profile. In any event, it would be interesting to see a plot of this covering 1-2 days.

12) Figure 3-8 of Reference [1] is interesting in that the Reactor Cavity Cooling System (RCCS) power never exceeds the core power. Normally RCCS power always crosses above the core power around the time of peak vessel temperature.

13) Notwithstanding the behavior noted above, the overall conclusion of the DCC evaluation is consistent with the normal expectation for a core of this size which is that DCC would be clearly acceptable for peak and average fuel temperatures.

14) It would be interesting to consider some bounding local flux and power increases/decreases due to the presence of possible irradiation experiments in the overall core criticality calculations. This would give an indication of the potential perturbation of the core power, power distribution, and control rod worth requirements. This would confirm that higher experiment densities in the core do not significantly affect the flux levels.

9.0 CONCLUSIONS

An advanced high temperature gas-cooled reactor is an innovative solution for providing irradiation test services. Such a reactor provides the capability for irradiation volumes significantly larger than other concepts. It is based on mature HTGR technology. This minimizes project risk. The large safety margins of the HTGR are important, because they provide greater latitude to accommodate a wide variety of irradiation tests. The HTGR also readily supports potentially important secondary missions such as power generation and isotope production to offset operating costs and reduce the net price of irradiation services.

Because of the large irradiation volume available, the reactor can accept a large number of conventional samples for simultaneous irradiation. More importantly, it can irradiate relatively large intact structures. Such a capability is important for the development of advanced composite structures for reactor applications.

The reactor can also accommodate multiple test loops with externally controlled coolant conditions. This enables the appropriate coolant and temperature environment to be provided for individual tests.

Of course, as for any advanced concept, there are challenges that must be addressed to develop an advanced gas-cooled test reactor.

One key challenge is to obtain the desired flux magnitude and spectrum for individual tests. INL has performed scoping calculations examining the potential performance of an HTGR test reactor [1]. The initial results are encouraging. While achieving high flux levels is a challenge, there are alternatives which could provide acceptable flux performance. More importantly, when the flux level and the potential irradiation volume are considered together, the irradiation throughput of the reactor is impressive.

A second key challenge is providing access through the reactor pressure vessel to reach the intended test locations. There are a variety of possible irradiation locations and a variety of access paths to reach those locations. Nonetheless, for any reasonable configuration, crowding of the reactor vessel top head penetrations for control rods and test well access is expected to be an issue. This is particularly the case for the smaller cores preferred for the irradiation reactor neutronics performance. Preliminary evaluation suggests that a workable top head configuration is achievable, but it may require the penetrations to be bored in a single large forging instead of individual vessel nozzles.

Therefore, slightly larger core configurations may be preferable. This would necessitate increased core power level in order to maintain target neutron flux. This would increase plant cost slightly, but it would further increase available irradiation volume and it would increase potential secondary mission revenue in direct proportion to the power increase (with only a small increase in plant operating cost).

A key point of intersection between top head penetration crowding and the optimization of the core design for best irradiation performance is the required control rod pattern. The control rod pattern is critical to the neutronic design of the core and to the layout of the top head penetrations. For the irradiation test reactor, the control rod pattern must provide sufficient total worth to cover the rise in power from cold to hot conditions, fuel depletion over the core life, operational flexibility for reactor control, required shutdown margin, and additional flexibility to accommodate the wide range of reactivity contributions from different irradiation tests, all while allowing for one rod pair to be removed for refueling. Therefore, it is extremely important that the control rod pattern options be evaluated as part of the initial scoping calculations for the concept.

The results of the initial evaluation of the HTGR as an advanced irradiation test reactor are promising. This concept appears to be a reasonable candidate to provide a wide variety of irradiation services. Additional scoping evaluations to confirm this conclusion are needed and ongoing at INL.

10.0 REFERENCES

1. INL/EXT-15-36340, “High Temperature Gas-Cooled Test Reactor Options Status Report”, J. W. Sterbentz and P. D. Bayless, August 2015.
2. C. Forsberg, J. Richards, J. Pounders, R. Kochendarfer, K. Stein, E. Shwageraus, and G. Parks, Development of a Fluoride-Salt-Cooled High-Temperature Reactor Using Advanced Gas-Cooled Reactor Technology 112, p. Transactions of the American Nuclear Society (2015).
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APPENDIX A: EXCERPTS FROM DRAFT TEST REACTOR METRICS FOR DOE ATDR STUDY

2.2 Goals, Criteria and Metrics

2.2.1 Background

A decision analysis approach is used to develop goals, criteria and metrics and associated weightings against which to evaluate the options. This section will describe how criteria and metrics were developed. Expert judgment was used to elicit criteria and metrics from large group of scientists and engineers from the nuclear community spanning industry, national labs and universities. The initial list was based on previous studies performed by Gen IV participants and the recent NE-5 Fuel Cycle Options Study appropriately modified for the goals here. A workshop was held in April 2015 to obtain input from the communities noted above that allowed the study team to make further refinements. Thus, two different sets of goals/criteria were developed: one for a demonstration system and one for a test reactor.

2.2.2 Desirable Outcomes and Requirements

As part of the process of establishing goals, criteria and metrics for a test and/or demonstration reactor, a number of participants in the workshop felt strongly about certain attributes that each reactor should have. However, when examining these items as part of the development of detailed metrics and weighting (see Section 2.3), several goals and their associated criteria and metrics, while important in terms of reactor safety or performance, are generically applicable and would not distinguish any reactor concepts from any of the others. Some goals associated with key features of a successful test or demonstration reactor are not dependent on reactor type (i.e., technology choices) but rather rely on project structures and/or operational paradigms. In other cases, because of the limited time associated with this study, the required level of detail necessary to compare against a quantitative metric would not be available. Thus, some of these items were felt to be desirable outcomes and others were identified as requirements that would be imposed independent of the details of the reactor designs that were evaluated. These desirable attributes and requirements are captured in this section.

Desirable Outcome 1. Test or Demonstration Reactor Project and Operations provides a focal point for United States nuclear energy R&D activities support diverse stakeholders.
The ability of a new reactor to provide a focal point for US nuclear energy R&D is a highly desirable attribute. However, independent of the technology option selected any reactor development activity will be a focal point. In particular such a facility would enable (a) training of next generation engineers and scientists, (b) engagement and access for U.S. industry, (c) engagement with regulator (U.S. NRC), (d) access and coordination with University Programs, and serve as a model for international users and collaborations.

Requirement 1. Test or Demonstration Reactor has a robust Safety Design Basis.

The ability to license the test reactor by NRC to conduct experiments was considered an important pre-requisite as is also the case for the demonstration reactor. Three important criteria identified for the test reactor were (a) the ability to be licensed by the NRC, (b) the ability to tolerate a broad range of upset conditions, and (c) having a safety envelope that would accommodate a wide variety of test conditions. For the demonstration reactor, safety is an important attribute since it will serve as a flagship for the technology going forward. One item was initially identified as important criterion toward this goal: the reactor's ability to tolerate a broad range of upset conditions (in terms of power, temperature, flow and pressure).

These items represent important requirements for each system going forward. However, because the concepts considered started from the Gen-IV concepts, all have the fundamental characteristics that provide a robust safety basis. Furthermore, the level of technical detail available in this study would not be sufficient to quantitatively differentiate among the options. Thus, this criterion was not used at this point in the assessment.

Requirement 2. Safeguards and security

The ability to meet safeguards and security requirements is another important pre-requisite for any test reactor to be built. The initial criteria identified were (a) to have prototypic material accounting by incorporating proliferation resistance features in the physical design and adhering to other relevant IAEA safeguards requirements and (b) to meet current standard for reactor security by including robust security features in the physical design and meeting associated NRC security requirements. Because the concepts considered started from the Gen-IV concepts, all have the fundamental characteristics that can provide for safeguards and security. They were not used in the assessment because they did not differentiate among the options at the point design stage.

Other Requirements

A handful of other metrics were deleted from the list because they were felt not to differentiate among the technology options or the level of detail available did not allow a useful qualitative or quantitative measurement in the assessment. In the area of test reactor metrics, the following items were not considered as metrics at this point in the assessment but were viewed as more valuable requirements once the designs matured: (a) ability to accommodate in-pile instrumentation of experiments (a capability required for all test reactors), (b) maintainability of the system, (c) margin available for future upgradability of the reactor system. For the demonstration reactor similar metrics related to maintainability and upgradability were identified.

2.2.3 Test Reactor Goals, Criteria and Metrics

The following section identifies the goals, criteria and associated metrics used for the test reactor evaluation. The goals, criteria and associated metrics are shown graphically in Figure 2.

Goal 1. Test Reactor provides irradiation services for a variety of reactor and fuel technology options. (Needs to provide necessary gross parameters, both current and potential)

Criterion 1.1. Irradiation Conditions

Rationale: The nature of the irradiation test conditions established by the test reactor is critical to evaluate its ability to meet the reactor and fuel testing needs. Thus, four metrics were established to characterize the irradiation conditions: (a) the magnitude of the fast and thermal fluxes which influence the level of radiation damage that can be accumulated on a test specimen and the rate at which burnup can be accumulated on a fuel specimen, (b) the available irradiation volumes and lengths which dictate the size of test that could be accommodated ranging from small material specimen to a scaled fuel subassembly, (c) the sustainable time at power which influences the number of (undesirable) shutdown/startup transients the experiment will have to experience until it can meet its dose and burnup requirements, and (d) the ability of the reactor to enable the creation of a prototypic and/or bounding test environment that is different from the test reactor environment (e.g., temperature, coolant and coolant chemistry) for fuels and materials testing.

Metric 1.1.1. Flux conditions (fast and thermal)

Note: Test reactors usually have a range of flux conditions within their testing environment to allow flexibility to meet a wide range of needs. In addition, the physical volume over which that flux exists also can vary (and is captured in Metric 1.1.2) For simplicity here, the fast and thermal flux conditions do not necessarily have to occur in the same location within the test reactor. Nor will a specific volume be required. The fast and thermal flux levels will be evaluated individually and the scores averaged to obtain a final numerical value.

Metric	>5 x 10^{15} n/cm^2-s fast (>0.1 MeV) >5x10^{14} n/cm^2-s thermal	5x10^{14} to 5 x 10^{15} n/cm^2-s fast (>0.1 Mev) 1 to 5x10^{14} n/cm^2-s thermal	<5 x10^{14} fast (>0.1 MeV) <1x10^{14} thermal
Score	3	2	1

Metric 1.1.2. Irradiation volumes and length

Note: As with metric 1.1.1 both volume and length will be evaluated separately and the scores averaged to obtain a final numerical value.

Metric	Volume > 10 liters Length > 1 meter	5 to 10 liters volume 0.5 to 1 meter length	Volume < 5 liters Length < 0.5 m
Score	3	2	1

Metric 1.1.3. Maximum sustainable time at power, to provide a time-at-power for a single irradiation (i.e. cycle length)

Metric	> 90 days	45 to 90 days	< 45 days
Score	3	2	1

Metric 1.1.4. Provisions for testing prototypic and bounding conditions (Temperature, Coolant, Chemistry)

Metric	Does the reactor allow for testing at prototypic and bounding conditions	
Score	Yes = 3	No = 1

Criterion 1.2. Support diverse irradiation testing configurations concurrently (accommodate various sizes and tailor irradiation parameters to wide group of simultaneous users)
Rationale: Test reactors have historically provided for extensive flexibility in terms of the number of testing configurations that can be accommodated within the facility. Three metrics were established to characterize the level of flexibility: (a) the number of test locations available for irradiation, (b) the number and type of test loops with cooling systems that are independent of the primary test reactor coolant to enable tailoring of the test environment and (c) the ability to insert/retrieve an irradiation specimen while at power.

Metric 1.2.1. Number of test zones

Metric	> 25 locations	10 to 25 locations	< 10 locations
Score	3	2	1

Metric 1.2.2. Number and type of distinct irradiation test loops each with a different cooling system independent of the primary reactor coolant

Metric	3 or more	1 or 2	None
Score	3	2	1

Metric 1.2.3. Ability to insert/retrieve irradiation specimen while staying at power

Metric	At power (e.g. rabbit)	Limited handling capability	Only at shutdown
Score	3	2	1

Goal 2. Test Reactor will be built and operated reliably and in a sustainable cost-effective manner. (Need to be able to justify initial and long-term expense)

Criterion 2.1. Project Costs and Schedule (including contingency that reflects technical maturity of the concept)

Rationale: Total project cost and construction schedule are important metrics to compare different options.

Metric 2.1.1. Project cost

Metric	< $1.5 B	$1.5 – 2.5 B	> $2.5 B
Score	3	2	1

Metric 2.1.3. Schedule - The time from site preparation to first operation

Metric	Within 3 years from site preparation	3 to 5 years from site preparation	Greater than 5 years after site preparation
Score	3	2	1

Criterion 2.2. Operational Costs and Schedule (including contingency that reflects technical maturity of the concept)

Rationale: Test reactor operating cost is important to understand relative to the experiment flexibility that it enables and thus the annual operating cost if a good metric. This operating cost does NOT include any cost recovery revenue, potential products are separately identified in the secondary mission Metric 3.1.1.

Metric 2.2.1. Annual operating costs

Metric	< $75 M/yr	$75-150 M/yr	> $150 M/yr
Score	3	2	1

Criterion 2.3. Reliability of operations

Rationale: The overall availability is a key measure in terms of the number of full power days available per year to provide neutron irradiation services.

Metric 2.3.1. Availability factor

Metric	>80%	60-80%	<60%

Score	3	2	1

Goal 3. Capability to accommodate secondary missions (e.g., electricity, isotope production, etc.) without compromising primary mission of testing fuels and materials for advanced reactor technologies

Criterion 3.1 Identification of Secondary Missions

Rationale: Secondary missions can be useful to offset operations costs and provide a different measure of value of the test reactor

Metric 3.1.1 Number of secondary missions

Metric	More than one	One	None
Score	3	2	1