NRC DOMESTIC TRIP REPORT

Subject:

First Program Review Meeting for the "Basic Research on High Temperature Gas Reactor Thermal Hydraulics and Reactor Physics" Cooperative Agreement

Dates of Travel, Location, and Organizations Visited:

February 23-26, 2009.
Texas A&M University, College Station, TX

Travelers, Title, and Affiliation:

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Desired Outcome:

Hold program review for the OSU/TAMU/UM Cooperative program to review progress and program direction. More specifically, to hold a review of the scaling, design, and instrumentation plan for the OSU integral effects test facility. A secondary objective was to initiate discussions with representatives from the NGNP program (DOE & INL) to explore the possibility of the DOE becoming an official partner in the cooperative agreement.

Results Achieved:

The scaling study for the OSU VHTR integral effects test facility, the High Temperature Test Facility (HTTF), demonstrated that most of the important thermo-fluid phenomena for a depressurized loss-of-forced-circulation (D-LOFC) event both with and without air-ingress could be well simulated with a reduced scale ¼ height facility. The program plan is well thought out and the project is on schedule. Two more scaling and design review meetings will be held before procurement and construction are initiated. The current schedule calls for this decision point at the end of September 2009.

Enclosure

Progress on the other tasks has just begun due to a late start caused by contracting matters between the partner universities. Still, good progress has been made on exploring the need for the coupling of MELCOR and PARCS, and in exploring different transport solvers for possible inclusion in PARCS. Also, work has begun on one of the three small-scale separate effects tests, namely the task on flow and heat transfer phenomena in a pebble-bed. More details of this work are given below in the "discussion" section.

Finally, both the DOE and INL were very interested in the HTTF program and requested that a side meeting be held at the NRC during the RIC to discuss the possibility of their participation in the program.

Background/Purpose:

The revised Advanced Reactor Research Program (ARRP)[1] documents NRC's current assessment of its research infrastructure needs and the NRC's planned safety research to support its review of high temperature gas reactor (HTGR) and very high temperature reactor (VHTR) licensing applications. These include a combined license (COL) application for a VHTR to be constructed at the Idaho National Laboratory (INL) in connection with the Next Generation Nuclear Plant (NGNP) Project, as directed by the Energy Policy Act of 2005, and a potential design certification (DC) application for the pebble bed modular reactor (PBMR). In accordance with the ARRP, the NRC is beginning to conduct the research necessary to help support the licensing review of these potential design applications.

In the past, for first of a kind plants such as the VHTR will be, the NRC has performed confirmatory research in an integral test facility. Specifically, this confirmatory research has focused on the performance of passive safety systems in Design Basis Accident (DBA) conditions and for Beyond DBA events to investigate the potential for "cliffs" in their behavior that would degrade their safety margin. Examples of such integral test programs would be the ROSA-AP600 and OSU/APEX test programs in support of the AP-600/1000 design certification and the PUMA facility for the ESBWR.

In April of 2008, the NRC received an unsolicited proposal from a consortium of universities (Oregon State, Texas A&M, and the University of Michigan) entitled "Basic Research on High Temperature Gas Reactor Thermal Hydraulics and Reactor Physics." This proposal for a cooperative agreement proposed work in three areas:

- Coupled reactor physics and thermal hydraulic modeling,
- Separate effects tests for gas reactor thermal hydraulic phenomena, and
- Integral effects tests for gas reactor thermal hydraulic phenomena.

It was the staff's opinion that this cooperative agreement would provide the NRC with an opportunity to conduct an integral effects test program for the NGNP program in both a cost effective and timely manner. Also, this agreement would aid in both the development and validation of the NRC's evaluation model for the NGNP. Consequently, the cooperative agreement was awarded in late September 2008. Work on the scaling and design of the integral test facility subsequently began in October 2008. For the other tasks, contractual matters between the partner universities delayed the start of work until late December.

This first program review meeting for this cooperative research program had the following objectives:

[1] ADAMS accession no. ML082590538.

- Scaling and Design Review: for the integral effects test facility a total of three scaling and design review meetings will be held before any facility construction is begun.
- Progress Review: for the task involving coupled reactor physics and thermal-fluids modeling, and for the small-scale separate effects experimental program.
- Dialogue with NGNP Partners: representatives from the NGNP program (DOE & INL) were invited to attend this meeting to learn about the status/plans for the cooperative program and to explore the possibility of the DOE becoming an official partner in the cooperative agreement.

A summary of the program review meeting and of the dialogue with DOE/INL is provided below in the "Discussion" section.

Discussion:

The first program review meeting for the "Basic Research on High Temperature Gas Reactor Thermal Hydraulics and Reactor Physics" Cooperative Agreement was held at Texas A&M University on February 24-25, 2009. The meeting was attended by the principal investigators from each of the three partner universities, NRC staff (as noted above), contractors from two DOE laboratories (SNL and ORNL), and representatives from the NGNP program from both the DOE and INL.

A few key meeting results will be presented before a more detailed summary of the entire meeting is given. The key results were:

- The scaling study for the High Temperature Test Facility (HTTF) demonstrated that most of the important thermo-fluid phenomena for a depressurized loss-of-forced-circulation (D-LOFC) event both with and without air-ingress[2] could be simulated with a reduced scale ¼ height facility.
- The scaling and design of the reactor cavity cooling system (RCCS) that will be part of the HTTF should focus on the water-cooled design as that is the direction the NGNP program will be pursuing.
- The scaling of the HTTF for pressurized loss-of-forced-circulation (P-LOFC) events and for normal operation may be improved by using nitrogen as the coolant instead of helium. OSU was directed to examine this possibility.
- The scaling and design studies will be conducted for both prismatic and pebble-bed cores. The vessel design would then accommodate either of the two core modules. As part of this cooperative program, only one of the core modules would be fabricated and tested, namely that chosen as the NGNP design. However, should the DOE end up with a second design, a possible follow-on program would then be possible.
- For the next design review meeting, more information of high temperature instrumentation (thermocouples and oxygen sensors) should be presented.
- A total of three scaling and design review meetings will be held. The second meeting will be held May 27-28 at the NRC.

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[2] Air-ingress testing would focus on the onset of natural circulation flow not on graphite oxidation.

- The usage of point kinetics for the calculation of a D-LOFC without scram would substantially over-predict the peak fuel temperature (by about 250 K). Therefore, coupling of the PARCS reactor kinetics code with the MECOR system analysis code may be necessary.
- The DOE and INL were very interested in the HTTF program and requested that a side meeting be held at the NRC during the RIC to discuss the possibility of their participation in the program.

A more detailed summary of the meeting follows.

NRC staff member Joe Kelly opened the meeting by welcoming the attendees and giving an overview of the program. He stressed that the NRC's primary interest in the integral effects test program was not to provide design information but rather to investigate the performance of the passive safety systems for B-DBA conditions. In particular, to look for possible "cliff effects" in areas not far outside the DBA envelope that would significantly degrade the performance of the safety systems. Next, an introduction to the NRC's evaluation model development program for the NGNP was given to put the code development and validation work pursued in the cooperative agreement in context of the overall program. Finally, an example was given from a recent CFD analysis that indicates current pebble-bed pressure drop and heat transfer models are inadequate for the regions near the bed-reflector boundary. This result was posed as one of the areas to be investigated in the small-scale separate effects experimental program.

The rest of the morning and early afternoon sessions was dedicated to a program review of Task #2 activities in the area of "Coupled Reactor Physics and Thermal-Hydraulics Modeling." Prof. Tom Downar from the University of Michigan gave an overview of this task and discussed the current SCALE and PARCS work in support of the NGNP program stressing both current capabilities and development needs. He detailed the development of computational benchmarks for the application of SCALE/TRITON (lattice physics) and PARCS (neutron diffusion) to gas reactor analysis that will be finalized at a meeting March 23-24 at the NRC. Results of a comparison of the PARCS-AGREE code to the IAEA PBMR-400 benchmark were presented.

Next Prof. Downar gave a presentation on the "Evaluation of Point Kinetics for NGNP VHTR Core Analysis." The impetus for this work was to provide some quantification of the error that may be incurred when using a simple point kinetics model as part of a coupled reactor physics and thermo-fluids analysis. In turn, these results will give us insight as to whether or not the PARCS code will need to be coupled with MELCOR for transient analysis or whether MELCOR using a stand-alone point kinetics model is sufficient. As expected, for a control rod ejection case having a significant flux variation, point kinetics was demonstrated to be inadequate. Unexpectedly, point kinetics was also demonstrated to be inadequate for the analysis of a D-LOFC without scram where multi-dimensional kinetics effects are not important. Indeed, the peak fuel temperature was over-predicted by about 250 K in the point kinetics calculation. This surprising result was determined to be due to the dependence of the xenon worth on the graphite temperature rise during the transient. Consequently, if such transients will need to be modeled with MELCOR (to provide fission product transport capabilities), then a PARCS-MELCOR coupling will be necessary.

Dr. Volkan Seker from the University of Michigan gave the next presentation entitled "High Temperature Gas Cooled Reactor Analysis with PARCS-AGREE." This presentation focused on the AGREE code module of PARCS. AGREE is a modern three-dimensional version of the legacy THERMIX-DIREKT code for gas dynamics and heat transfer in a pebble-bed reactor. A two-temperature porous medium approach is used to provide a detailed thermo-fluids solution that is tightly coupled with the nuclear analysis performed by PARCS. Simulations of both the SANA D-LOFC experiments and the PBMR-400 benchmark were presented. AGREE will be modified to be applicable to prismatic core designs as well.

Prof. Todd Palmer of Oregon State University next discussed “Improving Core Solvers and Accelerating Monte Carlo Eigenvalue Calculations.” He noted that while core analysis techniques for LWRs are well developed, new modeling challenges exist for VHTRs, specifically:

- Neutron streaming in coolant flow channels,
- Core/reflector interface effects, and
- Characterization of burn-up gradients and control rod effects.

To accurately and efficiently model these effects, the fine-mesh diffusion solver in PARCS will be upgraded. First, to improve efficiency, a coarse-mesh finite difference acceleration scheme is being developed. Second, to improve accuracy for the heterogeneous effects in gas reactors, two types of transport solvers are being investigated: a simplified P_3 solver, and a quasi-diffusion solver. Finally, acceleration of large-scale Monte Carlo eigenvalue calculations is being pursued to provide computational benchmarks for the diffusion codes.

Prof. Karen Vierow of Texas A&M University gave the next presentation entitled “MELCOR Modeling of HTGR’s.” First, she presented the results of earlier work where MELCOR was used to model both pebble-bed (PBMR-268 benchmark) and prismatic (GT-MHR) designs. Although the base MELCOR code did not have this capability, it was extended through the usage of control functions to simulate the missing constitutive models. Though this approach did allow for rapid prototyping of some of the models, imperfect coupling with the existing MELCOR models (e.g., radial conduction in a debris bed) led to significant over-predictions of the core temperatures. The remainder of the presentation discussed how the developmental HTGR version of MELCOR will be used in the cooperative program. Specifically, as part of the integral effects test in task #4, MELCOR will be used for: aiding the designers in the placement of instrumentation and verification of the scaling analysis, performing pre-test analyses to guide selection of initial and boundary conditions, performing data analysis and testing of new constitutive models, and post-test analyses for code validation.

Next, Prof. Pavel Tsvetkov of Texas A&M presented an “Overview of MELCOR/PARCS Coupling” that is being pursued under task 2 of the cooperative agreement. A range of different coupling schemes were discussed varying from a non-intrusive low-level coupling to a fully intrusive high-level coupling scheme. In the discussion that followed, the fully intrusive option was eliminated due both to a perceived lack of need and the impact on MELCOR development. Also, the results presented earlier by Prof. Downar indicated that the non-intrusive coupling, where the codes are not really coupled at all but merely run in sequence, would be inadequate for accuracy reasons. The two remaining options for coupling that will be investigated further are:

- MELCOR/PARCS Power Coupling: this would be similar to the coupling scheme used for TRACE/PARCS with PARCS acting as a subroutine of MELCOR and the two codes exchanging information after every time step. MELCOR calculated fuel and moderator temperatures would be passed to PARCS and the resulting power transient returned.
- MELCOR/PARCS-PK Coupling: this coupling scheme would take advantage of the point kinetics solver in MELCOR. Basically both the thermo-fluid and point kinetics solutions would be performed by MELCOR. PARCS would be called periodically to

update the point kinetics parameters used by MELCOR to accommodate effects such as the dependence of the xenon worth on the graphite moderator temperature.

The afternoon session began with a presentation by Randall Gauntt of Sandia National Laboratory on “MELCOR Development for HTGR Applications.” This code development work is not a part of the cooperative program but rather is funded by the NRC under a separate contract. The purpose of the presentation was to improve coordination between the development effort at Sandia and the cooperative program participants that will be using MELCOR both for pre- and post-test simulations as discussed above. The presentation described the new “COR” models that will be used to describe pebble-bed and prismatic cores. A sample calculation for the steady-state conditions of the PBMR-268 benchmark was shown as well as the future code development plans.

Kevin Clarno of Oak Ridge National Laboratory was invited to the meeting to describe some work he has been pursuing under an LDRD funded project with SNL. The title of his presentation was “BRISC Development: Purpose, Plan, Progress, and Opportunities.” He described an exploratory research program to develop a computational platform for the coupling of disparate codes to create a multi-physics modeling capability using advanced numerical methods such as the Jacobian Free Newton Krylov method. This presentation was of interest due to its potential as a tool for coupling MELCOR and PARCS. However, the staff feels that this is more of a long-term research program than the near-term solution needed for the cooperative program.

The final presentation of the first day was given by Prof. Yassin Hassan of Texas A&M and was titled “Pressure Drop and Multi-Scale Flow Structure Measurements in a Pebble Bed Reactor.” This work is being funded under Task #3 of the cooperative agreement, “Separate Effects Tests for Gas Reactor Thermal Hydraulic Phenomena,” and is one of three small-scale separate effects experiments. In this task, a mock-up of a pebble-bed will be used for flow studies and single-sphere heat transfer tests. The bed will have an outer diameter of 0.94 m and a height of 2.0 m. Separate effects tests will be conducted to validate constitutive models for:

- Pressure drop in a cylindrical randomly packed bed.
- Pressure drop in an annular randomly packed bed where the “wall-bypass” effect is more important.
- Convective heat transfer for a single sphere in a packed bed as a function of distance from the bed-wall interface.
- The dispersion coefficient, this is the packed bed analog of the turbulent mixing coefficient that is used in subchannel analysis.
- Radial profile of the bed axial flow rate. Used to determine how the local bed loss coefficient varies with porosity and distance from the bed-wall interface, important for determining the wall-bypass effect.

There are two other small-scale separate effects tests that will be conducted as part of this cooperative program. One to look at the heat transfer in a prismatic core under both prototypic full power and D-LOFC flow conditions that will be initiated later this year. The third experiment, flow studies in the outlet plenum, will be a follow on to the packed bed experiment and be initiated later in the program.

The entire second day consisted of a series of presentations given by Prof. Brian Woods and Brian Jackson of Oregon State University detailing the scaling and design activities for the HTTF integral test facility. A listing and brief summary of these presentations is given below:

- "Task 4: Integral Effects Tests": overview of task 4, including status and goals for this fiscal year. Major milestone is the final scaling and design report, due September 29, 2009.
- "Experimental Objectives and General Scaling Methodology": defines the primary goal as determining the physical dimensions and operating characteristics of a test facility capable of simulating the important flow and heat transfer behavior of a VHTR during a D-LOFC, also gives an introduction to the "hierarchical two-tiered scaling methodology" that is being used.
- "Very High Temperature Gas Reactor": presents the design description of the NGNP with a prismatic core including dimensions and operating conditions.
- "Phenomena Identification and Ranking": uses the NGNP PIRT tables generated by the NRC and recapitulates the "high importance" and "medium/low knowledge" phenomena as those that the HTTF should be scaled for.
- "Depressurized Conduction Cooldown[3] Event Scaling": as stated above, the correct simulation of this event is the primary purpose for the HTTF facility, so the associated scaling is critical. Scaling results are presented for all three stages of this event: depressurization, air-ingress (lock-exchange followed by molecular diffusion), and natural circulation.

  As the depressurization phase is expected to last less than one second and not be accompanied by a large heatup (as opposed to LWR large break scenarios), it is considered unimportant to simulate and this allows for the HTTF to be a relatively low pressure facility (0.4 – 1 MPa) so that costs are not prohibitive. The tail end of the depressurization would be simulated so that the entrance to the air-ingress phase occurs naturally.

  For the air-ingress and natural circulation phases, the usage of prototypic fluids at prototypic temperatures and pressures greatly simplifies the scaling. For the lock-exchange phenomena, the proposed reduced height facility scales well with one exception. Specifically, the Reynolds no. of the air flowing into the outlet (lower) plenum cannot be matched. Basically, the air velocity will be matched due to its dependence on the gas density differences (hot helium vs. cold air), however, the Reynolds no. is dependent upon the outlet pipe diameter, and with the velocity and fluid properties matched, perfect matching of the Reynolds no. would require the outlet pipe to have the same diameter as the actual plant. We can assure that both the model and the plant conditions are both in the fully turbulent flow regime however. Also, consideration will be given to using a larger than scaled outlet pipe diameter to reduce this distortion.

  The latter phase of the air-ingress event is projected to be dominated by molecular diffusion of the air through the hot helium trapped in the upper portions of the reactor vessel. Because of the prototypic fluid conditions, this phase can be very well simulated

[3] A depressurized conduction cooldown (DCC) and a depressurized loss-of-forced-circulation (D-LOFC) are essentially the same event. The DCC terminology is routinely used by the prismatic core community while D-LOFC is used by the pebble-bed.

by the HTTF except that the time scale will be accelerated. The acceleration of the time scale is due to the dependence of the diffusion on the square of the length scale and so "time" in the experimental facility will pass 16 times faster than in the plant for this phase. While seeming to be a large distortion, the main impact will be to shorten the duration of the tests and that turns out to be beneficial (this phase could otherwise last for several days).

Once enough air has diffused through the core into the inlet (upper) plenum so that temperature induced density differences outweigh those associated with concentration differences, the natural circulation phase begins. Flow now enters the outlet plenum from the break and flows upwards through the core to the inlet plenum and out to the reactor cavity. This has a small effect on core heat transfer but determines the flow of oxygen into the reactor vessel which in turn determines the amount and location of the graphite oxidation. Though the HTTF will not simulate the graphite oxidation, it will be capable of providing data for code validation for the timing of the onset of natural circulation and the magnitude of the induced flow. The scaling study indicates that the HTTF can accurately simulate the natural circulation phase of a D-LOFC.

- "Pressurized Conduction Cooldown Event Scaling": a secondary objective for the HTTF would be the simulation of a P-LOFC. In this event, forced circulation is stopped while the system is still pressurized. A natural circulation flow ensues with the helium rising through the core and entering the inlet (upper) plenum as buoyant turbulent jets. The concern is that hot spots might develop leading to subsequent structural failures. Because of the reduced pressure[4], it is not possible to match both the core exit temperatures and the jet Reynolds nos. when helium is used as the coolant and the flow is driven by natural circulation.

  Two suggestions were made to improve scaling for the P-LOFC event. The first is to evaluate the usage of nitrogen as the coolant. The second was to consider using forced flow, albeit in the opposite direction from normal operation, to better match the temperatures and velocities for the buoyant jets exiting into the inlet plenum. OSU will re-examine the scaling for the P-LOFC event considering both of these suggestions.

  In addition to the scaling for the P-LOFC, this presentation also presented the scaling for normal operation. Using helium as the coolant, matching the core temperature rise and Reynolds no. would require that the power be 1/57th of the full-scale plant. The required power would then be greater than 10 MW and this is not practicable. The maximum power available will be about 1 MW, so using helium, the maximum core Reynolds no. that could be simulated would be less than 1/10th that of the prototype.

  Again, the suggestion to consider using nitrogen was made to improve the scaling for normal operating conditions. It appears that with nitrogen as the coolant, the core Reynolds no. could be increased to about ½ of the prototype. OSU will address this suggestion at the second scaling and design review meeting.

- "Core Scaling Analysis": this presentation gave the scaling analyses for both the prismatic and pebble-bed core modules. For both core types, the major design challenge is to properly scale the radial heat transfer during a conduction cooldown

[4] The peak operating pressure for the HTTF will be limited to the range 0.4 – 1 MPa, depending upon the maximum vessel wall temperature, so that the cost of the pressure vessel will not be prohibitive.

event. For the HTTF, the ratio of the surface area to volume will be about 7.5 times larger than for the NGNP. Consequently, if a material with the same thermal conductivity as graphite is used in the HTTF, the radial temperature profile, and hence the peak fuel temperature, would be poorly simulated. For the prismatic core, calculations showed that by using a ceramic with a relatively low thermal conductivity the core radial temperature profile could be well matched.

For the pebble-bed core, the scaling effort is not as far along as for the prismatic design. In the governing equation for radial heat transfer, the terms for radial conduction and thermal radiation were treated separately. OSU was advised to repeat this analysis using an empirical formula for the effective thermal conductivity that treats conduction and radiation together (e.g., the Breitbach-Barthels modification of the Zehner-Schlunder model).

One of the important considerations in designing the pebble-bed core is the bed-wall bypass effect due to the change in the bed porosity near the wall. This wall effect persists for a distance of about 4-5 pebble diameters from the wall. The NGNP design is for an annular core that is only about 15 pebbles wide, so nearly 2/3 of the core will be in this near-wall region. To match this in the HTTF, with the chosen vessel diameter (see below), the pebble diameter will be 3/8 inch resulting in an annular core region that is 13 pebbles wide and has about 550,000 pebbles. Heating each individual pebble is, of course, impractical. So, the scaled decay heat will be provided to the inner reflector and then conducted radially outwards through the core.

- “Reactor Cavity Cooling System Scaling”: this presentation presented the scaling rationale for an air-cooled RCCS. During the meeting, the NGNP program representatives from INL informed us that the design is now being focused on the water-cooled RCCS. Consequently, this scaling analysis will need to be redone.

  Also discussed in the presentation was the design challenge presented by the relatively large surface area to volume ratio for the reactor vessel. With respect to the RCCS performance, the too large vessel surface area would result in a vessel wall temperature that is too low. Consequently, the dominant phenomena would switch from radiation heat transfer to natural convection in the reactor cavity. To prevent this distortion, OSU first proposed to insulate about 2/3 of the vessel exterior surface to force the vessel wall to the correct temperature. However, as noted in the meeting, this would cause asymmetries in the radial heat flow within the vessel. One possible solution is to place a low emissivity clad on the vessel exterior to account for the excessive surface area. OSU is studying the practicality of this proposed solution.

- “Scaling Choices”: this presentation brought together all of the information from the individual scaling efforts to determine the set of similarity criteria that will be used to determine the dimensions and operating conditions for the HTTF. Still, a number of decisions must be made by the analyst to achieve closure in the design process. First, two requirements were imposed, namely that both kinematic similarity and friction and form loss similarity be maintained. To preserve kinematic similarity, the ratio of all of the individual component flow areas to the core flow area for the model must match those of the prototype. The requirement on the friction and form losses will be met by adding orifices as needed to the facility.

  Next, a number of choices must be made that govern the overall facility size. The diameter scale is most critical as it directly affects the necessary core power, the construction costs, and the material costs. A consideration of costs and the commercial availability of components (e.g., the reactor vessel) led to a selection of a quarter height facility with a diameter scale of 1:7.54. This yields a volume scale of 1:227.4.

Previously, a decision had been made that the HTTF will be a full temperature facility (note: this greatly simplified the scaling analyses). Then, based on both economics and safety considerations, the maximum pressure will be limited to 0.4 MPa for a designed outlet temperature of 1000 C. As discussed above, this pressure limit does not affect the D-LOFC event but will impose some scaling compromises for the simulation of both the P-LOFC event and normal operating conditions. Some of these compromises may be able to be remedied by using nitrogen as the coolant to better simulate these high-pressure conditions.

- "HTTF Design Description": based on the scaling analysis and the design choices discussed above, the design and operating conditions for the HTTF were given. The following three tables summarize this information:

HTTF Operating Conditions

Parameter	Value	Units
Coolant	Helium	
Maximum Core Power	600	kW
Maximum Coolant Pressure	0.4	MPa
Mixed Outlet Helium Temperature	1000	C
Inlet Helium Temperature	490	C
Maximum Mass Flow Rate	0.32	Kg/s

HTTF Pressure Vessel Data

Parameter	Value	Units
Inside Diameter	0.99	m
Outside Diameter	1.02	m
Core Barrel Inside Diameter	0.89	m
Core Barrel Outside Diameter	0.92	m
Cylindrical Section Length	3.99	m
Inlet Duct Flow Area	0.15	m^2
Outlet Duct Flow Area	0.11	m^2

HTTF Prismatic Core Data

Parameter	Value	Units
Inner Reflector Equivalent Radius	0.19	m
Inner Reflector Cross Sectional Area	1.76	m^2
Number of Coolant Channels	190	
Coolant Channel Diameter	0.0159	m
Coolant Channel Flow Area	1.99e-4	m^2
Total Coolant Channel Flow Area	0.0378	m^2
Active Core Outer Radius	0.34	m
Active Core Inner Radius	0.19	m
Active Core Height	1.98	m

- "Preliminary Test Plan for the High Temperature Test Facility at Oregon State University": this presentation discussed the types of tests that may be conducted within each of the following three categories: depressurized loss-of-forced circulation (D-LOFC), depressurized loss-of-forced circulation with air-ingress, and pressurized loss-of-forced circulation (P-LOFC). For example, in the D-LOFC category, the break size and location would be varied to include:
    - o Double-ended inlet-outlet duct break,
    - o Control rod drive nozzle break,
    - o Instrumentation tube break,
    - o Inlet duct break, and
    - o Small beaks.

Three comments were made during the discussion of the test plan as discussed below:

- For the pressurized loss-of-forced-circulation test, consider running a P-LOFC with a subsequent failure (e.g., a CRDM), this is a particular concern of the INL NGNP team.
- For the depressurized loss-of-forced-circulation with air-ingress test, consider doing a parametric study where the reactor cavity back-pressure and temperature are varied.
- For the depressurized loss-of-forced-circulation with air-ingress test, there was a discussion about what controls the onset of the natural circulation phase: molecular diffusion or rising thermal plumes from the combustion products resulting from oxidation of the outlet plenum core support structures. The HTTF is not being designed to simulate graphite oxidation. However, there might be a possibility for including sacrificial graphite sleeves on the core supports for a couple of the final tests. OSU will investigate the possibility of performing such a test without compromising the integrity of the facility. If it proves feasible, INL offered to provide the necessary nuclear grade graphite for the test.

This completes the summary of the presentations on the scaling and design of the HTTF integral effects test facility. At the next scaling and design review meeting, OSU was asked to provide more information on the instrumentation plan for the facility. Specifically, OSU is expected to address the following comment from Randy Gauntt of SNL based on their experience in performing severe accident experiments:

> "While thermocouples are available for high temperature applications, such as type C varieties, their use can be very problematic in practice. Typical problems encountered include junction thermal inertia and thermal radiation effects when measuring gas temperatures, perturbations introduced by the measurement device itself caused by thermal conduction down the wires when the wires run through a thermal gradient (run the leads along isotherms near the junction). Maybe the most problematic that we have encountered are errors associated with running the wires through hot regions, especially if the wires run through regions potentially hotter than the junction itself. The error is associated with the insulator dielectric properties at very high temperature where the electrical resistivity decreases, changing the emf coming off the junction."

<u>Pending Actions/Planned Next Steps for NRC:</u>

There are two next steps for the NRC: 1) hold the 2nd HTTF design and scaling review meeting on May 27-28 at the NRC in Rockville, and 2) meet with the DOE/INL NGNP program representatives in a side meeting at the RIC to discuss the possibility of the DOE becoming a partner in the cooperative agreement.