UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 – 0001

September 15, 2009

MEMORANDUM TO: ACRS Members
Subcommittee on Future Plant Designs

FROM: Maitri Banerjee, Senior Staff Engineer, ACRS ***/RA/***
Reactor Safety Branch B
ACRS

SUBJECT: CERTIFICATION OF THE SUBCOMMITTEE ON FUTURE PLANT DESIGNS REGARDING ADVANCED REACTOR RESEARCH PROGRAM ON JANUARY 14-15, 2009

The minutes of the subject meeting were certified on September 14, 2009, as the official record of the proceedings. A copy of the certified minutes is attached.

Attachments: Certification Letter
Minutes

cc w Attachments: ACRS Members

cc w/o Attachments: E. Hackett
C. Santos
S. Duraiswami
A. Dias
N. Mitchell-Funderburk

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MEMORANDUM TO: Maitri Banerjee, Senior Staff Engineer
Reactor Safety Branch – A
ACRS

FROM: Michael Corradini, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE SUBCOMMITTEE ON FUTURE PLANT DESIGNS REGARDING ADVANCED REACTOR RESEARCH PROGRAM ON JANUARY 14-15, 2009

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting held on January 14-15, 2009, are an accurate record of the proceedings for that meeting.

/RA/ 9/11/2009

Michael Corradini, Chairman Date
Future Plant Designs Subcommittee

Certified by: Michael L. Corradini
Certified: September 11, 2009

Issued: September 15, 2009

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON FUTURE PLANT DESIGN REGARDING ADVANCED REACTOR RESEARCH PROGRAM
ON JANUARY 14-15, 2009, IN ROCKVILLE, MARYLAND

On January 14-15, 2009, the ACRS Subcommittee on Future Plant Design held a meeting in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to receive a briefing from the NRC staff regarding the advanced reactor research program (ARRP) developed by the NRC Office of Nuclear Regulatory Research (RES). The ARRP was updated to incorporate the needed NRC research for review of the next generation nuclear plant (NGNP) application scheduled to be submitted in 2013 and to support the pre-application review. The meeting was convened at 8:30 a.m. on January 14 and adjourned around 4:15 p.m. on January 15. The meeting was open to the public.

Attendees:

ACRS Members	**NRC Staff Presenters**	**NRC Staff**
Michael Corradini (Chairman)	Jocelyn Mitchell, RES	Michael B. Rubin, RES
Said Abdel-Khalik	Paul Rebstock, RES	Appajosula Rao, RES
William Shack	Mourad Aissa, RES	Kimberly Tene, RES
Harold Ray	Mary Drouin, RES	Nathen Siu, RES
J. Sam Armijo	Kevin Coyne, RES	Jeffery Wood, RES
Dennis Bley	Imtiaz Madni, RES	Sean Peters, RES
ACRS Consultant	**NRC Staff**	Rob Tregonnig, RES
Thomas Kress	Stuart Richards, RES	Jose Pires, RES
ACRS Staff	Kathy Halvey Gibson, RES	Paulette Torres, NRR
Maitri Banerjee (DFO)	John Jolicoeur, RES	Ganesh Cheruvenki, NRR
NRC Staff Presenters	John Monninger, RES	**DOE/INL Staff**
Stuart Rubin, RES	Tim Lupold, RES	James Kinsey, INL
Joseph Kelly, RES	Richard Lee, RES	Mark Holbrook, INL
Anthony Ulses, RES	Syed Ali, RES	George Hommer, INL
Stephen Bajorek, RES	Lauren Gibson, RES	**NRC Consultants**
Allan Notafrancesco, RES	Tarek Zaki, RES	Cecil Parks, ORNL
Sudhamay Basu, RES	Thomas Kenyon, NRO	**Pubic**
Jay Persensky, RES	Don Carlson, NRO	Alan Levin, AREVA
Amy Hull, RES	Kamiar Jamali, NRO	Takushi Kiba, Secretariat of NSC (Japan)
Shah Malik, RES	David DeSaulniers, NRO	Yuzu Fujii, JANUS
M. Srinivasan, RES	Jack Donohew, NRO	Katsuo Suzuki, JNES
Herman Graves, RES	Michael Case, RES	Toshio Morimoto, Japan NUS

The presentation slides and handouts used during the meeting are attached to the Office Copy of the meeting transcript. The presentation to the Subcommittee is summarized below.

Opening Statement

Chairman Corradini convened the meeting by introducing the ACRS members and the consultant present. He mentioned the 2008 HTGR PIRTs and the NGNP licensing strategy report to be the source documents for the ARRP revision. The licensing strategy document was submitted to the Congress in August 2008. Dr. Corradini explained that due to his and Dr. Powers' participation in certain PIRT panels, he and Dr. Powers (not present) would not involve themselves in discussions regarding those particular panels. He then called upon the RES Program Manager, Stuart Rubin, to begin the staff presentation.

Introduction

Dr. Rubin started by providing a history of the ARRP development for HTGRs. He noted that the first version of this plan, issued back in 2003, was revised to incorporate the Energy Policy Act of 2005 that mandated development of the NGNP. The ARRP addresses the development of the technical infrastructure needed for staff review of the NGNP application and to a much limited extent a sodium fast reactors application. The ARRP identifies the gaps in NRC infrastructure in terms of data, information and modeling know-how, and other R&D needs required to review the future NGNP application. As the Department of Energy (DOE) has not selected a design for the NGNP yet, the ARRP addresses two types of HTGR reactors, the prismatic block reactor (PMR) and the pebble bed reactor (PBR). The members were interested to understand the division of responsibility between the NRC and the applicant's research work. Mr. Rubin pointed out that if an issue (e.g., metallic radionuclides in graphite dust) has important implications for the safety analysis technical basis, then the applicant would be expected to develop the data and the modeling to account for any new issues.

Dr. Rubin then discussed the design and safety approaches that are different from the past Fort St. Vrain reactor. The technical areas that contribute to the development of accident analysis evaluation models, staff priorities for model development, targeted events and figures of merit for such events were mentioned. The staff pointed out that, due to a lack of information at this time, the hydrogen plant is considered in the accident evaluation model as a small thermal load on the reactor. Also, due to the lack of a selected design, the staff can not model events like an intermediate heat exchanger (IHX) failure. Upon member Abdel-Khalik's question, the staff stated that it is important to know about the core inlet flow distribution not only for local effects but also to determine accident response and the impact on balance-of-plant due to non-uniform core exit temperature distribution. The staff is doing some CFD analysis to characterize the core outlet distribution. Chairman Corradini's question on dimensional changes of graphite with time in a high temperature environment and need for inservice inspection was deferred to a later presentation on graphite. In closing, Dr. Rubin pointed out the focus of NRC research is on the development of capabilities needed to perform the technical review of the NGNP application, with leveraging of information, where possible, consistent with the NRC role. The staff is collaborating with international research communities (e.g., European Union RAPHAELE program, Japanese atomic energy agency, OECD TAREF program).

Evaluation Model Development

Dr. Joe Kelly from RES provided an overview of the staff's evaluation model (EM) development plan. The scope of reactor plant systems analysis model development includes four areas of analysis: nuclear; thermal-fluids; fuel performance; and fission product transport. Each area would address normal operation, initial fission product (FP) release, and delayed FP release. The key elements the EM needs to address are: (1) determination of the source term with consideration of circulating activity and activation products; (2) modeling the initial release with consideration of plate-out, lift-off, and events that result in coated fuel particle (CFP) failure; and (3) delayed release of FPs from intact and failed CFPs with or without air or steam ingress and holdup in helium pressure boundary/ confinement.

The graphite dust source term model is the subject of future work. Upon member questions, Dr. Rubin noted that such modeling would involve detailed understanding of (1) fission product transport within the particles and the matrix to determine how much FP (mainly cesium) is available to be bound up in the dust (dependent on diffusion coefficient of cesium through silicon carbide layer at the temperatures and burnup condition); (2) how much dust containing the cesium is actually generated; (3) dust transport and settlement (CFD analysis); and (4) what happens to that dust in an event where it can be blown out of the system into the surrounding building. The amount of radioactivity in the dust would determine the design of mitigation components (e.g., containment/confinement with vent filters). The staff deferred member Bley's question on dust explosion and fouling of heat transfer surfaces by dust to a later presentation on graphite research. Upon Member Abdel-Khalik's question on maximum core heat generation rate, Dr. Rubin pointed out that the need to keep the core temperature within a certain design value (around 1600 degrees C) using only passive cooling in an accident dictates that power density in the core be low. Annular core design allows radial transport of heat in support of passive cooling.

Dr. Kelly provided some examples of transients to be analyzed, and discussed the limiting phenomena, parameters and related code capabilities. Member Ray's question on reactivity impact of seismic events for PMRs was deferred to the structural presentation scheduled for the following day. Dr. Kelly presented the individual components of the evaluation model (for both steady state and transient analysis) by function and by the usage of codes for each function. Some codes need to be modified for HTGR application. Treatment of uncertainties like the bypass flow and the radial and azimuthal variations of core flow distribution affecting the core temperature and power distribution were discussed. The staff pointed out that consideration of hot channel factors may be one way of modeling this type of potentially significant uncertainties given that core flow distributions may not be known. The staff envisions doing integral effects experiments to help model flow distributions. It was noted that many uncertainties diminish in value as core temperature decreases.

The steps of EM development work includes code development followed by code integration, uncertainty analysis with incorporation of model bias and uncertainty factors into the codes, code validation for the PIRT identified high ranked phenomenon using available separate effects test data, and a code adequacy report. Dr. Kelly discussed the capabilities and current limitations of the codes included in the EM, and provided a schedule for the EM development work with code validation that need to be completed by the end of 2013 in preparation of the COLA review. Regarding the role of CFD analysis, Dr. Kelly discussed the potential applications. The staff is planning to use CFD as a tool to better understand and account for local phenomena. Dr. Kelly provided some examples of ongoing studies of CFD applications.

Fuel Analysis

Dr. Rubin defined the objectives of the fuels analysis program and the key fuel safety and licensing issues. The staff's objective is to develop a model to predict FP release at various temperature and burnup conditions. To develop models for fuel behavior, FP transport and accident analysis, the staff needs design and manufacture specific fuel irradiation and accident condition test data. The staff plans to leverage the fuel R&D that DOE/Idaho National Laboratory (INL) is conducting to support licensing, and obtain finite element PARFUME code from DOE. In NGNP design, CFPs would serve as miniature containment vessels during normal operation and accidents which is a departure from the current LWR designs. The members were interested to understand how various limitations and ongoing test results could impact the Commission's policy decision on the need for a containment structure. This policy decision has not been made yet. It appears that DOE believes that use of TRISO fuel with its high performance characteristics is going to allow implementation of the vented confinement concept (vs. leak-tight containment). DOE has set design limits on the CFP manufacturing defect rate and normal and accident condition failure rates for meeting the dose criteria at the site boundary.

The staff approach for modeling individual fuel particle performance involves the use of the PARFUME computer code which mechanistically models fuel phenomena/behavior. The staff approach for modeling core-wide fuel fission product release may involve the use of TMAP-4 and/or PARFUME. Also included are sensitivity studies to understand how changes in parameters like temperature and burnup affect fuel behavior, and an empirical failure probability model that will be developed directly from fuel qualification test results.

DOE's fuel test must ensure that the test samples and conditions are representative and bounding of the future production fuel and the HTGR core. The staff believes that ATR test will very closely match the projected limiting locations in the VHTR core. Regarding the question of conservativeness of DOE's accelerated fuel qualification tests, the staff discussed how test acceleration could impact the simulation of various failure mechanisms, DOE's method for determining the sensitivity of accelerated tests, and need for limiting the acceleration. The NRC MOU with DOE provides for a timely regulatory review of the DOE fuel test program.

The staff discussed the failure mechanisms for the TRISO fuel particles associated with normal operation and accident conditions including fission product diffusion (mainly cesium) through intact coating layers, which in the case of metallic radionuclides contributes to the graphite dust source term. Regarding particle property related to important phenomenon during normal operations and accident conditions, and given the statistical distribution of these properties, the question of how to assure quality of as-loaded fuel needs to be addressed. DOE is using the PARFUME code to address acceptable tolerances to fuel properties. All sources of radionuclides, including the ones from failed particles and the metallic elements (like Ag110m) that diffuse through intact particles ending up in graphite dust, need to be accounted for. The staff plans to obtain the PARFUME code, and do sensitivity studies to better understand the tail ends of the statistical distribution of properties and their implications on fuel performance. Additional research will be needed in this area to establish quality requirements and set values for technical specification (TS) limits.

Dr. Rubin then discussed the staff's plan for core-wide particle failure rate model development. The CFP failure fraction versus temperature and burnup during normal operation and accident conditions will be established from the test data. The staff will use PARFUME code to develop conservative and best estimate CFP failure fraction values for normal operation, design basis and beyond design basis accidents. The staff will need Commission decision regarding when to use conservative vs. best estimate values in the EM. Given the lack of actual fuel qualification test data for NGNP at this time, the staff is using the German test results. Upon Chairman Corradini's question, Dr. Rubin explained the fuel and set up differences between the current staff approach and that of Fort St. Vrain.

The staff plans to use the available effective diffusion coefficient vs. temperature graphs developed using German, US, Russian and Japanese data for the CFP layers and the fuel matrix, and use the TMAP4 code to model FP transport for contamination, intact and failed fuel particles. This code, obtained from INL, solves the 1D diffusion equation and calculates diffusion rates for various species of radionuclides. Dr. Rubin discussed the various features of the code. Fuel temperature is the most important parameter, and local values must be known to determine FP release. The staff plans to do sensitivity studies to evaluate the impact of variations in propertied like diffusivities, etc. In the long term, the staff plans to get the data from DOE and change the diffusion coefficients to represent the NGNP fuel. This will enable the staff to develop FP transport knowledge to review NGNP application. The staff is also considering if the diffusion and release models can be simplified by developing an effective diffusion model code to generate one diffusion coefficient for all the fuel layers. Upon member Armijo's question, the staff explained their process for integrating the TMAP4 capabilities in their code suite.

Dr. Rubin then discussed the staff's plan on FP release for three kinds of events: water ingress; air ingress; and reactivity events. The principle effect of water ingress is the mobilization of fission products out of failed particles resulting in an increased release-to-birth ratio. Until DOE does some testing with the NGNP fuel, the staff plans to use the available data to run the MELCOR code to account for these phenomena. Addition of a steam generator in the NGNP design will make water ingress a design basis accident.

For air ingress accidents, the particle failure mechanism is likely oxidization of the outer pyrolytic carbon layer and the silicon carbide layer, thus releasing FP by means other than diffusion. DOE may include air ingress in their AGR program as the existing data and models are not typical of NGNP. Reactivity accidents that depend on core excess reactivity are less significant for PBRs because the continuous online refueling limits the amount of excess reactivity. Due to the higher reactivity worth of an inserted control rod for a prismatic reactor, the assumed concurrent helium pressure boundary failure with control rod ejection would provide a motive force for FP transport. For HTGRs rod ejection was considered as one of the limiting events, which however may change due to the risk informed approach for licensing the NGNP. Limited test data and models exist. In the near term, the staff plans to use the available data and model.

Given that integrity and fission product retention are key to HTGR safety, manufacturing of high grade fuel with minimal lot-to-lot variation is important. The staff is developing a fuel manufacture quality assurance oversight strategy and an NRC inspection program for NGNP fuel fabrication facilities.

Nuclear Analysis

Mr. Anthony Ulses of RES started the presentation by explaining that the staff was at the starting point on this subject although they had done some very basic assessments while reviewing the PBMR submittals. The staff's objective is to have the ability to assess the safety analysis methods of the NGNP applicant with a fully independent set of NRC methods. In order to do so, the staff plans to leverage the existing SCALE code suite, developed over the past five or six years, consisting of extremely accurate methods for processing cross section library data for lattice physics calculation with high fidelity. The staff added the ability into SCALE to handle the pebble or the prismatic block fuel and double heterogenic systems. The Oak Ridge National Laboratory (ORNL) has done a HTR-10 validation model development for the staff to support the pebble bed model development. This has been accepted for the International Reactor Physics Evaluation Handbook after extensive vetting. A similar HTTR validation model is in the process of development to provide a validation model for the PMR core.

The staff is establishing very detailed models of pebble and prismatic systems to allow for exploration of sensitivities and uncertainties and to look at the linkage between the detailed calculations and the PARCS-type analysis. Staff plans to use GenPMAXS as a code processor to fit SCALE output into PARCS. The staff expects to put a considerable amount of research within the PARCS code to translate the nodal diffusion solver to recreate the relevant reaction rates. Mr. Ulses showed the PBMR-400 benchmark comparison results of PARCS with four other code calculations for slow control bank withdrawal indicating consistent performance.

The staff research plan for flux and power profile calculation includes four steps. To better understand the uncertainties in the system behavior, the staff will use the TSUNAMI method sequence in SCALE. Multi-tiered approach will start with small scale studies of kernels and pebbles, using available data from HTTR, HTR-10, etc. to develop an understanding of the system, and then to develop a very detailed model for NGNP to ensure the needed tool set for licensing review is achieved. The other two steps include preparing PARCS interface and preparing for FP release calculations. Mr. Ulses explained why he expects the PBR calculations to be more complex than PMR, and discussed some common challenges. The current inability to instrument a pebble and unavailability of kernel/pebble power measurements are making the validation of predictions difficult. A long discussion between the members and Mr. Ulses ensued as to what it meant to have or not to have the ability to measure in-core or in-pebble parameters, and Mr. Ulses noted that this was an area where the staff would continue to engage the INL.

The staff's research plan for decay heat prediction uses the ORIGEN code within SCALE with some calorimetric data needed for validation. Analytical prediction of xenon instability has to be confirmed via startup tests. In the area of reactivity coefficients, some recent research work in Germany (Dagan) suggested that the treatment of neutron scattering resonances may be nonconservative. The staff is planning to modify the continuous energy dimensional transport code CENTRM to assess the impact. The staff may need high temperature data if it turns out that the fuel temperature coefficient is nonconservative as suggested by Dagan.

The staff plans to take advantage of the large amount of currently available international data, and engaging with INL to establish the need and the sources. The staff sees currently operating facilities like the HTTR, HTR-10, ASTRA and certain historic and prototype facilities as sources of data to validate parameters related to criticality, power distribution, reactivity, decay heat and source term.

The members were interested to know how the staff is planning to handle crosscutting issues like core flow bypass that affects core neutronics, fuel performance, materials. The staff is holding periodic working group meetings to flush out those issues and make sure everybody is "playing from the same sheet of music," and setting up communication channels to talk to DOE "peer-to-peer." The staff has signed an MOU with DOE for the cooperative work between the two agencies and is completing an interagency agreement. This is in line with the Energy Policy Act of 2005 that says DOE shall engage with the NRC to get the NRC input so that the DOE research is responsive to the safety requirements for the NGNP.

Neutron scattering in graphite is another area where some recent work has indicated concerns about the adequacy of models. The staff continues to follow such developments. In order to circumvent the lack of data on actually depleted fuel pebbles, among others, the staff is developing a standard problem for pebble burnup for presentation to OECD in February. The staff expects this will give them considerable information to help develop the models. Identifying the data needs and using the data to validate the codes is an area of near and long term focus of staff actions. Although the HTR-10 core configuration is different from the PBR, the staff expects to use the data in assessing the codes. Other longer term work involves enhancing the execution speed of SCALE code suit, and completing the SCALE to PARCS interface. They want to make sure that no information is lost while going from a detailed method to the nodal diffusion theory methods. In summary, Mr. Ulses pointed out that the challenges remain in: validating methods to be able to predict reactivity of the system; handling the stochastic nature of burnup, and the ability to homogenize and then be able to recreate that information to a sufficient level of detail to do the analysis; handling multilayered heterogeneity; reactivity effects of moisture ingress; and reliable prediction of fuel isotopics. The staff is starting with small scale studies and plans to scale them up.

<u>Thermal-Fluids Research</u>

After a lunch break, the Subcommittee convened with Dr. Stephen Bajorek of the staff presenting the research plan for developing thermal-fluids evaluation models. Dr. Bajorek discussed the PIRT identified significant thermal-fluids phenomena with low staff knowledge level. Four such issues involving core and vessel thermal-fluids, air ingress, RCCS performance, and graphite dust were discussed in terms of how staff sees the problem and their general approach for resolving them.

Dr. Bajorek outlined the staff approach for evaluation model development, major thermal-fluids issues for gas reactors and experimental data needed to benchmark the NRC codes and various parts of the evaluation model. Although thermal-fluids processes are fairly well understood, some of them are being driven into ranges of conditions in NGNP outside the ranges where the correlations were developed, thus resulting in larger uncertainties. Limited convective heat transfer data are available at flow rates and temperatures of PBR and PMR with helium as coolant. Limited data are available to validate core effective thermal conductivity for PBR. Also, properties of helium and the gas mixtures at high temperatures, and uncertainties associated with measurement of maximum fuel temperature and in evaluating bypass flow could be significant additional problems.

To support NRC evaluation model development by 2013, there are three different steps staff is taking to obtain or generate integral and separate effects test data. This involves interfacing with the DOE, collaborating and entering into agreement with international organizations, and conducting independent experiments. Staff effort includes obtaining data from HTR-10, HTTR, IAEA cooperative research program, and working with the CSNI Task on Advanced Reactor Experimental facilities (TAREF) international group. The third step of conducting independent experiments, incase enough data is not generated from the DOE research, could be through the Thermal-Hydraulic Institute or small-scale experiments conducted by universities.

The staff will compare currently available data and the applicant's new data to existing models (e.g., core convective heat transfer and bypass) to identify uncertainties, and develop new evaluation models as necessary. Review of existing correlations for core effective thermal conductivity and use of CFD to examine sensitivities were in progress. Also in progress were improvements to MELCOR to incorporate gas mixture properties generic to both PBR and PMR. As a lot of the thermal-hydraulics test facilities that did LWR integral experiments used electrical simulators for fuel, members asked how staff would simulate PBR fuel. With a discussion of a few possibilities, the staff acknowledged that it was a difficult issue and needed more work.

The staff has initiated research to develop models for the core, vessel and containment thermal and fluid flow such that parameters like pressure and temperature that will affect FP release during postulated accident conditions can be predicted. The staff will need data to support the evaluation model development and is using CFD to help guide decisions and determine needed test programs. In addition to data from separate effects tests, the staff considers a well-scaled integral effects test as vital in order to investigate multiple system failures and the safety system performance. To help achieve this, the staff has compiled a survey of gas cooled reactor facilities, and is participating in CSNI TAREF.

Thermal properties, like emissivity and pebble bed porosity in addition to coolant bypass flow need to be accurately modeled into the code to predict fuel temperature. Upon member's question, Dr. Bajorek discussed how the three modes of heat transfer, conduction, convection and radiation, compete with each other under different scenarios. The members were interested to know how variability of reactor parameters, for example, porosity of the pebble bed, is being accounted for. The staff has examined the effect of porosity on heat transfer and pressure drop. Large database and correlations in use at chemical process industry that uses packed beds were mentioned. The staff is using CFD to model local areas like regions of the porous bed near the wall to ensure properties like drag coefficients are appropriately accounted for.

For modeling of air ingress phenomena, recent studies show that duct exchange flow or lock exchange flow can get air into the core much rapidly (compared to molecular diffusion), in several minutes as opposed to several hours, as helium escapes through the break. This may be a difficult process to calculate, also the needed data are lacking. Graphite oxidation needs to be modeled into MELCOR (staff's main code for accident modeling). The staff plans to conduct tests to assist model development. The staff plans to develop a small multi-purpose integral effects test rig for air ingress among others, and design a test apparatus for a separate effects test to provide data for air ingress flow rates for various geometries to help in model development. The staff discussed the results of the simulation of a break at the vessel top resulting in air ingress fairly early in the accident such that oxidation of the fuel in that region could be a concern. The staff's integral multi-purpose test loop will look at particulate transport also.

Regarding RCCS performance, following issues could impact the modeling of thermal response: factors affecting emissivity of RCCS (graphite dust) and gray gas effect on the participating media; RCCS failure assumptions (failure of one or both channels); and internal heat transfer inside the RCCS tubing and associated uncertainties. Lack of data is one of the major concerns. The staff plans to participate at the ANL Natural Convection Shutdown Heat Removal Test Facility (NSTF) to investigate RCCS performance, and review international test data or sponsor independent tests, as needed. Chairman Corradini asked if sharing of experiments would impact the staff's independent confirmatory analysis capability. The staff pointed out that they would collaborate with the DOE on setting up the test and will attend the tests to ensure they meet NRC standards. Although, the staff will use the jointly shared test data, the staff will perform independent calculations using staff's own correlations or computer code suites to derive judgment about safety. The staff has done some preliminary CFD calculations to help understand how to model this gray gas in the reactor cavity.

Graphite dust is considered to be an issue primarily for PBR under depressurized loss-of-forced circulation scenarios. Hydrodynamic conditions for dust suspension, liftoff and carryover need to be modeled for FP release and cavity filter performance. Dust properties, like the generation rate, FP content, size and shape distribution of graphite particles also need to be defined. The staff has completed a literature survey and may need to conduct separate effect tests to develop models for MELCOR on dust particle generation and transfer. The members questioned and a discussion ensued on historical data regarding graphite dust generation at different gas cooled reactors, including dust detonation and combustion.

<u>Accident Analysis</u>

Mr. Allen Notafrancesco of RES staff presented their plan for accident analysis, the objective of which is to develop the accident source term and FP transport models to support the review of the NGNP license application. The TRISO fuel, nuclear and thermal-fluids performance models developed separately will need to be integrated into accident analysis (MELCOR) models thus requiring extensive coordination with these technical areas. Mr. Notafrancesco discussed the current capabilities of the code and the activities initiated to achieve needed improvements. Initial activities include updating MELCOR to incorporate graphite oxidation models and diffusion of air in helium (NIST data is used to confirm ideal gas law modeling in MELCOR for helium-air properties). Both PBR and PMR core models are implemented into MELCOR 2.1. The staff's accident analysis evaluation model development strategy is to provide the capability to perform analyses for pressurized loss-of-forced circulation, depressurized loss-of-forced circulation with air ingress and water/steam ingress from the secondary system, and then benchmarking the code against relevant plant data.

RES staff plans to incorporate models and methods into the MELCOR framework for core-wide fuel FP release and transport modeling, and will use PARFUME insights for CFP failure rates. Needed improvements in MELCOR include adding an RCCS model, stratified flow air ingress model, modeling of the balance-of-plant components, FP release model, and building re-entrainment model. Mr. Notafrancesco completed his presentation by noting that MELCOR-2.1 modeling extension was well underway and assessment activities would follow.

Hydrogen and Process Plant Analysis

Dr. Sud Basu presented the hydrogen and process plant R&D plan. The objective is to develop independent capabilities to support the staff review of safety implications posed by the hydrogen production facility. Another objective is to develop confirmatory safety analysis tools that maintain adequate safety margin but are not overly conservative. The safety issues considered by the staff include: (1) chemical releases including ground hugging heavy gas, hydrogen detonation, and combustion of flammable gas or liquid; (2) transients in the chemical plant that lead to reactor trip or component failure and IHX/PHX tube failures and other piping failures; and (3) events at the VHTR that effect hydrogen plant, like tritium transport. The EM development work has just began that will predict response of the VHTR to transients in the hydrogen plant, and vice versa. Dr. Basu provided an overview of the needed capabilities. The staff will use existing tools as much as possible and assess the EM against historical data. The staff will establish a tritium activity limit in the intermediate coolant loop measurable through a radiation detector installed in the gas volume. ACRS members noted that the staff's plan seems reasonable, but common-mode initiators like an earthquake which could affect both the HTGR and the process plant needs to be considered.

Human Performance

Drs. Jay Persensky and Valerie Burns presented the staff's R&D plan for advanced reactor human performance. A paradigm shift in the new concepts of operation, compared to current LWRs, need to be considered in establishing the new methods and tools. This paradigm shift embraces advanced control room designs with new approaches to human-system interface and use of digital I&C, and impacts operator's role and activities in the control room and to a lesser extent in the rest of the plant. The staff's objective is to issue new review guidance for highly-integrated control room human factors for NGNP. The staff is participating in OECD Halden reactor project and the NEA/CSNI working group on human and organizational factors. Dr. Persensky discussed the planned R&D areas and schedule for starting each project. The results of the R&D are expected to identify the needed safety enhancements and regulatory actions, and provide the technical basis for design acceptability. The ACRS members noted that the staff's plans appear to be preliminary and need to be integrated into the NGNP design review.

Metallic Component Analysis

The second day of the Subcommittee meeting started with a presentation on the staff's R&D plan for metallic components by Drs. Amy Hull and Shah Malik. Dr. Hull presented the R&D objectives, key safety and licensing issues and historic background. The PIRT results identified 16 phenomena ranking high in importance with low staff knowledge level. Dr. Hull noted that the absence of an NGNP design selection decision (from DOE) regarding the type of reactor and its parameters, mainly the temperature, is causing uncertainty in her work planning. She stated that the staff's work, including some confirmatory research, will be supplementary and complementary, and not duplicative of what is being done elsewhere. In addition to providing sufficient technical basis for regulatory decisions, conducting research to understand aging related degradation mechanisms in HTGR environment will address some of the key safety/ licensing issues and NDE/ in-service inspection needs, and help quantify material performance and its variability. Dr. Hull mentioned a 2008 ORNL paper on material issues for advanced reactors. Also, research was done at the ANL in the early 2002-2004 under an NRC contract on codes and standards for metallic components in HTGRs (NUREG/CR-6816). This work addressed effects of the HTGR environment (NUREG/CR-6824) on degradation of metallic components. Confirmatory testing (creep test program) was also conducted. The staff wants to continue this work.

The staff is involved in ASME and ASTM Code Committees. This involvement informs their activities related to metallic component analysis. Between the two of them, Drs. Hull and Malik represent NRC on the ASME Boiler & Pressure Vessel (BPV) subgroup on elevated temperature designs and the special working group on HTGRs (BPV XI); the working group on liquid metal reactors (BPV III); and the Generation IV Reactor Materials Projects. Particular focus is placed on BPV Code Section III Subsection NH, "Components in Elevated Temperature Service" as well as Section XI for evaluating in-service inspection needs and a risk-informed in-service inspection for HTGR application. Work is being done on HTGRs both in ANS through Standards 53.1 which has more of a systems approach, and at ASME that is more of a component approach mainly towards the PBMR. The staff is supporting work on the ASME BPV code roadmap development that would link these two approaches.

Regarding the development of design and fabrication codes and design standards, the staff is working with the DOE ASME Gen IV Materials Project where the EC, Japan and Korea are also active participants. Regarding the development of in-service inspection requirements, there is a desire to extend the period in between inspections because of longer refueling cycles in most HTGR designs.

The staff is reviewing the NGNP PIRT results to further prioritize research at the metallic component level. There are 16 phenomena that ranked high in importance with low knowledge level which break down in five areas. These five areas represent the following: crack initiation and subcritical crack growth; compromise of surface emissivity; inspection/NDE; design methods and material property control during fabrication and manufacturing; and irradiation induced creep. Dr. Hull described each area and the resulting key R&D issues for safety and licensing.

The staff is updating the creep-fatigue design rules for high temperature use. The staff is also assessing the degradation phenomena, impact of corrosion mechanisms, and emissivity requirements and retention for the life of the RPV and the core barrel material.

The staff presented a list of the ongoing R&D in the area of surface emissivity which focuses on the RPV, core barrel and the RCCS. Member Apostolakis wanted to know if the staff intends to develop any probability distribution of experimental results. The staff was not sure if that is being done, but agreed that an uncertainty analysis would be important to have. The members asked questions on design aspects of metallic component surface emissivity, and it became apparent that not knowing the actual design or materials for NGNP is adding to staff's work scope in many areas.

In the metals R&D area of Gen IV/NGNP materials project, developed by the DOE, Dr. Hull mentioned 12 projects that were undertaken, with 6 being complete. On one of these tasks, collecting creep-fatigue data for Grade 91 steel and Hastelloy XR, an enormous database of material parameters and degradation values have been compiled through an international effort. Upon members' questions, the staff stated that their current objective was to be able to get their test systems "up and running" for good results such that once DOE specified the design/materials, more research can be done in earnest on the actual materials that would be used.

Dr. Malik presented the work that staff is doing on modeling of creep and creep-fatigue crack growth processes which account for subcritical crack growth at high temperatures and is a high importance and low knowledge area according to the PIRT. Crack growth in the RPV and IHX could develop pathways for FP release. The staff has contracted ORNL to develop an independent capability and expertise to understand the phenomena and develop a technical basis for the regulatory framework. Dr. Malik discussed the staff's current scope of work. The members noted again that the lack of design and material selection by the DOE was limiting staff work in this area. Upon Chairman Corradini's question regarding reactor exit temperature, Dr. Kinsey from INL stated that the DOE plans to go out with an offer of financial assistance in the near future to establish a public-private partnership to move the project forward. It was noted that DOE would not specify the temperature at this point because they wanted that to play out through the responses to the request for assistance offer, although they expect it to be in the range of 750 to 800.

Dr. Malik presented a slide that addressed the key aspects of creep and creep-fatigue crack growth processes. The creep behavior is dependent on material and will impact fracture mechanics calculations. Dr. Malik presented typical creep behavior curves from literature and stated that the modeling of material response is much more complicated once one has to account for both creep and fatigue as a cyclic loading. Dr. Malik discussed the crack growth mechanisms with Alloy 800H representations and additional considerations including environmental effects. The issues of transferability of correlation developed from fracture specimen to structures, extrapolation of accelerated test data, and additional degradation mechanisms need to be addressed before any crack growth correlation can be applied with confidence. The staff plans to develop flaw evaluation procedures similar to Section XI from these correlations and modular probabilistic computer codes for independent assessment of licensee submittals.

Dr. Malik summarized the strategy for staff's metal R&D to be as follows: maintain staff awareness and expertise by participating in Code committees, technical meetings and international programs; continue existing R&D programs for phenomena of high PIRT ranking; further refine PIRT prioritization in this area by monitoring international R&D and determine need for confirmatory testing; and perform scoping studies on the NDE and ISI technology.

In his remarks, member Armijo noted that a lack of design/material selection and identification of environmental conditions is making the staff's job very difficult and adding to the expense of the program. Dr. Armijo noted that the staff's plan seemed to include a vast amount of experiments (to develop an underlying database) given the time available. Upon member Abdel-Khalik's inquiry regarding data on the material for radiation effects, the staff stated that fusion work is also producing lots of data related to radiation damage.

Research Plan for Graphite Materials

After a short break, the meeting reconvened with Dr. Srinivasan of RES presenting the graphite research plan. The leading objectives of the staff program are data and information acquisition for safety/licensing decisions on HTGRs and confirmatory analysis of the applicants' data using independent analysis tools. NRC research is expected to generate technical bases for developing regulatory positions on the requirements for the structural and functional abilities of graphite, core and core support components. These technical bases will lead to the development of regulatory positions on: (1) inspections needed to ensure the existence of adequate structural and functional factor of safety during normal operations and anticipated operational occurrences; and (2) for input into accident analysis calculation tools. The design should account for potential air, water, or steam ingress into the pressure boundary and changes to core geometry. Graphite research is intended to generate technical data and information to identify and quantify degradation processes by analytical models, and provide information and data for HTGR accident analysis evaluation models and for evaluating PRAs. Graphite that was used years ago in Fort St. Vrain and Peach Bottom reactors is no longer available and technical expertise for evaluating the new graphite in NGNP VHTR needs to be established.

The lack of consensus codes and standards for graphite components is a significant technical issue. Therefore, the staff has been participating in national and international codes and standards activities over last several years. During 2002-2003 the staff contracted ORNL to organize a working group under ASME to develop graphite codes and standards and to coordinate the ASTM materials subcommittee to develop graphite material specifications and properties test standards for HTGR applications.

Dr. Srinivasan discussed various technical considerations for graphite code development. Nonlinear and non-uniform irradiation-induced dimensional changes of graphite during normal operation, in addition to service stresses, chemical reactions and other fracture mechanisms, need to be accounted for in estimating core component stresses using nonlinear finite element models. Dr. Srinivasan stated that the challenge was to correlate the effects of certain characteristic issues with significant properties of graphite. These include: graphite constituents and microstructure which depend on graphite manufacturing methods; and reactor design and operation, which provide fluctuations in the field variables such as dose and temperature distribution within the individual graphite blocks, which make up the core.

The majority of members of the ASME Division III Subgroup on graphite core components are experienced technical experts from European Union nations, South Africa, Japan, and Korea. Dr. Srinivasan discussed the articles that had been drafted in a preliminary form including Article 3000 on design which was most extensive. Additional work will incorporate data on several grades of graphite being irradiated at many parts of the world. To aid in the development of some of these articles and to provide technical bases for various codes, the subgroup is also developing many mandatory appendices.

As a result of the ASTM efforts during the last five years, two material specification standards are currently available for nuclear graphite. One ASTM specification is concerned with graphite components subjected to high doses, such as moderators and reflectors in HTGR; the other provides material specification for components subjected to low neutron dose, for example, graphite core supports. Dr. Srinivasan discussed various aspects of these specifications including: purity, chemical composition; physical, thermal and mechanical properties; and degree of anisotropy. Because of insufficient data or knowledge, these specifications do not contain any information on irradiated properties.

Upon Member Armijo's question, a long discussion ensued regarding the nature of inherent anisotropy in graphite and how it is minimized. The ASTM standards require the measured ratio of the coefficient of thermal expansion (CTE) in two orthogonal directions to be in the range of 10 to 15 percent, as an acceptable value for nuclear applications. Dr. Srinivasan mentioned that directional ultrasonic velocity measurements are used as quality control of the bulk graphite manufacturing process and a discussion ensued regarding the use of two different techniques, CTE vs. ultrasonic.

Upon Member Abdel-Khalik's questions, Dr. Srinivasan discussed how thermal and irradiation effects are compensated for in the Japanese and AGR reactor designs given the flux and temperature distribution within graphite blocks in a reactor. The Japanese (IG-110) used the more expensive approach of using very isotropic and homogeneous graphite and optimized the design density throughout the core to compensate for irradiation and thermal gradients such that the resulting cumulative effect is nearly uniform. In the AGR design, stress variations and their effect on the overall dimensional changes throughout the core are predicted and then measured over a length of time. This is to ensure that the reactor is operated with adequate clearance, required by technical specifications, for the unimpeded movement of fuel stringers and control rods. In the AGR they also cut out samples periodically and measure properties (e.g. mechanical, thermal). A long discussion took place regarding the graphite property changes and the end of life determination. Member Ray reinforced the need for and challenges in developing ISI/NDT requirements for the core support structures.

Dr. Srinivasan then described the results of NRC-DOE's 2007 graphite PIRT which identified several graphite behavioral phenomena that could potentially impact the radionuclide release. This resulted in five high ranking phenomena (high importance, low knowledge), with irradiation-induced creep as the leading one. Dr. Srinivasan mentioned international research being done on the first three (irradiation-induced creep, changes in CTE, and changes in mechanical properties), but not much work yet on the fourth, namely graphite spalling that could lead to blockage of fuel element coolant channels. Dr. Srinivasan also discussed the phenomena of high importance where knowledge base is considered medium. Tribology of graphite in impure helium environment is another area where NRC research may be needed.

Dr. Srinivasan then presented a slide showing the results of the EU nuclear graphite irradiation project using the HFR Petten reactor. 12 different graphite samples were being irradiated at 750 and 950 degrees Celsius. The results show typical graphite shrinkage characteristic with neutron dose, with reversal in the magnitude of shrinkage at some dose and then swelling beyond the turnaround dose. The shrinkage turnaround dose is traditionally accepted as the time when graphite blocks should be replaced. However, uncertainty of data is a very important factor in this determination. In addition to lots of questions from the members on graphite failure modes, Chairman Corradini noted that allowing any graphite shrinkage will lead to increased bypass flow that needs to be considered. The staff acknowledged that various safety issues including the above phenomenon need to be considered in research. The staff has recently contracted

ORNL to compare the PIRT results with DOE sponsored research and determine the gaps for NRC to pursue. The staff also scheduled an international workshop on graphite in March 2009.

Dr. Srinivasan described the staff strategy which was to conduct research in selected areas (graphite tribology and dust generation, air and water ingress) where current research needs to be augmented by additional confirmatory research. The staff will also monitor consensus codes/standards development, participate in national/international topical area meetings and graphite irradiation programs. For carbon-carbon and ceramic insulation material, the staff would await lessons-learned from the graphite and metallic material research experience.

While discussing dust generation from graphite and ceramic insulation material, the staff noted that the hot gas transfer piping from the reactor to the IHX is a concentric vessel with ceramic insulation protecting the metallic RCPB. Dr. Abdel-Khalik asked if the staff has looked at the possibility of a failure of the internal piping carrying hot gas resulting in its release into the return gas and eventual failure of the RCPB. Staff noted that the PBMR licensee was expected to address this type of new component internal failure scenarios, and agreed to get back to the members on this issue. [Post meeting note: The staff has informed the Committee that this particular scenario has been addressed in PBMR submittals. The transfer pipe, called the gas cycle pipe, is subject to a higher external pressure of the colder return gas. In case of the most likely failure of this pipe, which is a crack, the cold higher pressure helium return gas will leak into the pipe. Although this results in an immediate reduction of the cycle efficiency, nuclear safety is not compromised. In the extreme case, a collapse of the gas cycle pipe will result in pressurized loss of forced cooling. During events in which forced circulation is lost (pressurized or depressurized conditions), the pressure external to the gas cycle pipe will rapidly decline to equilibrium with the internal pressure].

In closing Dr. Srinivasan noted that the staff is keeping their research options open as DOE has not selected a design yet. However, the staff expected to do research in the areas of dust generation, and air/water ingress effects on graphite to support the NRC evaluation model development.

Members' questions resulted in discussion of amount of dust generation in pebble bed vs. prismatic designs, ISI requirements for structural graphite, and dust ignition. Dr. Srinivasan mentioned that graphite dust is a safety issue from fission product transport considerations. Dust, settled on various surfaces, could affect the emissivity, sometimes in a good direction. The dust could also become an issue for ISI video surveillance of core support posts/components because of deposition on surfaces and potentially hiding surface cracks or other imperfections. Currently there exists well-researched experimental data and literature, which indicate that the ignition and deflagration of graphite dust is an unlikely event during reactor operation. He also showed a simulation of graphite dust and corn dust ignition, in a confined environment, recorded as part of an experiment conducted to support decommissioning of graphite reactors. While ignition and explosion for corn dust occurred under these conditions, graphite dust showed much less propensity for ignition and hardly resulted in deflagration.

Structural/Seismic Analysis

Mr. Herman Graves of RES presented the staff's plan for structural/seismic analysis. The staff's objective is to develop its analytical capability to independently confirm the technical and licensing basis for safety related concrete structures. The staff envisions revising the Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion" to address advanced reactors. Upon members' questions, the staff noted that the RG is written for light-water reactors because it uses the core damage frequency as the performance criterion. So, the staff will need to develop similar performance-based criteria for reactors for which the core damage frequency may not be appropriate. The staff noted that they are working with INL in developing a licensing specification to address gaps in the current regulatory framework. Also, the staff issued NUREG/CR-6896 in February 2006 that addressed seismic analysis for deeply embedded structures.

Mr. Graves discussed the safety and technical issues related to NGNP. The structural analysis R&D areas include development of analytical models for reactor vessel internals and core support structures for non-linear seismic response, effect of sustained high temperature and transient aspects of heating and cooling of concrete structures, and seismic response/capacity of the multi-modular NGNP plant. Mr. Graves also discussed the related PIRT insights.

In the area of non-linear seismic analysis, the staff needs to determine the applicability and limitations of the existing finite element codes given the non-linear configurations of the HTGR core which is much taller than current reactor designs. The staff plans to take advantage of the analytical work that was done at Fort St. Vrain and Peach Bottom 1 in looking at the seismic behavior of the fuel elements, but no research has been done at this point. The only area the staff was working on was the effect of high temperature on concrete. Mr. Graves discussed the normal operation and accident condition temperature limits specified by the American Concrete Institute Codes for the current class of reactors, which are well below the expected HTGR conditions. However, there appears to be concrete mixes that are capable of surviving higher temperature conditions. The existing analytical methods for high temperature concrete designs have not been validated because of the lack of high temperature test data. The staff has contracted ORNL, to evaluate the existing concrete high temperature test data, not only from the U.S but also from Europe and Japan, to review physical property changes and determine design and evaluation criteria. The ORNL report is scheduled to be published sometime this year.

Mr. Graves presented a graph showing how residual compressive strength of concrete varies with temperature going up to 500 degrees Celsius. Mr. Graves stated that the aggregate type, interaction between the aggregate and the cement paste, and the presence of stress during heating are the main factors that influence the compressive strength of concrete. The staff expects the normal operating temperature for structural concrete to be between 300-400 degrees Celsius, although the design conditions are not known. The staff noted the need for temperature monitoring during HTGR operation. Concrete that encloses the RPV and supports the RCCS is important for containment/confinement integrity, and appropriate operating temperatures need to be established.

Mr. Graves presented a graph showing the effects of thermal cycling on concrete (e.g. refueling outages). At temperatures above 200 degree Celsius, the first thermal cycle produces a big decrease in the compressive strength. This type of change is not seen at 65 degrees Celsius normal operating condition. So, this behavior needs to be accounted for in the concrete design for the reactor cavity. Upon member Ray's question, Mr. Graves mentioned that reinforced concrete was used in the test represented in the graphs, but he did not know the ratio of the steel to the concrete. The staff pointed out that being inside the concrete, steel would not see the same temperatures and would be much less affected. Some members discussed the need for insulating the concrete.

For modular units, the seismic response will vary depending on the number of modules and design of the foundations. The staff expects to take advantage of the work being done in South Africa for the PBMR reactor. Mr. Graves then talked about various options for reactor cavity cooling as offered by the General Atomic PMR design before closing his presentation with a summary slide.

In his comment, member Armijo stated that he expected the problem of high temperature for concrete to be resolved by designing improved concrete, insulation on the concrete or a concrete cooling system (e.g., Fort St. Vrain design had active cooling of concrete). Member Armijo also wanted to hear about the pebble bed fuel reactivity response during a seismic event due to changes in fuel compaction. Some members were concerned about the graphite-ceramic support columns (for core integrity) since inservice inspection may not be possible. Chairman Corradini noted that at the next ACRS briefing the staff should have DOE and the contracted laboratories present to allow more detailed discussion of the design aspects.

Reactor Consequence Analysis

The meeting reconvened after a lunch break with Dr. Jocelyn Mitchell from the staff presenting the research plan on the reactor consequence analysis relative to the advance reactor technologies. The staff plans to use the MACCS code (MELCOR Accident Consequence Code System), Version 2, which in itself is technology neutral. However, the current input to the code is developed for light-water reactor technology, resulting in a need to consider the important differences in input that could stem from the advanced reactor technologies. The mix of the radionuclides and the chemical forms may be different for advanced reactors resulting in a different offsite consequence analysis. A discussion arose as to factors related to design and accident response that could impact the analysis. Dr. Mitchell mentioned that in the past, the staff analysis assumed a catastrophic failure of the containment resulting in a big (puff) release followed by an extended low level release. For NGNP, the staff would assume containment failure by excessive leakage, and hence no big puff release.

The staff uses ORIGEN code to produce the inventories of the radionuclides. Other analyses like the MELCOR code would produce the chemical forms and the amount of the release. The staff's effort would determine if there are any additional biologically important radionuclides that have to be added to the list and what the dose conversion factors are for all chemical forms of all radionuclides. The staff awaits input from other areas while the methodology for reactor consequence analysis is pretty well developed.

Digital I&C – Advanced Process Monitoring

Dr. Paul Rebstock of RES presented the research plan in the area of digital I and C (DI&C). Upon member Bley's question a discussion arose regarding the need for the DI&C staff and the human performance staff to work together, and if DI&C should play a role in the upfront design of an advanced reactor. Member Abdel-Khalik wanted to know how one would infer the reactivity state of the core. Because of the difficulty of installing in-core detectors inside the PBR core, the staff noted the availability of ex-core detectors for reactivity measurements like in a PWR, adding that unlike the PWR the ability to run in-core instrumentation periodically to calibrate the ex-core detectors, however, may not be possible. With the added complexity of independently moving pebbles in the PBR core, the staff stated that the subject of 3-D flux mapping was an open technical issue, and they intend to engage the DOE and the Chinese regarding their pebble bed reactor designs.

In staff's objective to develop a regulatory infrastructure in the area of DI&C to review an advanced reactor application, computer based control room designs with deeply integrated human factors would play a major role. Dr. Rebstock discussed some of the new technologies and advances in this area that staff has not reviewed yet, although some research has been ongoing to investigate the failures and vulnerabilities in the operation of some of the new devices.

Dr. Rebstock discussed the technical and safety issues associated with the advanced reactor challenging environmental conditions, and control schemes associated with multi-modular design. He mentioned that DOE has a group that is organized to develop advanced sensor technology for application to the HTGR high temperature, high-flux conditions. Upon members' questions, the staff indicated that in case of high uncertainty in measuring certain local reactor parameters (e.g., 3-D flux, temperature or mass gas flow) the staff would require the applicant to provide appropriate compensatory measures to ensure that there is adequate safety margin. For a multi-modular reactor design, one of the key elements of staff review would be the nature of the load and the way the load gets balanced among the modules. A discussion ensued regarding operation of multi-modular reactor plant especially in automated condition (as allowed in Europe).

Dr. Rebstock then presented three main areas of planned research, which included developing regulatory infrastructure in the area of advanced instrumentation, advanced controls, diagnostics and prognostics. The staff planned to begin work in FY2009.

Out of Reactor Radiological Safety & Safeguards R&D

Dr. Mourad Aissa from RES presented staff's plan for research in the area of non-reactor safety which included fuel fabrication, onsite storage, transportation and disposal of HTGR spent fuel and graphite. The staff's objective is to develop analytical tools for staff review in this area involving criticality safety for fresh and irradiated fuel and radiation shielding. The staff did not perform a PIRT in this area, and did not start any research work in this area at the time.

Member Abdel-Khalik wanted to know if a refueling strategy was developed for the prismatic design and how the fuel is to be moved. The staff explained that they expect procedures to be similar to Fort St. Vrain where individual graphite blocks were removed.

The two safety issues identified by the staff included ensuring subcritical condition for higher enriched fuel during fabrication and radiation shielding methods that address HTGR conditions. Dr. Aissa mentioned that graphite may act as a super-moderator sustaining fission in natural uranium, thus necessitating a review of the current regulations in this area and for developing guidance. The unique scattering characteristic of graphite (due to its crystalline structure) needs to be addressed in staff's methodology.

The staff plans to adapt the SCALE code system for analysis (criticality, radiation shielding etc.) of spent HTGR fuel storage with irradiated graphite and to address the higher expected burnup. This is consistent with the staff's overall plan to leverage the existing NRC code fleet and update only the modules that are impacted by the new reactor designs. The staff plans to get involved in the international data gathering activities, characterize HTGR spent fuel and graphite for analysis, and identify where more work is needed.

Risk-Informed Regulatory Infra-structure

Dr. Mary Drouin of RES presented the staff's research plan for developing a risk-informed licensing basis for advanced non-LWRs. Staff's research plan has been designed to support the recommended NGNP licensing strategy of Option 2 that uses deterministic engineering judgment and analysis complimented by design-specific PRA information to establish the licensing basis. In Option 2 the categorization and selection of LBEs will use conservative deterministic judgments to select the bounding LBEs and use PRA to select other LBEs to complement the bounding LBEs. The safety significant SSC selection and classification will be done deterministically. The staff will use a mechanistic source term.

The staff has identified three tasks in their research plan. The first task is the development of an integrated technical basis (scoping level PRA) for prioritizing and selecting the needed research. The second one is to develop regulatory guidance for establishing a risk-informed licensing basis. The last task is to develop regulatory guidance for the staff and licensees on how to implement the Commission's policy on defense-in-depth.

The scoping level PRA will augment the PIRT process and address seismic issues not considered in the PIRT. The staff will first determine the feasibility of developing it given the availability of methods and data, and then establish a plan which will identify the boundary conditions, needed inputs, level of PRA etc., and how this PRA would be used.

The second task of developing regulatory guidance for the identification of the licensing basis events and the safety classification has three major subtasks. The first one, developing a draft regulatory guide for internal review will involve policy and technical issues. Then the staff will perform a test of this regulatory guide on an actual design before finalizing it. The RG will provide guidance in six areas including, the selection of LBEs, safety classification of SSCs, attributes of the PRA, treatment of uncertainties, updating the PRA and LBEs, and documentation requirements. The staff currently plans to have a draft RG ready for ACRS review by the end of 2009, but that schedule involves some unknown factors.

An extensive discussion arose regarding the process of LBE selection and how to use the results. Use of NUREG 1860 methods, expert elicitation on HTGR failure events to consider (failure of the pressure boundary needed for escape of the source term), use of deterministic judgment to address areas of higher uncertainty, analysis of the LBEs (once selected), and acceptability of plant response were subjects of lengthy discussions. Dr. Drouin then discussed the staff's plan for the other five areas of the RG.

Regarding the defense-in-depth policy statement implementation guidance, the staff needs to develop a policy statement first. Dr. Drouin mentioned there were no activities in progress in the area of scoping level PRA development. The RG on LBE selection is being worked on and staff expected to prepare a draft for internal review by the end of 2009. The staff was working on a Commission paper in the area of the defense-in-depth policy statement as part of a planned development of risk-informed and performance-based rulemaking. The staff expected to use the lessons from the development of the NGNP licensing strategy and the review of the PBMR white paper submittals. [Post meeting note: As a result of a lack of lessons learned in the area of risk application from the licensing strategy work and effectively no staff review of the PBMR submittals, the staff decided to postpone work on the defense-in-depth policy statement].

Probabilistic Risk Assessment

Dr. Drouin then presented the advanced reactor research plan in the area of the PRA. Staff's plan includes development of the necessary review guidance to ensure applicant's PRA is of adequate technical quality (similar to RG 1.200), and PRA tools to support the NRC oversight process for advanced reactors (longer term). Dr. Drouin indicated that Commission direction might be needed in the area of risk metrics, quality and scope of the PRA and PRA maintenance.

To develop this regulatory guidance, the staff needs to go thru the iterative process of identifying the uses of the PRA, scope and level needed to support the intended use, and attributes needed for technical quality. Dr. Drouin also mentioned that an ASME working group had been developing a draft standard for the advanced non-LWR PRA, starting in a technology neutral manner. The staff is participating in this effort.

To support the PRA technical acceptability review, the staff would need certain tools, methods and data. Specific research areas have been identified and staff expects to use the experience in developing the regulatory guidance discussed above. Although no activities are in progress at the time of the meeting, the staff expected to initiate work in 2009 in the area of human reliability analysis, system reliability analysis and treatment of uncertainty, preferably in collaboration with the industry. Upon Member Ray's questions discussion arose on how to address uncertainty related to the design areas of NGNP that can not be possibly verified (e.g., lack of an ISI technology for core support structures). The staff indicated that similar to the current process for LWR PRAs, staff would ask the applicant to identify assumptions, sources of uncertainty and their impact. The staff will impose special treatment requirements, inspection and monitoring, and finally, if needed, design for failure mitigation.

On a longer term, the staff will need a baseline plant specific PRA for developing tools for the reactor oversight process. The staff will start with the scoping PRA and the extent of staff's task will depend on the quality of the applicant's PRA. After a short discussion on the status of the tasks and the schedule, Dr. Drouin completed her presentation.

Sodium-Cooled Fast reactors

Dr. Imtiaz Madni of RES presented the plan on Sodium-Cooled Fast Reactor (SFR) research. This area is not part of the staff's NGNP research. The staff's R&D objective is to conduct a technical infrastructure survey as no SFR design/construction work has been done in the US for a long time. Similar to the NGNP PIRT process, but limited in scope, the survey will be done in several technical areas: thermal fluids; nuclear; severe accident and source term; fuels and materials analysis. An initial limited scope survey has identified significant technical, safety and R&D issues and gaps in staff's knowledge. Upon members' questions, Dr. Madni stated that later work may include additional areas like instrumentation and PRA, and based on NRC review priorities a more expanded scope survey resembling a PIRT may be done.

As part of staff's objective of knowledge management, the staff has contracted ORNL to implement an SFR knowledge management project which produced technical documents, seminars and recommendations for training courses. The staff has developed a plan for additional work in 2009.

The staff is interacting with DOE in technical activities related to the advanced burner reactor and vendor interfaces like in Toshiba-4S. The staff had informed Toshiba that due to limited resources staff would not be doing any review of the 4S design at this time. Chairman Corradini asked if the staff is participating in OECD/NEA knowledge management work looking at cross-cutting areas.

Dr. Madni discussed the superior properties of sodium, as most liquid metal reactors (LMRs) are cooled by sodium, pointing out the ones (e.g., reaction with air and water) that require special approaches in design and handling. Upon Chairman Corradini's questions, Dr. Madni discussed relative merits of metallic and oxide fuel and how certain issues (e.g., sodium void) can be addressed by design. Various SFR technical and safety issues, design features to address them, domestic and international LMR experience, and major SFR test programs were mentioned or discussed. The staff intends to enter into collaboration with international test facilities.

The staff plans to evaluate the existing SFR models and analytical tools. Dr. Madni mentioned that the Super System Code series, developed by Brookhaven national Laboratory for the NRC was used to review and prepare the PSER for PRISM reactor. Upon Chairman Corradini's questions, Dr. Madni indicated that the staff wanted to pursue access to a later version of the SIMMER code, originally developed by Los Alamos for the NRC, and later improved by a collaboration of Japan, France and Germany.

Closing Statements

Dr. Rubin from the staff mentioned the desire for an ACRS letter after the then-planned March Full Committee meeting. Chairman Corradini opened the floor for members' comments. Member Bley stated that given such a gigantic catalogue of things to do, there needed to be a structure, a plan with a timeline taking into consideration how various parts of research activities fit together. Member Abdel-Khalik agreed with that observation. Consultant Kress said the staff's plan was very comprehensive and a good start. He was going to send rest of his comments to Chairman Corradini. Member Ray in his comments noted the short amount of time available given the broad range of tasks that need to be completed (for the staff to be able to review the NGNP application). He recommended more clarity and specificity regarding what the staff is trying to accomplish and early engagement with the applicant so that any needed changes can me made sooner than later.

In response to above comments, Dr. Rubin from the staff noted that more design specific information will be needed and the pre-application review in 2010 timeframe will help the staff obtain that information. The staff expects to improve specificity of their plan during the three years of the pre-application review. Upon Chairman Corradini's question Dr. Rubin indicated that he expected the DOE to select a design during the time before the pre-application review in 2010. But Dr. Kinsey of INL stated the DOE, in the relatively near term, was going to be putting out an offer of financial assistance to the industry. More than one response was expected and there was a potential for more than one design to be accepted. In that case, there would be a need for adjusting resources and schedule. As the NRC plan is based on one design selection, the staff will need to engage the Commission for direction if that happens. Chairman Corradini noted that he would put together his observation for the members to review in preparation for the Full Committee meeting. Where upon the meeting adjourned at 4:12 p.m.