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

June 9, 2011

MEMORANDUM TO: ACRS MEMBERS

FROM: Maitri Banerjee, Senior Staff Engineer **/RA/**
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGN SUBCOMMITTEE MEETING, APRIL 5, 2011, ROCKVILLE, MARYLAND

The minutes of the subject meeting have been certified on June 9, 2011 as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via e-mail: C. Santos
E. Hackett
H. Nourbakhsh
S. Basu, RES
S. Rubin, RES
M. Srinivasan, RES

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NUCLEAR REGULATORY COMMISSION
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WASHINGTON, DC 20555 - 0001

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

FROM: Dennis Bley, Chairman
Future Plant Designs Subcommittee
ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE FUTURE PLANT DESIGNS SUBCOMMITTEE REGARDING HTGR RESEARCH PLAN – APRIL 5, 2011

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting held on April 5, 2011, are an accurate record of the proceedings for that meeting.

/RA/ 06/09/11

Dennis Bley, Chairman Date
Future Plant Designs Subcommittee

Certified: June 9, 2011
By: Dennis Bley

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON
FUTURE PLANT DESIGNS REGARDING HTGR RESEARCH PLAN
ON APRIL 5, 2011, IN ROCKVILLE, MARYLAND

On April 5, 2011, the ACRS Subcommittee on Future Plant Designs held a meeting in Room T-2B1, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to receive a briefing from the NRC staff regarding their high temperature gas reactor (HTGR) research plan. The meeting convened at 8:30 AM. The meeting was open to the public.

Attendees:

ACRS Members	**RES Presenter**	Al Csontos
Dennis Bley (Chairman)	Jose Pires	Richard lee
Said Abdel-Khalik	Jeffery Wood	Kimberly Tene
Joy Rempe	Mary Drouin	**NRC Staff**
Sam Armijo	Stephen Fleger	Neil Ray, NRO
Mike Corradini	Yaguang Yang	Thomas Boyle, NRO
Harold Ray	Russell Sydnor	Matt Hamberstone, NRO
Tom Kress (Consultant)	**RES Staff**	Gregg Makar, NRO
ACRS Staff	Kathy Halvey Gibson	John Honcharik, NRO
Maitri Banerjee (DFO)	Don Carlson	Mike Norato, NRO
RES Staff Presenters	Kevin Coyne	
Sudhamay Basu	Thomas Herrity	**DOE and Others**
Joseph kelly	James Corson	Madeline Feltus, DOE
Mourad Aissa	Arlon Costa	Steven Reeves, DOE
Hossein Esmaili	Thomas Weaver	Dave Petti, INL
Stuart Rubin	Tarek Zaki	Janelle Zamore, DOE
Mukuteswara Srinivasan	Joe Kezly	Carl Sink, DOE
Shah Malik	Michael Rubin	Peter Pappano, ORNL
Amy Hull	Rasool Anooshehpoor	Jim Kinsey, INL*

* Participating over telephone bridge line

The presentation slides and handouts used during the meeting are attached to the Office Copy of the meeting transcript and may be consulted for additional information. The presentation to the Subcommittee is summarized below.

Opening Statement

Chairman Bley convened the meeting by introducing the ACRS members and the consultant. He noted that the current briefing was to discuss the RES staff's High temperature Gas-Cooled reactor (HTGR) Research Plan developed to support the review of a future NGNP application. He noted that the last subcommittee meeting on the subject was in January 2009. After stating that the telephone bridge line would be opened in both directions for receiving comments and questions at the end of the meeting, Chairman Bley invited the staff to begin the presentation.

Introduction and Staff Presentation

Ms. Kathy Gibson started the staff presentation by stating that the staff welcomes ACRS feedback on the program. Dr. Sudhamaya Basu of RES presented an overview of the research plan and expressed the staff's desire for a letter report from the ACRS after the planned May Full Committee briefing. The purpose of the NRC HTGR research plan is to develop analytical tools and capabilities to perform confirmatory safety analyses and provide technical basis for regulatory decisions in support of the NGNP licensing review. Also developing technical basis for identifying and resolving safety issues, any needed regulations, regulatory guidance, and staff technical expertise in the area HTGR were all parts of the plan. Many NRC staff involved in previous HTGR review have retired. Upon Member Ray's question about the utility of NRC experience from the past licensing of Ft. St. Vrain, Dr. Basu pointed out that that plant was licensed a long time ago when the rules were somewhat different.

Dr. Basu then presented an outline of the upcoming staff presentation and introduced the presenters. He mentioned the areas not covered in the staff's R&D plan, namely, fuel cycle (except for the neutronic aspects of spent fuel), security and safeguards, and siting. He pointed out that the 2005 Energy Policy Act mandated the first NGNP (an HTGR as defined in the 2008 joint DOE/NRC licensing strategy) to be a prototype located on DOE property under their ownership and site security program. Dr. Basu envisioned revisiting these issues through the development of the NGNP program.

The NRC NGNP R&D plan, in the most part, addresses areas that are common to the pebble bed reactor (PBR) and prismatic reactor (PMR) designs (given the lack of a down selection by DOE). It plans to use data from DOE and international research, and that furnished by the NGNP applicant. The staff's plan addresses only the NRC R&D. Since the last ACRS briefing in 2009, certain changes were made in the NGNP program. Given the current trend of lowering the reactor outlet temperature from 950 to around 750 ^{o}C, NGNP may be able to use materials that are already code qualified and the NRC may be able to validate its evaluation methods to the existing data. Also, the steam cycle (SC) for power conversion is being re-considered (making moisture ingress an issue), and the application of the process heat is being broadened from co-generation of hydrogen. Other developments that could impact regulatory considerations include co-location at an industrial site (vs. DOE facility), multi-module design, and UCO vs. UO_2 fuel. Dr. Basu noted that the arrival of the NGNP application would most likely be delayed beyond 2013.

Dr. Basu mentioned the status of the staff's code and analytical model development in the areas of thermal-fluids, nuclear analysis, and accident analysis. Experimental programs funded under the DOE-NRC Interagency Agreement are supporting NRC data needs. In addition to the expected participation in the Japanese High-Temperature Engineering Test Reactor (HTTR) loss of forced circulation (LOFC) program through OECD, NRC has a number of test facilities and experiments ongoing at US universities under the Nuclear Energy University Program (NEUP) initiative. The staff work in the area of fuel and graphite are mostly analytical (PARFUME and MELCOR development). In addition to monitoring the Idaho National Lab (INL) advanced gas reactor (AGR) fuel program, the NRC is participating in the INL advanced graphite creep (AGC) program and Oak Ridge National Lab (ORNL) graphite research (stored energy release). Dr. Basu completed his briefing by mentioning NRC work in other areas like high temperature material, seismic, PRA, human factors and digital I&C. Dr. Bley noted that degraded I&C interface with human factors is a consideration that must be taken into account.

Dr. Joe Kelly presented an overview of the staff's program for NRC NGNP evaluation model (EM) development. The program includes development of SCALE (nuclear analysis), MELCOR (accident analysis), PARCS/AGREE (core neutronics simulation/ thermal-fluids module for gas reactors) and the supporting experimental work. The goal is to develop the capability to evaluate the radiological consequences to the public. The EMs address normal operation (for distribution of fission products (FP) within the helium pressure boundary), initial (blow-down) and delayed (heatup) FP release phases. He presented a schematic of the EM and noted the emphasis on normal operation as it determines the magnitude and distribution of FPs within the system over the life of the plant. SCALE code for nuclear data preprocessing will use 23 to 26 cross-section groups unlike only 2 to 4 in LWR models. He then mentioned how MELCOR is tied in to calculate the distribution of the released FP within the system.

To elaborate upon how changes in the core geometry with irradiation is accounted for, Dr. Kelly noted a box labeled "Thermal & Irradiation Geometry Changes" that collects the PARCS/AGREE thermal-fluids output to determine bypass flow, which is then fed into MELCOR steady state thermal-fluids calculation for determining peak fuel temperature and the FP release during normal operation. A long discussion followed regarding modeling the thermal and irradiation response of PMR graphite structural elements and Dr. Kelly noted an NRC program at Argonne National Lab (ANL) on irradiation damage structural analysis of one fuel element (DOE/INL plans a much bigger undertaking of modeling the full core). Member Armijo pointed out the need to use the lessons from the past HTGR reviews.

Consultant Kress noted the need for modeling the pebble flow in a PBR and the possibility of pebbles getting stuck as they pass through the vessel. The INL model, PEBBLES, is a discrete element mechanics code to model pebble flow that calculates the forces between the pebbles and the rolling and sliding friction. Regarding the need for experimental data to verify code predictions, Dr. Kelly noted the international cooperation with HTR10 in China.

Dr. Kelly discussed the tasks involved in EM development and the status of MELCOR code tasks. He showed the results of a meso-scale conduction benchmarking of the MELCOR prediction of

PMR core temperatures against a 2D finite element calculation. While indicating how the codes would be utilized, Dr. Kelly mentioned the activities related to the SCALE, PARCS and AGREE codes, and how the Symbolic Nuclear Analysis Program (SNAP) is utilized as an interface tool. He summarized his presentation by saying that the preliminary code assessment against the existing database is beginning, and the staff plans to have its independent confirmatory analysis capability ready by the end of 2013.

Dr. Mourad Aissa discussed the SCALE code (version 6.1) development in more detail. Nuclear data are taken from ENDF 7 to generate the cross-section library. He discussed cross-section processing, improvements made to multi-group resonance processing methods and code applications, and benchmarking against the HTTR annular fuel block and full core k_{eff}. Members noted the full core results were consistent code-to-code (continuous energy and multi-group), but varied from the HTTR experimental result (reason yet unknown). Then a long discussion followed regarding the meaning of the reported sigma (uncertainty). It was concluded that while the sigma for the code calculations represented the statistical uncertainty, the sigma for the experiment resulted from the Japanese team's aggregation of all the classical uncertainties involved in an experiment. A better match between the experiment and code predictions was observed for HTTR control rod worth numbers. Dr. Aissa also presented similar comparisons against the HTTR temperature coefficient and shutdown margin results, and code-to-code comparison results for the HTR-10 first critical core and HTR-PROTEUS (hexagonal closed packed pebbles with different core configurations) benchmarking. After discussing SCALE-PARCS integration where some exercises were run both for the PBR and PMR cores, and the ongoing activities that include code prediction biases and uncertainties, Dr. Aissa completed his presentation.

Dr. Hossein Esmaili presented the HTGR MELCOR FP release and transport modeling approach. The staff will use the existing MELCOR models for transport and deposition in the primary system and containment, and develop additional models as needed for specific HTGR phenomena (e.g., diffusion release for TRISO fuel particles, release of FP through the matrix and graphite block for the prismatic, and additional transport models for dust generation, deposition, and lift-off), and will use experiment (e.g., AGR for diffusion coefficient) and other code inputs. The question of MELCOR version control came up as the same code is used for LWR applications, and Dr. Esmaili explained the code flexibilities and mentioned the QA control. The staff has developed an air ingress model following LWR hot leg natural circulation, and water ingress is yet to be modeled.

Consultant Kress noted the treatment of FP transport through graphite as a diffusion process may or may not be right. One could question the use of Booth-type models because such models do not reflect the very large thermal gradients (~1000 degrees/cm) and the consequent thermal diffusion that would be encountered in the fuel. A long discussion ensued on how diffusion coefficients for fuel kernel, matrix and graphite are treated in the Booth model, with Dr. Petti from INL explaining how the effective diffusion coefficient is affected by fuel burn-up, and the radionuclides involved. He also noted that the AGR program on designed to fail (DTF) fuel under a range of burn-ups. Dr. Esmaili presented the IAEA CRP-6 benchmarking results for Cesium (Cs) release fractions for the bare and intact kernel and the kernel with SiC and outer

PyC removed at 200 hours in three different temperature conditions. He explained how an accelerated normal operation case is run to obtain FP distribution and release rates into the primary system for each isotope. He then showed results (circulating and deposited Cs mass at different locations of primary system in PBR400) obtained from running the code for test purposes. The staff expects to complete the MELCOR modeling by the end of 2011.

Dr. Kelly presented the development of the core analysis code PARCS/AGREE for both normal operation steady state conditions, and for reactivity insertion transients where a 3D kinetics capability is needed. PARCS, the core neutronics simulator, had some preliminary validation work done for the PBR (OECD PBMR-400). Dr. Kelly discussed how the code would be applied to a prismatic core (triangle-based polynomial expansion nodal method), its interface with SCALE for cross-section generation, and mentioned the future work on the microscopic depletion capability, modifications for moisture ingress, and participation in a new OECD benchmark for the MHTGR-350 design led by INL. He then discussed four cases of numerical benchmarking to the HTTR and areas where more work was needed.

Dr. Kelly discussed the AGREE (Advanced Gas Reactor Evaluator) code already validated for the PBR. PARCS/AGREE was extended to model the PMR core and was used to review Oregon State University (OSU) High Temperature Test Facility (HTTF) scaling analysis. The new PMR model includes a bypass flow model, and a three dimensional triangle-based heat transfer model. He showed a comparison between a calculation done for the MHTGR-350 and the HTTF (reduced-scale model) for depressurized conduction cooldown (DCC). The results, isotherms calculated at the core mid-plane and at an axial plane, compare favorably with some differences. In the first case, there was a minor difference between the two for the temperature drop between the core barrel and the reactor pressure vessel, explained to be due to the impact of scaling on emissivities (too much surface area in a reduced-scale facility). For the axial case, the isotherms in the HTTF were too close at the top. This was thought to be because the top reflector wasn't thick enough. Also, in the full-scale, there was a very small recirculating flow that didn't show up in the HTTF. The staff is reviewing the differences.

The meeting reconvened at 1 pm after a lunch break with Dr. Kelly discussing the NRC supported experimental program for NGNP EM code validation. This is over and above the majority of the experimental database that will be provided by DOE and/or the applicant. The staff is developing a program with ORNL to monitor the manufacturing process for the fuel and a QC program. He mentioned two integral test facilities, the high-temperature (>700 ^{o}C) reduced pressure reduced scale HTTF (mentioned before) that is a joint DOE-NRC program, and a proposed OECD program run by JAEA at the HTTR for the LOFC test. He also mentioned several separate effects test facilities at Texas A&M University, Pennsylvania State University (PSU) and University of Wisconsin (UW).

The initial core of the integral OSU test facility is going to be prismatic based upon the MHTGR-350 design with future option to load a PBR core. It is designed to model a DCC transient for a range of breaks and an operational reactor cavity cooling system (RCCS). Dr. Kelly discussed the test setup, DCC phases, and other scenario (normal operation, pressurized CC) test

capabilities. To scale thermal conductivity (to get the right radial temperature drop during a conduction cooldown), the graphite blocks are replaced with ceramic blocks. Upon Member Armijo's question, Dr. Kelly noted that if a PBR core is to be installed in future it would look much like the old SANA test facility in Germany or the HTTU facility that was in South Africa at Potchefstroom. He mentioned the limitations of such test.

The HTTR is a 30 MW, graphite moderated, helium cooled PMR that has been operated with an outlet temperature as high as 950 ^{o}C. The three LOFC cases were run at varied reactor power, outlet temperature and flow rate conditions with simulated ATWS, and in one case with no vessel cooling or RCCS. The separate affects test at Texas A&M is the pebble-bed flow and heat transfer test with four components. The first one looked at pressure drop at both annular and circular beds. The second component is the radial porosity distribution test that used matched index of refraction with laser-induced fluorescence and image capture and image analysis software to map the entire randomly packed pebble bed. The radial velocity (flow) profile (to quantify the wall bypass effect), and the pebble convective heat transfer tests are the third and fourth components that are ongoing. The air ingress test at PSU is to study the geometric effects on the air-ingress flow rate during lock exchange including break orientation, and the L/D effect on the pipe and break geometry. Scoping studies with water-brine are completed and scaled experiments with helium-air and simulated break locations are starting.

The UW high temperature spectral emissivity test uses an infrared irradiation system with samples held in a SiC block with a heater and thermocouples on the block to control its temperature. Gas lines allow changing the atmosphere and aging of samples. The emissivity graphs are plotted for SA 508 (candidate material for reactor vessel) and 316 SS (candidate material for the core barrel and the actual reactor vessel at the HTTF) for 500 ^{o}C and 700 ^{o}C air at 1 to 4 hours. SA 508 showed very quick aging within 1 hr and emissivity above 0.8. 316SS at 500 ^{o}C showed bare metal emissivity (0.3 to 0.4) which at 700 ^{o}C went up to about 0.5 and aged very quickly. Within one hour of exposure 316SS got as much corrosion/oxidation on it as it was going to get, somewhere between oxidized and bare metal.

Dr. Kelly discussed the bypass flow test facility, with three hexagonal columns stacked two blocks high, being developed at the Texas A&M. It will be using the mixed index of refraction coupled with particle image velocimetry for flow measurements in the axial bypass channels and cross flow through horizontal planes. The goal is to get loss coefficients to be able to model the bypass flow.

Dr. Stuart Rubin made a presentation on HTGR fuel performance and fuel FP release. He provided a high level overview of the DOE's AGR fuel development qualification program and the NRC'S HTGR fuels R&D in support of the NGNP licensing review. He discussed the AGR fuel program objectives (manufacture high quality fuel that meets accident condition performance requirements, improve fuel performance and FP transport models, and generate data to support NGNP operational and safety analysis model development and validation), program elements and status of activities. The purposes of AGR-1 through 8 irradiation and accident condition simulation tests are many fold: AGR-1 is a shakedown test with lab scale fuel

production to optimize fuel design and test setup; AGR-2 is a large scale fuel performance demonstration; AGR-3/4 contain DTF fuel to obtain data to develop FP diffusion coefficients and retention behavior; AGR-5/6 are formal fuel qualification proof tests with enough particles to meet the confidence level on particle failure rates; AGR-7/8 is to obtain data to validate the FP transport codes and particle failure rate codes by running experiment to actually fail particles intentionally. The last part of the AGR program includes out-of-pile tests to obtain data for developing and validating FP transport models outside the fuel within the helium pressure boundary and in the reactor building (mechanisms: lift-off, plate-out, absorption etc).

Fuel irradiation has been completed in AGR-1 and post irradiation testing (PIE) underway with accident condition heatup testing later this year. AGR-2 fuel is undergoing irradiation in ATR, and AGR-3/4 fuel fabrication has started.

The NRC plans to use a state-of-the-art HTGR TRISO coated fuel particle code PARFUME developed by INL that simulates fuel particle behavior, calculates the failure probability for a population of fuel particles, and calculates FP transport and release for both the normal and accident conditions. It models phenomena that lead to fuel failure, and FP transport in fuel particles (failed and intact), fuel matrix and graphite elements. A long discussion followed regarding the statistical variations involved (e.g., fuel particle dimensional parameters etc) and how they are handled in the PARFUME code calculations (e.g., particle failure probability). The members requested a future presentation on this subject.

Dr. Rubin discussed the analysis steps involved in PARFUME and how they differ from the MELCOR code. PARFUME does the stress-strain calculation and calculates the peak stresses, and then compares it to a Weibull failure probability curve as a function of stress to come up with a failure probability for particles that are at that stress (i.e., failure fraction prediction). This time consuming failure predictive process is not part of MELCOR. He then discussed the future development activities for PARFUME with use of AGR data that would make the code more specific to the actual fuel properties rather than using the German data.

A discussion ensued regarding the definition of fuel particle failure. The code models particle failure as failure of all TRISO layers. The code does not differentiate failure of individual layers, but manufacturing defects in layers are handled statistically as fractions of defective particles and entered in the code for core-wide FP release calculation. During irradiation, online measurement of FP release (Kr, Xe) would indicate particle failure (as even the PyC layer would hold noble gases). During the PIE, particles are deconsolidated to measure the Cs content in the fuel matrix which would indicate defective SiC layer. Post PIE safety testing (heatup) of particles provides a measure of metal release.

The staff does not plan to use PARFUME as part of their EM because of the long run times, and for the hundred or more nodes of PARFUME that MELCOR would need to do a calculation of particle failure rate at any given time. For coated fuel particle failure rate predictions MELCOR will utilize an empirical fuel particle failure fraction versus temperature curve or failure fraction versus temperature and burn-up response surface as a user-specified input. The staff will use

PARFUME to evaluate the applicant's failure prediction model in their safety analysis codes and use the code for sensitivity studies to assess the affects of variations in fuel particle geometry and properties, and changes in the irradiation conditions on particle failure probability. The staff will also use the PARFUME predictions of IAEA CRP-6 benchmark cases for verifying MELCOR fuel FP release models.

Dr. Rubin discussed the UW research project on Cs transport in SiC. The objective of this project is to assess the applicability of historic (German) Cs diffusion data to new fuel fabrication. The modeling results for volume/bulk and grain boundary diffusion mechanisms were discussed. In bulk diffusion, Cs defects were found to be too unstable to allow adequate Cs solubility. No fast diffusion paths were found for Cs under grain boundary diffusion, unlike Ag. It was concluded that there may be other mechanisms affecting the solubility of Cs and the possible effects of neutron irradiation on Cs diffusion in SiC needed to be investigated. With a discussion of the staff activities for developing the regulatory oversight and inspection of the fuel fabrication facility, Dr. Rubin completed his presentation.

Dr. Srinivasan presented an overview of the current status of NRC's research on nuclear graphite, the outcome of which was to inform the technical assessment and independently confirm the applicant's technical basis for regulatory and safety decisions on graphite core components. Similar to the other areas of NGNP research, the staff plans to leverage DOE activities and international experience. Dr. Srinivasan discussed the staff involvement in nuclear graphite code development activities, with the ASME working document for Division V component design code prepared. Dr. Srinivasan noted that mitigation and compensatory measures may be needed to address risk conditions. Dr. Rempe wanted to know how no longer available H451 was being replaced. Dr. Srinivasan noted that ASTM issued specifications for two classes of graphite classifying by properties. The staff has been providing technical input to DOE research regarding many of the high importance low knowledge phenomena identified by a past PIRT. These include dimensional stability, irradiation creep, coefficient of thermal expansion under irradiated condition, etc. However, issues of spalling fracture, tribology, material loss under air/water ingress would require more research.

ORNL has been working for the NRC on stored energy release of previously irradiated graphite. Past work indicated it to be not significant for HTGRs operating above 300 ^{o}C. ANL has been working on a finite element stress analysis tool, to be available by 2013, for independent verification of design stresses and margins. This code will incorporate the inherent nonlinear elastic stress-strain behavior of graphite, spatial variation in temperature and flux, and the contribution and role played by irradiation creep to changes in graphite properties.

Dr. Srinivasan discussed the graphite burning issue by noting that although reaction with oxygen became significant around 475 ^{o}C, for solid graphite blocks the reaction is self limiting due to the byproducts of the oxidation process. Also, what burned at Chernobyl was the Zr cladding, and the long duration Windscale fire involved the burning of uranium metal with relatively small graphite oxidation. During graphite manufacture, high temperature conditions (2800 ^{o}C) make the graphite glow red waiting to cool in air. He mentioned past works, i.e., a 1978 Brookhaven National Lab (BNL) report that looked at combustion hazards resulting from the HTGR depressurization and air or water ingress, with a potential for flammable mixture of hydrogen/CO

mixed with air. He addressed the issue of graphite dust explosion, noting the past and present work indicated it not to be a concern.

Dr. Shah Malik presented the staff activities regarding high temperature material research, primarily in the area of creep and creep-fatigue crack growth under HTGR conditions. This is an issue for the reactor vessel, IHX/steam generator, and the cross vessel. Most of the available modeling work addresses creep crack growth under a steady state (time independent) secondary creep and not the significant non-steady state tertiary (and sometimes primary) creep regime. He discussed the staff's road map for the development of high-temperature, time dependent creep-fatigue probabilistic flaw evaluation computer code with validation data coming from DOE or applicant's research. The staff plans to incorporate the flaw evaluation module into an NRC modular probabilistic fracture mechanics code like the XLPR (extremely low probability of rupture) code. Dr. Malik then discussed the time-dependent fracture mechanics methodology development work at Oak Ridge, the data needs and major milestones. Upon Member Armijo's question, Dr. Malik noted that at present the data generation activities address nickel-based alloys 800H and 617, and not Haynes 230 and Hastelloy X/XR.

Dr. Amy Hull presented the staff activities aimed at the development of inservice inspection (ISI) and NDE codes. She noted that traditional ISI methods were not sufficient to address the high temperature longer duration operating cycles of an HTGR. The PNNL is looking at the reliability of acoustic emission and other on-line monitoring methods with a regulatory guide on HTGR ISI in mind. Dr. Hull discussed the ASME code development activities, staff involvement, and the new approach for reliability and integrity management (RIM) that would drive design, fabrication, inspection, operation and maintenance requirements.

Dr. Jose Pires presented the NRC research on HTGR structural and seismic analysis capability development. The staff completed a review of high temperature effects on concrete structures and its impact on the existing codes. Research on radiation effects on concrete and soil-structure interaction of deeply embedded or buried structures is starting. Multiple modules on a common foundation with addition of varying number of modules, seismic interaction between adjacent structures for deeply embedded containment buildings, and nonlinear dynamic response of graphite reactor internals (graphite blocks held by dowels and keys) are issues where more work is needed. Upon Member Ray's question on how time dependent changes to the graphite reactor internals dynamic response are addressed in this work, the staff referred to the work on ISI and RIM. Member Armijo wanted to know if seismic response of a pebble bed, its densification with the possibility of reactivity addition, was being addressed. Dr.Pires noted computer codes that model the phenomenon of densification, but staff has not started such work yet. Dr. Pires then discussed the effects of high temperature on concrete with research data showing no impact up to 200 ^{o}C. Codes and standards activities are envisaged. The staff would coordinate with DOE and INL to resolve key technical issues and close R&D gaps.

Regarding Member Ray's question on monitoring the performance of graphite and other core structure components for environmental and age-related degradation, given the longer refueling cycle of the NGNP reactors, thus requiring performance monitoring during reactor operation, the following information was provided by the staff after the meeting:

(A) During Outages:

a. The EDF-energy uses a channel bore measuring unit system (CBMU), which is a specially designed and constructed profile measuring device and which is inserted into selected fuel and

control rod channels. Essentially by tracing the bore surface, a tracing is obtained at several locations along the axis of the channel from which out-of-roundness or ovality can be obtained, which then can be translated to permanent deformation of the graphite block.

b. The same CBMU also can find cracks, if any, since the stylus will feel the surface. This way, the then British Energy found cracks in the fuel channel during 2002 outage of an AGR, and then later during 2003 in a control rod channel. They could follow the trajectory of the crack as well. There has not been any substantial dust deposit to prevent good and clean observation in AGRs. (If there is substantial dust in HTGR, the deposition of the dust on cracks could potentially yield ambiguous observations and pose problems contributing to false-reject or false-accept using flaw disposition methods.

c. EDF-energy also uses remote TV camera to image the surfaces of the channels to assess the surface condition of the channels. This method is also being used by the Japanese in their HTTR.

d. EDF-energy is also developing eddy current techniques for locating flaws and cracks and has made good progress during the last 5 years, external to the reactor. They intend to apply this success to in-reactor during outage.

(B) During reactor operation:

a. There are currently no convenient techniques, which are being applied. However, acoustic emission is being studied by the Japanese, in the laboratory, as a possible technique to observe the initiation and propagation of potential cracks. This technique is not applicable during outages, since it is expected that there will be no crack propagation due to the absence of loading. Also, there is potential for crack closure when the load is removed.

(C) ASME Activities:

For metallic materials, ASME inservice inspection code is being developed, akin to RG 1.45, for monitoring helium coolant leakage and establishing criteria. The ASME has yet to begin effort for Section XI inservice inspection requirements, which would, of course, be design dependent and will involve active participation by reactor designers and vendors. None in the industry has stepped forward. Presently, the staff does not intend to initiate any research on online graphite core monitoring.

Jeffery Wood and Mary Drouin presented staff activities in the area of HTGR PRA. The RES staff has recently prepared a draft regulatory guide on establishing licensing basis events (LBEs) using a PRA approach to support NGNP licensing. The staff completed a planning study to identify the gaps. High level issues identified from this study include the level of detail for the base PRA, risk metrics and technical basis for criteria, operating states to be analyzed, and unique hazard impacts (impact of seismic events on prismatic blocks). Additionally, detailed issues that need to be addressed include unique HTGR features affecting potential initiators, initiating event frequencies, success criteria for events, and unique HTGR environment and conditions for modeling system performance. New issues involve phenomena like air/water ingress modeling, aerosol transport, and human factor issue involving control of multiple units.

Stephen Fleger presented an overview of the confirmatory human factor (HF) research undertaken by the staff to support NGNP and to update regulatory guidance documents. The scope of work included human performance issues beyond the new Generation III design and that associated

with SMRs. A model was developed by the BNL for the staff to describe the concept of operations (ConOps) for SMRs, and identify human performance issues related to the multi-module operation. The ConOps "dimensions" address: staffing, qualifications and training; management of normal, off-normal and emergency conditions; management of maintenance and modifications; and roles and responsibilities. The staff plans to develop HF engineering review guidance. The staff's R&D is also investigating the effects of degraded I&C systems on human performance and on plant operations. This is a follow-up to a 2010 BNL report. [*Post meeting comment: the Committee issued a letter report on 5/24/2011 (ML11136A225) on the subject*]. Dr. Fleger mentioned that RES had procured a simulator in-house to support independent validation, and had been using a bilateral agreement with the European Halden reactor to support the integrated research.

Yanguang Yang presented the staff's R&D efforts related to the NGNP I&C needs. He noted the unique factors, namely the harsh environments (high temperature reactor), co-generation, operating parameters and reactor response, and variety in designs (Pebble bed vs. prismatic) that pose challenges. He discussed the expected I&C design features for the NGNP to be supported by digital I&C with advanced resilient control including cyber security. Three staff research projects in I&C were also mentioned. With a discussion of the design unknowns for I&C, Dr. Yang completed his presentation.

Sudhamay Basu presented the status of the staff's HTGR process heat application R&D. He noted limited progress primarily due to the DOE's lack of a design selection for the reactor and the process heat application domain. The scope of the research included assessment of the blast loading on the reactor to calculate safe separation distance, heat and mass exchange between the reactor and the coupled process heat plant, component degradation issues, and modeling of toxic gas dispersion and tritium migration. Some data and predictive tool exist for some of these elements.

The DFO Ms. Banerjee presented the follow-up items resulting from the day's discussion. The first one was about the members question on how the staff was factoring into the research program past licensing experience like that of Fort St. Vrain. Regarding presentation of uncertainties in the HTGR application of the SCALE code the members wanted to see a systematic process for considering and presenting the use of uncertainties in staff's decision-making process. The members wanted to see the user manual on PARFUME Code (*post meeting note: a copy of INL/EXT-08-14497, "PARFUME Theory and Model basis Report" was provided to the members*). The members also wanted a future presentation on PARFUME, the INL work on the fuel development process, process control and measurement science. The long-term verification of the integrity of graphite structural members, integrated fuel cycle safeguards and security, siting issues (if other than at the DOE facility) were areas of interest to be addressed at a future briefing.

Chairman Bley asked for members' comments. Member Armijo mentioned his interest in the INL work in the fuels area and the work at the University of Wisconsin. He also reflected positively on the staff's R&D program given the absence of a design selection by the DOE. A discussion ensued on the ability of the staff to carry out useful R&D in future before the work has to stop without a design selection.

ACRS Consultant Kress felt that the research plan was very thorough, but ambitious, due to the lengthy nature of developing FP release data. Regarding NGNP white papers developed for the staff, he commented positively on use of the frequency consequence acceptance criteria for the

licensing basis. He expected the R&D plan to address steam graphite oxidation, and technical justification for the assumption that the FP species would be the same as in LWRs after passing through high-temperature graphite. For the FP release (Booth-like) models, getting effective diffusion coefficients for every FP species may not be possible such that the LWR method of scaling to Cs may be necessary. The phenomena of graphite plate-out and resuspension, and the amount, nature, size, and shape factors of the dust particles generated during normal operation have to be modeled to address the amount of allowable dust and its contribution to FP release. Dr. Kress emphasized the need for control of the fuel manufacturing process, methods of statistical sampling, and testing to ascertain fuel quality. He noted the difficulty of movement of control rods through the irradiated and hot graphite pebble bed in older designs, and modeling of hot spots due to conglomeration of pebbles was an area of concern. He felt that defense-in-depth would be handled better under design basis vs. mechanistic source term considerations. For multi-module plant design the site acceptance criteria should consider all reactors on the site and common mode failures in multiple modules under natural phenomena like a seismic or flooding event.

After checking with the members of the public present in the room and over the telephone bridge line for comments, Chairman Bley thanked the staff for their presentation. The meeting was adjourned at 5:22 pm.