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From: Stephen Koenick -nrr
To: Adel El-Bassioni; Clifford Munson; -NRR David Jeng; Edmund Sullivan; -NRR Goutam Bagchi; -NRR Mark Rubin; Michelle Hart; Robert Palla; Shanlai Lu; Shih-Liang Wu; Undine Shoop; Walton Jensen; William Koo; Yuri Orechwa
Date: Wed, Jun 26, 2002 3:55 PM
Subject: Request for comment on draft Commission Memo of Reactor Research Plan

To the addressed,

I have been instructed to resend the attached Commission memo for comment. The paper highlights 4 topics of the plan, 1) Fuel analysis, 2) Materials analysis, 3) Reactor Systems analysis, and 4) PRA. Hopefully your RES counterparts worked with you in the development of this memo.

RES needs comments by COB this Friday. It would be expected that your review would be very high level and the comments would reflect that. This will not constitute concurrence since NRR was not asked to be on concurrence. Please send comments to me via email by Friday at 4:00pm and I will pull them together before I go home.

I apologize for the quick turn-around. Perhaps at the J.Flack briefing tomorrow at 11:00-12:00 pm in O 9B2, we could discuss the poor interoffice communication especially in relation to the period of time associated with NRR review of RES products.

Please let me know if you have any questions and if you are unable to attend tomorrows briefing I can relay any questions you may have.

Stephen S. Koenick
Project Manager
U.S. Nuclear Regulatory Commission
voice: 301 415-1239
email: ssk2@nrc.gov

CC: Bill Bateman; Eugene Imbro; F. Mark Reinhart; James Lyons; Jared Wermiel; Kamal Manoly; Mark Rubin; Michael Johnson; Ralph Caruso; Stephanie Coffin; Thomas Bergman

S-7

TAB 072

From: Donald Carlson
To: Koenick, Stephen
Date: Tue, Jun 25, 2002 4:26 PM
Subject: SECY

Steve,

As requested, see attachment.

Don

TAD 072.1

MEMORANDUM TO: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield

ROM William D. Travers
Executive Director for Operations

SUBJECT: COOPERATIVE RESEARCH TO SUPPORT KEY REGULATORY DECISIONS ON ADVANCED REACTORS

The purpose of this memorandum is to inform the Commission about the primary research areas and associated cooperative efforts that will support key regulatory decisions on non-light water advanced reactors, (i.e., high temperature gas cooled reactor (HTGRs) with pebble or prismatic fuel elements).

The information in this memorandum and its attachments builds upon the advanced reactor research plan that the staff has been developing pursuant to commitments made in the Future Licensing and Inspection Readiness Assessment (FLIRA) report (SECY-01-0188, dated October 12, 2001). A draft copy of the research plan that was sent to the Advisory Committee on Reactor Safeguards (ACRS) on March 28, 2002, was forwarded to the Commission. The staff intends to provide the complete advanced reactor research plan to the Commission in the Fall 2002.

A separate SECY paper is being prepared to inform the Commission of the key non-light water reactor policy issues on which the staff will seek Commission guidance early next year. These policy issues, resulting from the unique design features proposed for advanced non-light water reactors, are strongly related to the technical issues and research areas outlined in this memorandum. Accordingly, we see the development of data and capabilities in these areas as an important step for implementing Commission policy in the licensing and certification of these new reactor designs.

Key activities in the advanced reactor research plan include the following: (1) development of regulatory decision making tools based on the risk-informed, performance-based principles; (2) accident analysis (PRA methods and assessments, human factors, and instrumentation and

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TAD 072.1.1

control); (3) reactor/plant systems analysis (thermal-fluid dynamics, nuclear analysis, and fission
product release and transport); (4) fuels analysis and testing (fuel performance testing, and fuel qualification); (5) materials analysis (graphite behavior and high-temperature metal performance); (6) structural analysis (containment/confinement performance, external challenges); (7) consequence analysis (dose calculations, environmental impact studies); (8) nuclear materials safety (covering enrichment, fabrication, and transport) and nuclear waste safety (covering storage, transport, and disposal); and (9) nuclear safeguards and security. Of these nine research areas, four are highlighted as playing an especially significant role in shaping future regulatory decision-making activities for non-LWR reactors. The four research areas are summarized below and additional information is provided in Attachments 1-4.

In planning NRC research activities, the staff recognizes that the applicant of a new non-LWR design has the primary responsibility to demonstrate the safety case of the proposed design. The staff's role is to develop the safety limits a new reactor design must meet, and evaluate the adequacy of the safety case through the identification and assessment of underlying technical bases. Toward that end, cooperative research agreements will be established, which involve exchange of research results, with the European Commission (EC), the Japanese Atomic Energy Research Institute (JAERI), MIT and the Department of Energy (DOE). However, It is necessary for NRC to contribute its own research results to gain access to extensive research taking place in Europe and Japan.

I. Fuels Analysis and Testing

Considerable research has been conducted worldwide to understand the behavior and safety performance of particle fuel for high temperature gas-cooled reactors (HTGRs) such as the pebble bed modular reactor (PBMR) and the gas-turbine modular helium reactor (GT-MHR). However, significant technical issues still remain that will require safety research to support the safety review of HTGR license applications.

Significant HTGR fuel irradiation and accident condition testing has been conducted to understand the behavior and safety performance of HTGR fuel during normal operation, design basis accidents and accidents beyond the design basis. However, these programs were not specifically directed toward PBMR or GT-MHR conditions and were generally focused on showing acceptable fuel performance within the licensing-basis conditions. Only minimal testing has been conducted at temperatures, burnups, fast fluences, power levels and accident conditions(e.g. air ingrers) that probe the margins to determine where increased fuel failures occur. This has resulted in large uncertainties surrounding the failure margins for HTGR fuels. The advanced reactor research plan includes fuel irradiation and accident condition testing to (a) obtain data needed to establish operating and accident fuel safety margins, (b) assess the acceptability of an applicant's fuel irradiation and accident simulation testing programs, (c) verify an applicant's claims of fuel performance and fission product release during operations and accidents, and (d) provide data to develop and validate independent analytical fuel performance tools. These data will be needed to support the application of probabilistic risk assessment in the selection of design basis as well as beyond design basis events, develop the requirements to implement the use of plant-specific mechanistic source terms in lieu of a prescriptive sources term specified in TID-14844 or NUREG-1465, and hence facilitate HTGR licence application reviews.

Analytical tools provide a powerful method to understand and analyze fuel performance during tests, plant operations and postulated accidents. Despite progress, significant modeling uncertainties remain in predicting ceramic-coated fuel failure and fission product release. The research plan includes activities to develop an independent state-of-the-art fuel performance analysis code for predicting fuel failure and fission product release in support the staff's HTGR licence application reviews.

A comparison of irradiation data has shown that U.S.-made fuel releases roughly a thousand times more fission products during irradiation than fuel made in Germany. Studies show that these performance differences arise from fuel fabrication processes rather than design or product specifications. The importance of manufacturing process was recognized in Germany and was included in the fuel manufacturing specification along with the product specifications. Activities are included in the research plan for the staff to develop the requisite knowledge and expertise of the critical manufacturing product and process parameters needed to ensure fuel quality and fuel performance.

To reduce research costs, leverage resources and obtain data in a timely manner, the staff has held discussions with the U.S. Department of Energy (DOE) and the European Commission, and the Massachusetts Institute of Technology (MIT) to develop cooperative agreements for HTGR fuel testing research and to develop an independent HTGR fuel performance analysis code. NRC discussions with the European Commission on fuel testing and analysis code development include potential access to European research and data on HTGR fuel fabrication. Such research and data would be expected to address, in large part, the NRC's needs in this area.

II. Materials Analysis

The NRC staff needs to develop independent research and expertise in the high temperature materials area for HTGRs to evaluate and establish a regulatory technical basis regarding the safety of advanced reactors. A sound technical understanding of the behavior of metallic and graphite components is needed to evaluate expected lifetime and failure modes of reactor components whose failure would result in loss of containment, loss of core geometry, loss of adequate cooling of the core, loss of reactivity control and shutdown systems, and/or air, water, or steam ingress.

The licensing approach for HTGRs relies heavily on the use of PRA. Since failure probability data for components of advanced reactors is not available from operating experience, very large uncertainties are inherent in the values selected and in the results of the PRAs. To reduce the uncertainties and render this approach viable, failure probabilities can be calculated from quantitative knowledge of the initiation and propagation of potential degradation (fatigue, creep, creep-fatigue, oxidation, thermal aging, stress corrosion cracking, crevice corrosion cracking, irradiation damage, and dimensional changes) of components in the operating environment of advanced reactors. Research is needed to fill the gaps, particularly in the high temperature coolant environment containing impurities. This research is also needed for NRC to evaluate the validity of the limited codes and standards available for the design of high temperature metallic components which were developed from data that did not fully reflect the operating environment and degradation mechanisms of HTGRs. The connecting pipe in HTGRs which carries hot helium from the core to the power conversion system is treated as a vessel. The consequence of this assumption is that a design basis double ended break is not considered for the connecting pipe and, therefore, no mitigating systems are incorporated in the design. Considering this pipe as a vessel raises questions since the pipe is much thinner and cracks can propagate through the wall in relatively short times. The results of the above research are also needed for NRC to evaluate the probability of failure for the connecting pipe and determine if a pipe break needs to be accommodated in the design. A break of this pipe could potentially lead to a large air ingress event and damage to the core. There is a similar lack for the design of graphite components, in areas of irradiation effects, oxidation, strength, and toughness. Research in these areas is needed for NRC to independently evaluate the structural integrity of graphite components and maintenance of core geometry during operating and accident conditions. Research is needed to address questions regarding the availability of existing codes and standards for metals and graphite, or using codes and standards that may not be appropriate to the operating environment of these advanced reactors.

In-service inspection (ISI) of important reactor components is a major aspect of defense-in-depth to ensure that unexpected degradation is identified and assessed in a timely manner. For some advanced reactors, the intervals between refueling outages are relatively long and many components may be inaccessible for inspection. Therefore, there is a need for NRC to evaluate the effectiveness of various ISI programs proposed for advanced reactors. If periodic ISIs are found to be ineffective for maintaining safety, licensees may need to utilize continuous online monitoring techniques for structural integrity and leakage detection.

Considerable research on high temperature materials for HTGRs of interest to NRC is ongoing and planned in the European Community (EC) and Japan. To leverage NRC resources and obtain data in a timely manner, the staff has visited facilities and met with members of the international community to initiate a dialogue on cooperation.

III. Reactor Systems Analysis

Because HTGR designs are substantially different from current LWRs, revised computer codes and new models will be needed to give NRC staff the necessary independent capabilities to realistically predict reactor system response. The development of a suite of validated reactor system analysis tools and data will permit the NRC staff to (a) conduct confirmatory analyses in the review of applicants' reactor safety analyses, (b) support development of the regulatory framework by assisting in the identification of safety-significant design basis and licensing basis events and associated success criteria, and (c) conduct exploratory analyses to better understand the technical issues, uncertainties, and safety margins associated with new designs.

In advanced HTGR designs, the integrity of the coated particle fuel in its function as primary fission product barrier depends strongly on the maximum fuel temperatures reached during irradiation and in accidents. These fuel temperatures are predicted by reactor system calculations using a combination of codes and models for core neutronics, decay heat power, and system thermal hydraulics. So-called melt-wire experiments performed in Germany's AVR (Arbeitsqem einschaft Versuchsreaktor) reactor showed the unexpected presence of in-core hot spots, where maximum local operating temperatures were much higher than predicted with codes like those now being used by the PBMR developers. Moreover, the AVR's true maximum local operating temperatures remain unknown due to measurement inadequacies in those experiments. For all advanced HTGR designs, significant uncertainties also exist in predicting the maximum fuel temperatures and vessel temperatures during heatup accidents. Such uncertainties relate to basic data like irradiation- and temperature-dependent thermal conductivities as well as the integral effects of variable local power densities with conductive, radiative, and convective heat transfer through the core and surrounding structures. Appropriate data measurements and system analysis tools will therefore be needed to support the staff's understanding and assessment of factors that govern fuel temperatures and uncertainties in relation to fuel integrity and HTGR safety margins.

Related research activities with analysis codes and data will also be needed for assessing the safety-related technical and policy issues associated with severe accidents and fission product release phenomena that differ dramatically from those in current and advanced LWRs. To meet research needs on all aspects of advanced reactor system analysis (i.e., nuclear analysis, thermal hydraulics, severe accidents and mechanistic release of fission products), the staff will seek to minimize costs and maximize benefits to the agency through active engagement in the planning and performance of domestic and international cooperative research. The staff has held discussions with the Massachusetts Institute of Technology (MIT) to develop cooperative agreements for HTGR on core neutronics and thermal hydraulic and heat transfer research, and will capitalize on past and ongoing international HTGR research projects that include Arbeitsqemgemeinschaft Versuchsreaktor (AVR) and the Thorium Hochtemperaturreaktor (THTR) in Germany, the High-Temperature Engineering Test Reactor (HTTR) in Japan, and the 10-MWe High-Temperature Reactor (HTR-10) in China.

IV. Probabilistic Risk Assessment

Industry proposes to use probabilistic risk assessment (PRA) in the design process for advanced reactors to a greater extent than current generation LWRs. The approach and criteria that lead to the selection of events to be considered in the design, for emergency planning purposes, and the accident source terms used in the analysis may be based, in large part, on the results of the PRA. For example, the approach that had been proposed by Exelon for the PBMR, which is likely to be adopted by the GT-MHR applicant, used probabilistic criteria to select events for consideration in conjunction with accident scenario specific source terms and dose limits to demonstrate the acceptability of the plant design.

The approach used by Exelon established three frequency categories for events and event scenarios (along with existing dose criteria for public protection for each category). Probabilistic risk assessment (with consideration of uncertainties) insights were used to determine which events (or event scenarios) would fall within each category. The three frequency categories for events are: (1) anticipated operational occurrences, (2) design basis events, and (3) emergency planning basis events.

Because advanced reactors are new designs, current PRA experience will need to be expanded to capture this new technology. The limitations of current PRA experience applies to system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems), to failure data, and most importantly, to the design, materials, systems, and safety features. Although extensive use will be made of the NRC-reviewed existing HTGR PRA, additional tools, expertise, and data (including information related to uncertainties) will be needed to enable evaluation of advanced reactor designs and associated PRA insights from a regulatory perspective. Technical issues are discussed further in Attachment 4, (e.g., data collection and analysis, initiating event identification and quantification, uncertainties).

Taken together, the outcome of research in the above four areas will be used to help establish the technical foundation needed to effectively review advanced HTGR license applications, support regulatory decisions, and establish resolution pathways for open issues.

Opportunities exist for developing much of the data and tools needed in assessing key technical issues through domestic and international cooperative agreements. In accordance with the Strategic Plan, the staff has taken the initiative to seek out such agreements, with due recognition of the Commission's role as an independent regulator. Success at establishing and shaping the desired agreements will hinge on the NRC's ability to commit to participating in the cooperative efforts. Therefore, the staff's research plan will highlight the activities and resource needs for leveraging domestic and international cooperation in the key technical areas for advanced reactors.

RES staff has already met with the ACRS and obtained comments and feedback on the advanced reactor research plan, its objectives, structure and scope. Additional meetings and detailed discussions with the ACRS and the Advisory Committee on Nuclear Waste (ACNW) on research associated with specific technical and safety issues for advanced reactors and their fuel cycles are scheduled to continue throughout FY 02. RES is working closely with the licensing offices to ensure that their needs are clearly identified and prioritized. The plan itself will be maintained as a living document to reflect in greater detail how the necessary information and capabilities will be developed through internal efforts and emerging cooperative agreements, and will be modified to accommodate any new issues and technologies not previously considered.

Attachments: As stated

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FUEL ANALYSIS AND TESTING

Introduction

Modular High-Temperature Gas-Cooled Reactors (HTGRs), such as the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine Modular Helium Reactor (GT-MHR) have unique safety features and design characteristics. Foremost among these features is the all-ceramic fuel element containing many thousands of tiny TRISO coated fuel particles (CFPs). The safety concept and intended design characteristic for the fuel is to contain and retain essentially all radiological fission products within the billions of CFPs contained in hundreds of thousands fuel elements (e.g., fuel pebbles, fuel compacts) which comprise an HTGR core. Effective retention of the fission products in the fuel is critical for all licensing basis conditions. These conditions include those associated with normal operation and operational transients, postulated design basis accidents and, accidents beyond the design basis that will require radiological consequence assessment for emergency planning. Accordingly, to assure public health and safety, TRISO fuel must be designed for very high integrity, manufactured to high quality standards with proven fabrication methods, and extensively tested for both operating conditions and postulated accident conditions that the fuel may experience.

It is also expected that potential modular HTGR license applicants will propose that the licensing source-term be based on models that mechanistically calculate fission product release from the fuel for various accident scenarios. This is different from the traditional licensing approach used for by LWRs, which involved a pre-determined and conservative deterministic accident source term. Further, license applicants are expected to propose the modular HTGRs be licensed to operate with a non-leak-tight "confinement" structure rather than a traditional leak-tight, pressure retaining containment structure. Accordingly, acceptability of the HTGR licensing and safety analysis basis will hinge, in large part, on the applicant's capability to demonstrate fuel fission product retention behavior under all licensing basis conditions and the NRC's capability to independently evaluate the safety margins and the uncertainties in these analyses.

The technical basis for the safety performance of HTGR fuels, including the design, analyses, manufacture, irradiation testing, accident simulation testing and, in-reactor operations, has made significant progress over the last four decades. In this regard, considerable time, effort and resources have been expended to fully understand and predict fuel behavior and fuel safety performance. Several countries implemented fuel development and qualification programs with the coated particle as the basic unit. Cumulatively, these efforts have addressed many of the elements which can effect TRISO coated particle fuel performance. NRC's research in the area of fuel performance analysis should capitalize on this body of knowledge to establish the infrastructure of knowledge, data, and tools needed for HTGR reviews.

A significant number of unirradiated German archive fuel elements, that were fabricated for use in the AVR, are currently in storage at the Jülich Research Center. This fuel is stated to be of the reference design and manufacture for the PBMR pebble fuel, but of higher enrichment.

Additionally, GT-MHR fuel particles are expected to be fabricated using processes that are equivalent to the German particle coating process. A number of these archive elements are available to NRC and other third parties for use in fuel irradiation testing programs.

Attachment 1

The fuel analysis research plan focuses on establishing the staff's requisite level of knowledge, analytical tools and data for modular HTGR fuels. The areas addressed are fuel design, manufacture, analysis, operational performance and accident performance. The objective of the planned research, is to establish the capability to independently assess the technical and safety basis for an applicant's fuel, and support the development of NRC regulatory requirements and inspection program, as needed. Tools will be developed and validated to independently predict fuel performance (including coated particle fuel [CPF] failure rates and fission product releases) during normal operation, design-basis accidents and potential severe accidents. Research will also be conducted to support the staff's review of the mechanistic source term predictions for normal operation, design-basis accidents and accidents beyond the design basis. This planned research directly supports staff activities related to: ensuring fuel quality over the life of the plant; fuel qualification testing; licensing source term determinations and assessment of confinement versus containment.

Discussion

Although considerable time, effort and resources have been expended world-wide to fully understand and predict HTGR fuel behavior and safety performance, significant issues remain. Many of these issues will need to be addressed early on in order to support an effective and efficient HTGR licensing process at a future date. Discussed below are the major issues associated with HTGR fuel performance and qualification along with plans for research to address these issues.

Fuel Irradiation Testing

Operating (irradiation) conditions that are known to effect fuel (particle) performance include: air/water ingress, operating temperature, burnup, fast fluence and power. Extensive international irradiation (operational) testing has been conducted in Germany, UK, USA, Japan, Russia, Netherlands and China to understand and characterize the behavior and safety performance of TRISO coated particle fuel during normal operation and operational transients. The fuel development and qualification irradiation testing philosophy of the international programs has generally focused on demonstrating acceptable and predictable fuel performance within the design operating envelope applicable to a specific HTGR design. Far less research has been conducted to explore operational conditions beyond these design envelopes to understand and to know the conditions wherein large increases in particle failure rates can occur. The limited available data at higher operating temperatures, burnups, fast fluence and power results in significant uncertainties surrounding the failure margins for TRISO particle fuel. Additionally, to obtain test results in a short time, test reactor irradiations are accelerated to achieve burnups three to ten times faster than would be the case in an actual reactor core. Although the higher power may conservatively bound thermal-mechanical failure mechanisms, chemical interaction failure mechanism may not be bounded.

NRC fuel irradiation testing to obtain this data is included in the advanced reactor research plan. The purpose of the irradiation testing is to: collect data to more firmly establish the safety margins for TRISO fuel particle failure and fission product release behavior during normal operation and operational transients; provide data to verify an applicant's fuel failure and fission product release calculations; provide improved knowledge and insights to assess the acceptability of an applicant's fuel irradiation test program (e.g., test methods, QA program, data analysis methods), and; provide data for use in developing/validating NRC analytical models and methods. Fuel irradiation testing is a research area which requires as many as 3 to 5 years to complete planning and preparation, in-reactor irradiation testing, data analysis, post-irradiation examination and analysis and documentation. Fuel irradiation testing data also provides a key basis for NRC cooperative research agreements to provide NRC access to fuel testing programs funded by others. For NRC sponsored fuel irradiation tests to be most beneficial and timely, testing commitments must be in place as many as 6 years before an HTGR application would be expected.

The U.S. DOE currently has plans to conduct an HTGR fuel irradiation test program. The program is part of DOE's HTGR fuel development and qualification program. Testing involves both German archive pebble fuel, in the near- term, and PBMR and GT-MHR production fuel, in the longer term. The tests are intended to establish a benchmark and validate the performance of the German fuel for the more demanding design conditions (e.g., higher operating temperature) of a PBMR compared to the earlier German pebble bed reactor designs. The test program also provides opportunities to explore the margins to failure of HTGR fuels for operating conditions that are well beyond the conditions associated with the fuel design basis. To reduce costs, leverage facilities and obtain the needed information in a timely manner, the NRC staff has been working with the US DOE to develop an NRC/DOE cooperative research agreement for HTGR fuel irradiation testing. The cooperative testing would address, in part, the NRC's irradiation test data needs in this area.

The European Community has established a fuel technology project (HTR-F) within the EC's High Temperature Reactor Technology Network (HTR-TN). HTR-F includes near-term plans for irradiation testing of German, GA and Chinese fuel elements with TRISO particle fuel designs. The irradiations will involve very high burn-up in order to understand the failure and fission product release behavior of TRISO fuels at very high burnup. The planned burnups are far beyond what would be envisioned for a near-term PBMR or GT-MHR license application. Therefore, the EC irradiation tests will provide additional opportunities to explore the margins to failure of HTGR fuels for test conditions that are well beyond the conditions associated with the fuel design basis. The NRC staff has been working with the HTR-TN and HTR-F to develop a basis for cooperative research for HTGR fuel irradiation testing which would address in part the NRC's irradiation testing needs in this area.

<u>Fuel Accident Simulation Testing</u>

A major HTGR safety issue is fuel particle integrity and fission product release during a postulated accident. A key factor effecting fuel (particle) performance during an accident is the peak temperature in the particle(s) reached during the accident (following the prior degrading effects of the operating conditions). Temperature increases can occur due to heatup accidents, such as caused by a loss of normal cooling, a core power increase, or a significant increase in core reactivity. Other accidents of concern that can effect fuel performance include chemical attack (e.g., oxidation) of the fuel element and CFPs or a large prompt reactivity pulse. Fission product release from the fuel during an accident is a key input to the mechanistic source term calculation.

NRC research is needed to understand and quantify the safety margins to increased rates of CFP failure and increased fission product release that can occur during accidents. Past accident simulation fuel testing has generally emphasized testing up to the so-called 1600°C maximum allowable accident temperature or a few hundred degrees above. Far less research has been conducted to explore fuel performance involving accident temperatures well above the so-called 1600°C limit, especially for fuel that has been irradiated at operating conditions which significantly exceed the design envelope. It is also expected that HTGR applicants will request credit for on line coolant activity monitoring as a means of assuring the in-reactor condition of the fuel. However, little, if any fuel testing, has been conducted to show whether degradation of fuel due to either fabrication process anomalies or extended operation outside the design operating limits (which might significantly and systematically elevate particle failure rates during an accident) can be detected through coolant activity monitoring. Also, there is some evidence that traditional accident heatup testing involving a ramp up in temperature and hold constant at the maximum accident temperature may not be conservative relative to the actual accident temperature versus time profiles for determining particle performance.

NRC research to address these issues is included in the advanced reactor research plan. The purpose of the accident simulation testing is to: provide the data needed to verify an applicant's claims of fuel performance and fission product release during accidents; provide the data which explores the limits (i.e., safety margins) of fuel performance and fission product release for fuel conditions which are important to the margins such as prior operating conditions, peak accident temperature and temperature history, fuel oxidization, reactivity induced energy deposition and deposition rate in the fuel (due to reactivity accidents); provide knowledge and insights needed to judge the acceptability of an applicant's fuel irradiation test program (e.g., test methods, QA program, data analysis methods) and; provide data for developing and validating NRC analytical models and methods.

The U.S. DOE and EU HTR-F HTGR fuel irradiation test programs also include accident condition testing. For certain planned tests, the accident simulation conditions (e.g. fuel burnup when the accident occurs) are beyond what would be expected for a near-term PBMR or GT-MHR license application. Therefore these tests will also provide additional opportunities to explore the margins to failure of HTGR fuels for test conditions that are well beyond the conditions associated with the fuel design basis. The Japanese Atomic Energy Research Institute (JAERI) also plans to conduct prompt reactivity accident simulation testing of TRISO particle fuel at a test reactor in Japan. The NRC has been working with the JAERI to develop a basis for cooperative research for HTGR technology including fuel accident condition testing. These tests would address, in part, the NRC's accident simulation testing needs.

Analytical Codes and Methods

Fuel behavior analytical codes and methods provide a powerful tool to simulate fuel particle

behavior and fission product release for reactor operating conditions and accident conditions that are different from the fuel tests. Such codes and methods endeavor to model the various particle failure and fission product mechanisms. Failure mechanisms include: internal over-pressure and tensile stress failure of the SiC layer; chemical attack of the dense coating layers due to migration of the fuel kernel; thermal dissociation and failure of the SiC layer at very high particle temperature; SiC degradation and failure due to chemical interaction of fission products with the SiC layer and; overstress of the SiC layer due to external asymmetric loading by the particle layers. Fuel codes and methods also have been developed to predict fission product transport and release mechanisms during normal operations and postulated accident conditions. Models and methods also provide a useful tool to understand fuel behavior and analyze fuel performance for specific tests or reactor operations and potential accident conditions.
The analytical models used by fuel designers, developers and applicants to predict fuel performance and fission product release have made significant progress over the past 20 years. These models have been used by fuel designer to help quantify margins and by safety analysts to calculate the mechanistic source terms. Despite the progress, significant uncertainties remain in modeling fuel particle behavior, particle coating material properties, and particle failure, transport and release mechanisms.

Research activities are included in the advanced reactor research plan to support the development of a state-of-the-art fuel performance analysis code for predicting TRISO fuel particle failures and fission product release. NRC research activities in this area will address the current modeling issues. The tools developed would be benchmarked against existing empirical CFP fuel performance data, other codes and the results of NRC and applicant/vendor fuel performance and qualification test data The purpose of the fuel analytical codes and methods development activities is to provide the NRC staff with an independent analytical capability to evaluate HTGR fuel behavior. Planned uses of the tool would include: analysis of CFP failure, fission product release, and safety margins; analysis of the effects of variations in fabrication, irradiation and accident conditions (i.e., sensitivity studies); analysis of source term for NRC analysis of radiological consequences and; potential analysis of unexpected anomalous fuel behavior and planned corrective actions for HTGR plants licensed to operate in the future. Sensitivity calculations would be conducted to assess the effects of variations and uncertainties in fuel characteristics and reactor core conditions which may not be fully simulated in the fuel irradiation testing programs.

The INEEL , EU HTR-F and MIT are currently developing codes and methods to predict TRISO particle fuel behavior and HTGR fuel fission product release. The NRC has been in contact with these organizations with the objective of establishing a basis for cooperative research aimed at developing codes and methods for HTGR fuel analysis. Such codes and methods are expected to address, in part, the NRC's needs in this area.

Fuel Fabrication Knowledge

A comparison of the irradiation data for German made TRISO coated particle fuel with U.S. made TRISO coated particle fuel shows that the gas release rate (i.e., particle failure rates) during irradiation for U.S. fuel three orders of magnitude higher than the German fuel. A recent INEEL study of causes for these differences confirmed view that differences in the process parameters used for applying the individual coating layers of the TRISO coated particle was a dominant factor in the irradiation performance and accident condition performance. Although, the U.S. fuel met the established specifications for the measurable fuel particle layer physical, material and chemical characteristics (e.g., thickness, density, strength, impurities) which were consistent with the design and safety requirements and equivalent to the German, differences in processes were found to result in critical differences in layer micro structures, bonding between layers and layer anisotropy, etc. which affected critical differences in in-reactor (irradiation and accident) behavior. The importance of the manufacturing process was recognized in Germany and was included in the fuel manufacturing specification along with the product specifications.

The GT-MHR and PBMR fuel suppliers have stated that the manufacture of the TRISO fuel particles for their fuel be equivalent to the German fuel manufacture. To achieve this goal both fuel suppliers are currently implementing a fuel manufacturing development program with the assistance of available German fuel fabrication technology specialists. A major goal of these efforts is to develop the equipment, procedures, process specifications and product specifications which will re-establish the German production fuel quality and characteristics and hence the re-establish the German fuel irradiation performance. China and Japan implemented similar programs for the manufacture of fuel for the HTR-10 and the HTTR, respectively.

The regulatory oversight measures to ensure the requisite fuel quality of the fuel (supply) on a consistent basis over the lifetime of an MHTGR plant is a significant safety issue and a potential Commission policy issue. Such measures might include fuel fabrication technical specifications and fuel fabrication facility inspections. Other additional or alternative measures might involve reactor coolant activity monitoring and periodic fuel accident simulation testing using production fuel. These alternatives can have significant technical and regulatory advantages and disadvantages, however. An additional issue relates to how much fuel testing needs to be completed before a plant can be licensed, which links to whether the correct and complete fuel manufacturing process and product specifications have been identified and specified.

Research activities are included in the advanced reactor research plan to support establishment of staff knowledge and expertise of the critical manufacturing product and process parameters needed to ensure fuel quality and fuel performance. A major research element of the EU HTR-F is to re-establish know-how on TRISO coated particle fuel fabrication. The research includes both fabrication of fuel kernels and coatings. The NRC has been in contact with HTR-F with the objective of establishing a basis for cooperative research agreement involving fuel irradiation testing and fuel analysis code development. The cooperative agreement would include NRC access to the HTR-F research and data on fuel fabrication. Such research and data would be expected to address, in large part, the NRC's needs in this area.

<u>Fuel Technical Expertise and Information</u>

The current interest in modular HTGRs for electric power generation has resulted in a significant increase in design, development and research activities in the area HTGR fuels. As noted above, fuel research activity areas include, manufacture, irradiation testing, accident condition testing and code development. Other research efforts include collection, compilation, cataloging and electronic databases for historical fuel performance and test data, and the analysis of these data. An example is the recent INEEL analysis of the causes for the differences between German and U.S. fuel performance. IAEA is developing a new Coordinated Research Project (CRP-6) for HTGR fuel technology to coordinate member state HTGR fuel research activities in these and other areas. International conferences, workshops and specialized training addressing HTGR fuel behavior and performance factors now being renewed after several years of dormancy. Countries with recently licensed HTGR's such as Japan (HTTR) and China (HTR-T) have invested significantly in research and development in the areas of HTGR fuel design, analysis, irradiation testing, accident condition testing and manufacturing. These countries have recently expressed their willingness to share their knowledge in selected areas through information exchanges within the context of cooperative agreements that are being expanded to explicitly include the range of HTGR technologies relevant to the NRC's safety mission. Research activities are included in the advanced reactor research plan to interact with these sources as a means to efficiently and effectively establish NRC staff expertise and information in the area of HTGR fuels.

MATERIALS ANALYSIS

Introduction

The behavior of metallic and graphite components under normal operating and accident conditions is a key research area important to the safety of HTGR plants. The NRC staff needs to develop independent research and expertise in the high temperature materials area for HTGRs to evaluate and establish a regulatory technical basis regarding the safety of advanced reactors. A sound technical understanding of the behavior of metallic and graphite components is needed to evaluate expected lifetime and failure modes of reactor pressure vessel materials and other components whose failure would result in loss of containment, loss of core geometry, loss of adequate cooling of the core, loss of reactivity control and shutdown systems, and/or air, water, or steam ingress.

There is a general lack of data and codes and standards for design of high temperature metallic components particularly for the operating environment and potential degradation mechanisms of HTGRs. There is a similar lack for the design of graphite components in areas of irradiation effects, oxidation, strength, and toughness. There is a question about whether advanced reactors can be appropriately designed with the existing lack of codes and standards or using the codes and standards not appropriate to the operating environment of the reactors.

Since failure probability data for components of advanced reactors is not available from operating experience, very large uncertainties are inherent in the values selected and in the results of the Probabilistic Risk Assessments (PRAs) used to independently confirm design and support safety evaluations. To reduce the uncertainties, information on failure probabilities can be derived from research results on the behavior of high temperature materials. Failure probabilities can be calculated with knowledge of potential degradation mechanisms (fatigue, creep, creep-fatigue, oxidation, thermal aging, stress corrosion cracking, crevice corrosion cracking, irradiation damage, and dimensional changes) of components in the operating environment of advanced reactors and with quantitative information of the initiation times and growth rates. Research is needed because this information is not available, particularly in the impurity environments of these reactors.

In-service inspection (ISI) of important reactor components, which is generally conducted during refueling outages, is a major aspect of defense-in-depth to ensure that unexpected degradation is identified and assessed. For some advanced reactors, the intervals between refueling outages are relatively long and many components may be inaccessible. Therefore, continuous online monitoring techniques for structural integrity and leakage detection may be necessary for advanced reactors.

In HTGR designs the connecting pipe which carries hot helium from the core to the power conversion system (PCS) is treated as a vessel because this pipe is designed, fabricated, and inspected to the same rules as a reactor pressure vessel. The consequence of this assumption is that a design basis double ended break is not considered for the connecting pipe and therefore, no mitigating systems are incorporated in the design. Considering this pipe as a vessel is not realistic because the pipe is of much smaller diameter and therefore, much thinner wall than a reactor pressure vessel designed to the same working pressure. If an unexpected degradation mechanism should initiate in the pipe, because of the thin wall, it can propagate through the wall in a relatively short time and possibly not be detected by ISI. Conversely, if an unexpected degradation mechanism were to initiate in a pressure vessel, it would require longtimes to propagate through the greater wall thickness, allowing enough time to be detected by ISI.

Considerable research on high temperature materials for HTGRs of interest to NRC is ongoing and planned in the European Community (EC) and Japan. To benefit from this research and establish cooperative research agreements, the NRC staff has visited facilities and met with members of the international community. Cooperative research agreements will be established which involve exchange of research results and not funds. It is necessary for NRC to contribute its own research results to gain access to extensive research on high temperature materials taking place in Europe and Japan.

Discussion

Discussed below are the major technical issues in the materials area along with research needs and plans to address these issues. There is a general lack of national codes and standards for design of high temperature components for HTGRs. The technical bases needed for developing appropriate codes and standards should focus on the integrity of components at high temperatures and stresses over extended operating times taking into account irradiation and the actual helium environment with low levels of air, water, oxygen, and other impurities. Some current high temperature codes and standards were based on LMFBR data of the 70s and 80s. Subsequently, better correlations for creep and fatigue-creep interactions from data of the 90s were developed and need to be incorporated. Updated codes and standards also need to take into account the effects of irradiation and of helium coolant with impurities (particularly oxygen) on the reduction in strength, fatigue and creep life, and SCC resistance of components.

There is a lack of material specifications for nuclear grade graphite. Designers of HTGRs use measured properties of the particular graphite in their design calculations. However, nuclear graphites should meet certain minimum requirements with respect to life-limiting properties such as strength, density, and thermal conductivity. Specific impurities, such as halogens in graphite can be released during operation and cause cracking of other reactor components and therefore, they should be limited in nuclear graphites. There is a need to develop standards for nuclear grade graphite to address minimum requirements with respect to composition, impurities, and physical and mechanical properties. The NRC/RES has begun to co-sponsor, through ORNL, development of an ASTM standard for nuclear grade graphite.

Appropriate data bases for fatigue, creep, creep-fatigue, and stress corrosion cracking are needed for component life calculations using design codes and standards. Based on past experience and research with high purity water coolants in LWRs, we have found that low levels of impurities, such as less than one part per million of oxygen, can reduce fatigue lives and enhance stress corrosion cracking of metallic components. These effects were not originally addressed in the ASME Code. For example, the design data for fatigue was obtained from materials tests in air. Because helium is inert there has been a tendency to obtain design data in pure helium; in impure helium, but not all impurities included; or in air. The effects of all important impurities, such as oxygen, in high temperature helium need to be taken into account with respect to reductions in fatigue and creep lives, stress corrosion crack initiation and growth rate, crevice corrosion crack initiation and growth rate, and cyclic crack growth rate. Such data and understanding need to be developed. Degradation by mechanisms discussed above of power conversion system components such as turbine blades leading to catastrophic failure could compromise the primary circuit pressure boundary in HTGRs leading to air and water ingress with corresponding degradation of fuel and graphite components. Evaluation of potential degradation mechanisms and rate of progression for materials used for connecting piping between the reactor pressure vessel and the power conversion systems will provide the NRC an independent basis to determine the validity of the contention that pipe break analysis does not need to be evaluated. NRC/RES has initiated an effort at ANL to review and evaluate applicability of current codes and standards for design of HTGRs and to review available research on the effect of impurities in the helium environment on degradation of high temperature metals.

The operating gas coolant and particulate environment can lead to degradation by carburization, decarburization, and oxidation of metals in HTGRs. Carburization of carbon steels and stainless steels can lead to cracking, stress corrosion cracking, and oxidation of the HTGR components. Whereas, decarburization results in a softer steel and in reduced fatigue and creep lives. The presence of oxygen results in the formation of scale and general corrosion of metallic components and more importantly it can oxidize the graphite and render metallic components susceptible to stress corrosion cracking. To control the phenomena of carburization, decarburization and oxidation, a very careful control of the level of different impurities is required. Conditions that lead to avoidance of one of the above phenomena can lead to development of another. Some research has been conducted to study the phenomena described above; however, NRC needs to conduct confirmatory research and better define the conditions under which the phenomena occur for important components of HTGRs.

Aging and sensitization behavior of alloys during elevated temperature exposures can lead to embrittlement and susceptibility to intergranular stress corrosion cracking of alloys. Aging and embrittlement occurs, for example, in cast stainless steel components under temperatures and time conditions experienced in operating LWRs. At the operating temperatures of HTGRs, the aging and sensitization reaction rates can be even higher. These reactions need to be studied for the alloys and conditions of HTGRs to establish the potential for material property degradation and embrittlement during the lifetime of these reactors.

A potential issue related to the inspection of some advanced reactors is the long intervals and limited amount of inspection that may be conducted between scheduled short-duration shut-downs. Therefore, there is a need to evaluate the effectiveness of various in-service inspection (ISI) programs as a function of frequency of inspection and the number and types of components inspected including consideration of those internal components that are not accessible for inspection. If periodic ISIs are not adequate to maintain safety, an alternative is to conduct continuous on-line, nondestructive monitoring for structural integrity and leakage detection of the entire reactor or of specific components during operation. Continuous monitoring has been developed for LWR applications, however these techniques need to be evaluated and validated for the materials, environments, and degradation mechanisms of the

HTGRs.

To be able to effectively review the new HTGR designs, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behavior of nuclear-grade graphite under the temperatures, radiation, and environments expected during normal operating and accident conditions. Potential loss of strength and of resistance to fatigue and creep, shrinkage, swelling, cracking, and corrosion during operation could impact the performance and function of the graphite core structural elements and reflectors. Coke source, manufacturing processes, and microstructures determine the virgin and irradiated properties of the graphite component.

Irradiation affects, and in many cases, degrades physical and mechanical properties of the graphite. Important properties that change with irradiation are thermal conductivity, strength, and dimensions. During operation, thermal gradients and irradiation induced dimensional and strength changes result in significant component stresses, and distortion and bowing of components. These can lead to loss of structural integrity, loss of core geometry, and potential problems with insertion of control rods. At high doses of irradiation graphite structures will start to disintegrate and experience total loss of integrity. Key issues for the application of graphite in HTGRs are the lack of data on irradiated properties of current graphites, lack of data at high irradiation doses, and lack of correlations between as-received graphite properties and post-irradiation properties. Irradiated graphite properties are heavily dependent on its particular make-up and the manufacturing process; even small variations in these may have strong effects on the properties. Therefore, the irradiated materials properties from the "old graphites" can not be assumed to be the same as the "new graphites", and data needs to be developed for the new graphites up to high irradiation doses. To obtain an adequate irradiation data base for each graphite used is difficult, expensive and time consuming. Therefore reactor designers propose to use radiation data from studies conducted on older graphites. Since the exact raw materials and processes have changed and may continue to change in the future, the NRC will need to independently confirm the claims that a particular graphite will behave the same as the old graphites under operating irradiation conditions. To accomplish this, without irradiation testing every time a change occurs in the graphite raw materials or processing, correlations are needed for predicting irradiated graphite properties from the virgin graphite raw materials characteristics, composition, processing and properties.

Graphite corrosion and oxidation can occur in HTGRs from oxidizing impurities in the helium coolant from in-leakage during normal operation, or from air or water ingress during accidents. The oxidation of graphite from exposure to air is an exothermic reaction and it is important to know the rate of heat generation particularly during accidents. Oxidation also will remove the surface layers of graphite components resulting in loss of structural integrity. Further, oxidation will change the thermal conductivity, and reduce the fracture toughness and strength of graphite components. The oxidation rates vary for different graphites, and can be greatly affected by the impurities in the original graphite. Therefore oxidation rate data is needed for the graphites proposed for new reactors.

Issues and research needed for high temperature materials analyses are discussed in the draft

Advanced Reactors Research Plan. Descriptions of this research have been shared with the international community, in particular with Japan and the European Commission (EC). NRC staff has met with technical staff and officials of the EC to discuss potential cooperation. The EC has agreed with the importance and need for the research outlined in our plan and welcomes the NRC to participate in their high temperature materials research (HTR-M) program. Participation is through the exchange of research results from the parties' research programs. Much of the work described in the research plan will be addressed in the EC current program and their future program to initiate in 2003. Some of the key work possibly not fully addressed in the EC programs is in the areas of a) effects of the helium environment with impurities on degradation of materials, b) evaluation of in-service inspection programs and monitoring, and c) correlations of virgin graphite properties and manufacturing parameters to post-irradiation graphite properties. Exchange of NRC research results in these areas could be used for cooperation with the EC HTR-M programs. We would plan to exchange NRC research results in these areas with the EC HTR-M program which will address much of the research outlined in our Advanced Reactors Research Plan in the Materials Analysis area.

REACTOR/SYSTEM ANALYSIS

Introduction

In advanced HTGR designs, the integrity of the coated particle fuel in its function as primary fission product barrier depends strongly on the maximum fuel temperatures reached during irradiation and in accidents. These fuel temperatures are predicted by reactor system calculations using a combination of codes and models for core neutronics, decay heat power, and system thermal hydraulics. So-called melt-wire experiments performed in Germany's AVR reactor showed the unexpected presence of in-core hot spots, where maximum local operating temperatures were much higher than predicted with codes like those now being used by the PBMR developers. Moreover, the AVR's true maximum local operating temperatures remain unknown due to measurement inadequacies in those experiments. For all advanced HTGR designs, significant uncertainties also exist in predicting the maximum fuel temperatures and vessel temperatures during heatup accidents. Such uncertainties relate to basic data like irradiation- and temperature-dependent thermal conductivities as well as the integral effects of variable local power densities with radiative, conductive, and convective heat transfer through the core and surrounding structures. Appropriate data measurements and system analysis tools will therefore be needed to support the staff's understanding and assessment of factors that govern fuel temperatures and uncertainties in relation to fuel integrity and HTGR safety margins.

The following subsections (reactor/system analysis including thermal hydraulic analysis, nuclear analysis, and severe accident and source term analysis) describe the major issues and the research needs and plans to address these issues.

(A) Thermal-Hydraulic Analysis

For HTGR safety analysis, a code will be needed to reliably and efficiently predict cooling transients that evolve over time scales of days, not hours as we have become accustomed to in LWR analyses. Some design basis transients are driven by radiative and conductive heat transfer through porous and solid structures, rather than by fluid convection, and this capability, although it currently exists in all codes, will have to be extended to three dimensions. Appropriate fuel element model geometries (i.e., prismatic or spherical) will have to be developed for analyzing HTGR transients. The NRC analysis tools (TRAC-M) should be able to model the turbo-machinery and passive decay heat removal systems, and accurately model gases (i.e., helium or helium with air or steam) in natural and forced circulation. Computational fluid dynamics codes (e.g., FLUENT) may also be needed for HTGR analysis. The capability to model graphite as a solid structure will also have to be added. Experimental data will also be needed to evaluate the accuracy of codes and assess margins of safety. An effort to modify TRAC-M to add analysis capabilities for HTGR is being initiated at Los Alamos National Laboratory (LANL). An existing code, GRSAC (gas reactor severe accident code), will also be used to support TRAC-M thermal-hydraulics code development and assessment efforts. For HTGR, test data will be sought from past and ongoing foreign HTGR research projects such as the Arbeitsgemeinschaft Versuchsreaktor (AVR) and the Thorium Hochtemperaturreaktor (THTR) in Germany, the High-Temperature Engineering Test Reactor (HTTR) in Japan, and the 10-MWe High-Temperature Reactor (HTR-10) in China.

(B) Nuclear Analysis

Nuclear analysis encompasses the analysis of: (a) fission reactor neutronics, both static and dynamic, (b) nuclide generation and depletion as applied to reactor neutronics and to the prediction of decay heat generation, fixed radiation sources, and radionuclide inventories potentially available for release, (c) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, and radiation protection, and (d) nuclear criticality safety (i.e., the prevention and mitigation of critical fission chain reactions outside reactors).

Of fundamental importance to all work in this area is the updating of the nuclear data libraries in use today, which were developed in the 1980s or earlier. Such data libraries play an essential role in all applications of nuclear analysis. While these legacy nuclear data libraries have proven largely adequate in a variety of applications, they have known limitations and shortcomings that will be addressed by the updating process and that would otherwise need to be reevaluated in the context of advanced reactors and their fuel cycles.

The unique features of HTGRs include their use of fission-product retaining coated fuel particles, graphite as the moderator and structural material, and neutronically inert helium as the coolant. The PBMR and GT-MHR are modular HTGR designs that are fueled with uranium enriched to 8% and 19.9%, respectively. Both HTGR designs also have long annular core geometries and locate control and shutdown absorbers in the graphite reflector regions. In many respects, the PBMR and GT-MHR designs therefore have similar code modeling and validation issues for the prediction of reactor neutronics phenomena and decay heat generation.

Physics analysis issues unique to the GT-MHR relate mainly to the effects of burnable poisons, the presence of "fissile" and "fertile" coated fuel particles (with 19.9% enriched and natural uranium, respectively) in the fuel compacts, reactivity control for cycle burnup effects, and the power shaping effects of zoned fuel and poison loadings. Nuclear analysis issues anticipated in evaluating HTGR reactor safety include (i) temperature coefficients of reactivity of fuel, moderator graphite, central graphite region, and outer reflector graphite, (ii) reactivity control and shutdown absorbers worth, (iii) moisture ingress reactivity for depressurized or underpressurized accident conditions, (iv) reactivity transients caused by credible events such as overcooling, control rod ejection, rod bank withdrawal, shutdown system withdrawal or ejection, seismic pebble-bed compaction, and moisture ingress, (v) graphite annealing heat sources that may be significant enough to add to the nuclear decay heat sources used in the analysis of loss-of-cooling heatup events, and (vi) for PBMR, pebble burnup measurements and discharge criteria, hot spots, and analytical treatments of the quasi-random local mixing of pebbles with different burnups, fission powers and decay heat powers.

For PBMR and GT-MHR, NRC is reviewing RES in-house analysis and contractor projects conducted in the late 1980s and early 1990s in supporting the staff's preapplication safety evaluation of the DOE MHTGR. RES has initiated projects for (i) modular HTGR accident analysis, (ii) PARCS code modifications to incorporate the R-Theta-Z geometry needed for

PBMR analysis, (iii) preparation of modern nuclear data libraries, and (iv) exploratory and scoping studies for core neutronics and decay heat analysis in PBMR and GT-MHR. In parallel, NRC is exploring opportunities for HTGR-related domestic and international cooperation. NRC is considering participation in the research at MIT that includes sharing of pebble-bed reactor physics codes, models, related code development, and analysis tasks, and acquiring existing HTGR physics benchmark and test data from Fort Saint Vrain testing and operations, the HTR-PROTEUS critical experiments at PSI, the ASTRA critical experiments at the Kurchatov Institute, and the earlier CNPS critical experiments at LANL. NRC is also planning to participate on activities related to safety performance of HTGRs sponsored by the IAEA, the European Commission, and the OECD/NEA's Nuclear Science and Nuclear Safety programs, as well as where possible and appropriate, acquiring HTGR physics benchmark data from Switzerland, Russia, U.K., France, Japan, China and Germany.

(C) Severe Accident and Source Term Analysis

For today's LWRs, severe accident analysis methods using codes such as MELCOR have been developed to estimate the magnitude and timing of fission product release to the containment and subsequently to the environment. Accident and source term analysis will likewise be needed for advanced reactors to support the development of limiting sequences and to confirm applicants' analysis of the plants.

For HTGRs, both the types of sequences and the process by which fission products may be released from HTGR fuel will be different. As a result of diffusion during normal operation, rupture of coated fuel particles as a result of accidents, and vaporization during high-temperature degradation of the fuel, fission products may be released. The MELCOR code has most of the capabilities needed to analyze beyond design-basis accident issues for HTGRs, but models need to be added or modified for HTGR applications. In addition, the need for additional fission product deposition/transport experiments will be examined. For HTGRs, NRC has initiated (i) a review of past experiments and studies performed, (ii) MELCOR code development and assessment for HTGRs, including the use of GRSAC to support the development and assessment effort, and (iii) a TRISO Fuel Particle Phenomena Identification and Ranking Table (PIRT). Past experiments included those performed at the Federal Republic of Germany (FRG), and at the Japanese Atomic Energy Research Institute (JAERI), and numerous International Atomic Energy Agency studies. The European Commission (EC) is currently sponsoring approximately a $16M, 4-year research program on high temperature gas cooled (HTGR) reactors, a portion of which is related to reactor system anlysis. NRC is considering to participate in the EC research.

PROBABILISTIC RISK ASSESSMENT

Introduction

Future licensees have indicated that PRAs will be an integral part of their applications and NRC expects to play a crucial role in the licensing process for new reactor designs. Therefore, the NRC should be prepared with the tools and expertise to perform an independent review of the PRAs submitted as part of the licensing applications.

The NRC has promoted PRA use as a means of developing nuclear power plant risk perspectives and identifying improvements. As a result, the NRC has developed the capability to use PRAs in regulatory decision-making for current generation reactors. This capability is founded on the staff's in-depth understanding of the techniques and data employed in a PRA, the design and physical characteristics of the reactors modeled, and how the design and characteristics are modeled in a PRA in terms of underlying hypotheses and data.

However, advanced reactors are new designs and, therefore, the current PRA experience will need to be expanded to capture the new technology. The limitations of current PRA experience applies to system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems), to failure data, and most importantly, to the design, materials, systems, and safety approach. Extensive use will be made of the NRC-reviewed existing HTGR PRA. The tools, expertise, and data (including information related to uncertainties) will be developed to enable the staff to evaluate advanced reactor PRAs.

The PRA interfaces with virtually every other technical area. Given that PRA is an iterative process, knowledge of reactor systems, fuels, materials, human performance, and digital instrumentation and controls (I&C) will be used for postulating accident initiators, modeling of systems, and quantifying accident sequences. The results will indicate what issues are important from a probabilistic perspective and need investigation.

Discussion

The following are the technical issues that will need to be addressed as part of the PRA development plans.

Initiating event identification and quantification

The events that challenge the current generation of LWRs may not be applicable to advanced reactors. It is necessary to understanding what events can occur (as a result of design characteristics, equipment failures, and human errors) that challenge the plant operation comprises the first step in assessing the risk associated with a given reactor design. Extensive use of existing HTGR information and the PRA will be used, as appropriate. This quantification will provide the necessary initial data on initiating event frequencies for use in the PRA.

Accident progression and containment performance (including source term)

Success criteria, accident progression, and source terms for advanced reactors are likely to be different from those for LWRs. The accident progression for different advanced reactor designs needs to be understood. A probabilistic containment analysis (Level 2 PRA) is needed to assess the ability of a reactor containment or confinement with a filtered venting system to provide adequate protection against release of fission products. The knowledge of fuel performance is a prerequisite to performing an independent review of the PRA. We need to understand the quantification of core behavior during accidents such as overheating or immersion in media other than helium (in air or, if possible, in water) for inclusion in the PRA. This behavior should be understood not only for fresh fuel but also for end of life fuel to evaluate the impact, if any, of burn-up. A single PRA tool should be developed which incorporates, in a fully integrated and automated manner, what has historically been considered Level 1 (CDF), Level 2, and Level 3 (source terms and consequences) PRAs.

System modeling

The probabilities and failure modes of passive systems and advanced digital I&C systems need to be determined for incorporation into the PRA. Passive systems have been treated in PRAs as either initiators or complete failures. As a result, current PRAs model only the performance of active systems using a binary logic which is suitable for such purposes. It is not clear that this approach would be suitable for modeling passive systems exhibiting slow evolutionary behavior during accidents which could exhibit degradation or intermediate failure states. Therefore, the modeling approach needs to be reconsidered for potential modifications for advanced reactors.

Digital systems typically have not been considered in past PRAs. In advanced reactors, however, I&C systems will normally be digital. Digital I&C may have failure modes that have not been considered previously, e.g., more susceptible to what were previously minor voltage spikes and earlier failures under fire or loss of cabinet cooling events. PRA modeling needs to address the issues concerning digital system performance.

Data collection and analyses

Advanced reactors introduce different systems, materials and components and, hence, LWR data may not be applicable. The use of appropriate data is crucial in the assessment of the risk associated with a given reactor type. Therefore, collecting and analyzing data applicable to advanced reactors is essential. The existing NRC-reviewed MHTGR PRA will be used, as applicable.

This task includes addressing the data uncertainties. Understanding the uncertainties is a very important aspect for any PRA; it is much more crucial for these types of reactors given limited or lack of operating experience, experimental data, and the expected significant use of the PRA in the licensing process.

Human reliability analysis

The advanced reactor designs are proposed with strong reliance on the premise that they will be human-error free and that, if an event occurs, human intervention will not be necessary for an extended period of time. Human reliability methods were developed to assess the impact of human performance on plant safety. When dealing with long-term and slowly evolving accidents, such as those expected to be dominant in graphite-moderated reactor accident sequences, the probability of human intervention and error need to be considered. In addition, operator performance may be affected by having multiple modules that share the same control room, both from a common mode failure and as the result of operator workload from monitoring multiple modules. The likelihood of errors of commission or omission also need to be understood under these conditions.

Other events (internal flood, fire and seismic)

The response of digital electronics (not entirely unique to advanced reactors), in a fire or flood may be quite different from that of electro-mechanical components. The differences may not be just in probability but also in the kinds of failures that could potentially occur. Furthermore, current plants have shown that the core damage frequency from external events may be similar to that from internal events. Therefore, external events need to be considered for advanced reactors from a scoping perspective to identify unique vulnerabilities.

Quantification

The SAPHIRE code could be used in the performance of an independent PRA. The code needs modifications for a full scope PRA (external and internal events, full and low power). A full scope PRA will generate many more "cut sets" than can reasonably be handled now. In addition, the rationale developed for other designs for pruning the results may not be appropriate for advanced reactor designs. Source terms and consequences which will be evaluated as part of the severe accident and consequence work, need to be incorporated into the PRA tool. SAPHIRE needs modifications to integrate Level 1 analyses with Level 2 and Level 3 analyses, and dynamic modeling.

Uncertainties

Identification of uncertainties will help the decision-making process for deciding either to reduce the uncertainties by more research or to strengthen the regulatory requirements and oversight, e.g., defense-in-depth and safety margins. There are three types of uncertainty: modeling, data, and completeness. It is necessary to identify and understand the significance of the modeling and completeness uncertainties.

Other operational modes

Unique operating characteristics of advanced reactors, operating in other than full power mode, need to be examined to be incorporated into the PRA.

Multiple modules

Current PRAs are usually performed for a single unit or sometimes for two sister units operating independently, but considering cross-ties. Advanced reactors may operate up to 10 modular units at a site with a centralized control room. The PRA tool needs to address potential interactions among the multiple units and the potential effects of smaller operator staffs in a common control room. The latter is addressed under human reliability analysis.
Risk metrics

The concepts of core damage frequency (CDF) and large early release frequency (LERF) may not be the best figures of merit for some advanced reactor designs. Appropriate surrogates for Level 3 PRA results (offsite consequences) which will be performed as part of severe accident and consequence work) need to be considered for advanced reactors and incorporated into the PRA. Therefore, either the current subsidiary figures of merit need to be verified or more appropriate figures of merit identified, consistent with NRC top level safety goals.