ISSUED: JULY 31, 2002
CERTIFIED: BY T. KRESS, AUGUST 8, 2002

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FUTURE PLANT DESIGNS SUBCOMMITTEE
MEETING MINUTES- JULY 8, 2002
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs met on July 8, 2002, at 11545 Rockville Pike, Rockville, MD, in Room T-2B3. The purpose of this meeting was to review and discuss with the NRC staff the draft Advanced Reactor Research Plan and its implication on the NRC's regulatory framework.

The Subcommittee received no written comments from members of the public regarding the meeting. The entire meeting was open to public attendance. Dr. Med El-Zeftawy was the cognizant ACRS staff engineer and the designated federal official for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 5:40 p.m.

ATTENDEES

ACRS Members

T. Kress, Subcommittee Chairman
M. Bonaca, Member
P. Ford, Member
G. Leitch, Member
M. El-Zeftawy, ACRS Staff
V. Ransom, Member
S. Rosen, Member
J. Sieber, Member
G. Wallis, Member

Principal NRC Speakers

F. Eltawila, RES
M. Drouin, RES
J. Muscara, RES
R. Lee, RES
J. Flack, RES
S. Rubin, RES
D. Carlson, RES

Principal Industry Speakers: None.

A complete list of attendees is in the ACRS Office File and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. Thomas Kress, Subcommittee Chairman, convened the meeting at 8:30 a.m. Dr. Kress stated that the purpose of the meeting is to discuss with the NRC staff the proposed draft advanced reactor research plan and review its implication on the NRC's regulatory framework.

5|15

NRC Staff Presentation

Dr. John Flack, Office of Nuclear Regulatory Research (RES), stated that based on the Commission's direction, the NRC staff made a commitment to develop an advanced reactor research plan that would be used to develop and guide a comprehensive advanced reactor research program. This plan would help formulate and set directions for research programs, including programs to develop a regulatory framework for advanced designs, and analytical tools and experimental data to independently assess the safety capacity of the new reactor designs.

The new reactor designs that are being considered within the scope of the advanced reactor research plan include the Pebble Bed Modular Reactor (PBMR), Gas Turbine-Modular Helium Reactor (GT-MHR), Westinghouse AP-1000, and the International Reactor Innovative and Secure (IRIS). Generation IV reactor concepts have not been included in the plan, because of their preliminary stage of development. The plan, however, is expected to be a living document, and will be updated later on to accommodate GEN IV reactor concepts, and other designs such as European Simplified Boiling Water Reactor (ESBWR), Boiling Water Reactor (SWR-1000) , and Atomic Energy of Canada Limited (ACR-700)

The RES envision the proposed research plan to be used in identifying the following:

- Key research areas and activities,
- Technical and safety issues and pathways to resolution,
- Methods and tools to address technical or safety issues,
- Technical staff responsibilities,
- Links and flow of information among the various technical disciplines,
- Key research outputs results and links to the regulatory process,
- Priorities used to allocate resources,
- Key milestones and resources over a 5-year period (FY 02- FY 06)

The proposed research plan originates from a technology-neutral perspective. The plan is structured to capture both technical and potential safety issues that involve great uncertainties, and identifies capabilities that will enable the staff to independently ask the right questions, At this point, the plan does not delineate the research that will be conducted by the NRC. Rather, it identifies information gaps that exist at NRC in terms of analytical tools and data.

Ms.Mary Drouin, RES, stated that a regulatory framework is needed to license and regulate advanced reactors. NRC 's experience has been focused on current light water reactors (LWRs) and has limited applicability to advanced reactors. The framework for current LWRs has evolved without the benefit of insights from probabilistic risk assessments (PRAs) and severe accident research. It is anticipated that PRA will play a greater role for the advanced reactors. RES is proposing to develop an approach that would be applicable to all advanced reactors and it is "technology neutral". The approach will ensure an effective use of both deterministic and probabilistic methods.

Dr. Joseph Muscara, RES, stated that the NRC needs to develop independent research and expertise in the high-temperature materials area for high-temperature gas cooled reactors (HTGRs) to evaluate and establish a technical basis for licensing advanced reactors. The advanced reactors are significantly different from LWRs. The staff needs to examine the behavior of high-temperature metals and graphite, higher coolant temperatures, a coolant that

does not change phase, different degradation mechanisms such as creep, and the behavior of metallic and graphite components. In HTGRs, graphite acts as a moderator and reflector as well as a major structural component that may provide channels for the fuel and coolant gas, channels for control and shutdown, and thermal and neutron shielding.

The staff needs to develop independent research capabilities in the materials area beyond the licensing basis to understand safety margins and failure points and reduce uncertainties. Potential technical issues that need to be addressed include the availability and applicability of national codes and standards for design and fabrication of metallic and graphite components for service in HTGR high temperature helium environments, lack of appropriate data bases for calculating fatigue and creep, the effects of impurities including oxygen in the high temperature helium, aging behavior of alloys, sensitization of austenitic alloys, treatment of pipes as a vessel, degradation by carburization and oxidation of metals, performance of graphite under high levels of irradiation, and lack of data on oxidation kinetics of reflector grade graphite.

Dr. Muscara added another potential issue could be for the Pebble Bed Modular Reactor (PBMR) design is the understanding and prediction of the mechanics of pebble flow including temperature effects on pebble friction and flow, mixing of fuel and graphite pebbles at the central reflector core, compaction, hang-up, and bridging. The NRC staff needs to develop independent research to gain confidence and understanding of defense-in-depth for the advanced reactors.

Dr. Muscara indicated that the European Community (EC) and Japan (JAERI) have considerable research on high-temperature materials for HTGRs. The EC research is currently reviewing the state of the art on graphite properties to set up a suitable data base. The EC is planning to perform oxidation tests at high temperatures on fuel matrix graphite and on advanced carbon-based materials to obtain oxidation resistance in steam and in air. The EC welcomes the NRC's participation in its high-temperature materials program. NRC participation will be through the exchange of research results, and not funds. Cooperation with JAERI will provide access to their high temperature corrosion data and design codes for the HTGRs.

Mr. Stuart Rubin, RES, briefed the Subcommittee on the HTGR fuel analysis. He stated that the HTGRs (e.g., the PBMR and GT-MHR) have unique safety features and characteristics such as the all-ceramic fuel element containing high integrity high performance TRISO coated fuel particles (CFPs). The design of modular HTGRs involves many billions of CFPs contained within hundreds of thousands of graphite fuel elements that comprise the fueled core. The TRISO CFPs provide the principal safety barrier and primary containment function against release of fission products (FPs) to the environment.

Mr. Rubin stated that the qualification of HTGRs fuels will be based on a wide range of technical areas and specific factors such as FP release and particle failure rates. The technical areas include fuel design, fuel manufacturing process, design specific core operating conditions, design basis accident conditions, and postulated accident conditions beyond design basis.

The purpose of the regulatory research plan in the area of HTGR fuel performance is to establish NRC's infrastructure of knowledge, data, and tools needed for the performance analysis of HTGR fuels with TRISO CFPs and IRIS fuels. This infrastructure is needed to support the staff's review of a PBMR, GT-MHR, or IRIS application.

The scope of planned research include irradiation testing, accident condition testing, analytical codes and methods development, and fabrication knowledge and information. Mr. Rubin indicated that to predict CFP performance, and a deterministic approach to the source term, capabilities in a number of interfacing areas will be needed. These include nuclear analysis for fuel burnup, fast fluence and thermal fluence, thermal hydraulic analysis of normal operating core temperature distributions, and core temperature and flow distributions for fuel oxidation during postulated air intrusion events. In addition, the FP release rates from the fuel during normal operation and postulated accidents are key inputs to the accident source term calculation. RES plans to build on international knowledge and experience.

Mr. Donald Carlson and Mr. Richard Lee, RES, briefed the Subcommittee on research activities in the area of reactor systems analysis that includes thermal hydraulic analysis, nuclear analysis, and severe accident and source term analysis. For the thermal hydraulic analysis of helium-cooled, graphite moderated reactor systems, RES will develop an approach that provides the data and modeling tools needed for specific heat transfer and fluid flow phenomena, including "multi-phase" fluid flow with convective, conductive and radiative heat transfer in irregular and complex geometries. Research in the area of nuclear analysis will start with the development of modern, general-purpose nuclear data libraries that will support all nuclear analysis activities for reactor safety, materials safety, waste safety, and safeguards.

In the area of sever accident and source term analysis, Mr. Lee discussed the data and analysis tools needed for evaluating the progression of credible severe accident scenarios involving core damage phenomena such as fuel melting or high-temperature chemical attack, and modeling any resulting releases and transport of radioactive Fps within and outside the reactor system boundaries. RES will also capitalize on international program and activities and will build on existing LWRs tools (e.g., PARCS, TRAC-M, and MELCOR).

<u>SUMMARY OF SUBCOMMITTEE COMMENTS</u>

- Dr. Kress noted that the development of FP release models for the TRISO fuels should be considered as a key research need for the HTGRs. He indicated that the current models for FP release in the MELCOR code are empirical and based on data obtained from LWRs at burnup levels less than 45 Gwd/t.

- Mr. Rosen stated that the proposed Advanced Reactor Research Plan should include Generation IV concepts and should address issues associated with technology concepts that are significantly different from LWRs.

- Dr. Kress noted that the proposed plan should be developed for experiments to investigate degradation and FP release characteristics of the advanced LWR's core with very high burnup fuel.

- Dr. Apostolakis noted that a risk-informed approach for selecting design-basis events and choosing acceptance criteria for the advanced reactors needs to be developed.

- Dr. Powers noted that design-basis accidents should not be considered in the advanced reactor research plan. He added that design-basis accidents do create unnecessary burden for both licensees and regulators.

- Dr. Kress indicated that because there is a general need for large-scale integral testing of new concepts, RES should evaluate the viability of "testing by test" concept.

SUBCOMMITTEE DECISIONS AND ACTIONS

This matter will be discussed during the ACRS meeting on July 11, 2002. The Committee expects to write a letter on this matter.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee agenda

2. Subcommittee status report.

3. U.S. Nuclear Regulatory Commission, Advanced Reactor Research Plan (Draft), Revision 1, Office of Nuclear Regulatory Research, June 2002.

****

Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co., Inc. (Court Reporters and Transcribers) 1323 Rhode Island Ave. , NW., Washington, DC 20005 (202) 234-4433.

Reactor Systems Analysis for Advanced Reactors

Presented to the
Future Plant Designs Subcommittee
ACRS
July 8, 2002

by

Donald E. Carlson
Richard Y. Lee
Office of Nuclear Regulatory Research

Reactor Systems Analysis for Advanced Reactors

Scope and Goals

- Reactor Systems Analysis encompasses:
  - Nuclear Analysis
  - Thermal-Hydraulics Analysis
  - Severe Accident and Source Term Analysis

- Research Program will provide data and validated reactor system analysis tools appropriate for advanced reactors

- Allows independent check of applicant's analyses and better understanding of technical issues, uncertainties, and safety margins

Reactor Systems Analysis for Advanced Reactors

Outline

- Nuclear Analysis (D. Carlson)
  - ALWRs: AP1000 and IRIS
  - HTGRs: PBMR and GT-MHR
- Thermal-Hydraulics Analysis (R. Lee)
  - ALWRs: AP1000 and IRIS
  - HTGRs: PBMR and GT-MHR
- Severe Accident and Source Term Analysis (R. Lee)
  - ALWRs: AP1000 and IRIS
  - HTGRs: PBMR and GT-MHR

Reactor Systems Analysis for Advanced Reactors

Nuclear Analysis

Nuclear Analysis encompasses:

- ➢ Core neutronics – both static and dynamic
  - Reactivity effects - transients, feedback, control, shutdown
  - Spatial power distributions, stability
- ➢ Nuclide generation and depletion
  - for core neutronics
  - for decay heat power, radiation sources, and releasable inventories
- ➢ Radiation transport and attenuation
  - for material activation and fluence damage
  - for radiation shielding and protection
- Out-of-reactor criticality safety (burnup credit), decay heat, radiation shielding, nondestructive assay, etc.

Reactor Systems Analysis for Advanced Reactors
Nuclear Analysis

Advanced Light Water Reactors – IRIS
Nuclear Analysis Issues:

- Fuel depletion modeling and validation for analysis of fuel >5% initial enrichment, significantly higher moderator-to-fuel ratios, advanced burnable poison designs, and burnup levels to 80 GWd/t

- 5- to 8-year straight-burn core

- Decay heat power modeling and validation of high burnup fuel

Reactor Systems Analysis for Advanced Reactors
Nuclear Analysis

Advanced Light Water Reactors – IRIS

- Identify relevant physics benchmark data (Switzerland, Belgium, U.K., France, U.S.)

- Pursuing participation in international programs (IAEA, EC, OECD/NEA)

Reactor Systems Analysis for Advanced Reactors
Nuclear Analysis

HTGRs - GT-MHR and PBMR

Unique Features:

- FP retaining coated fuel particles, graphite as the moderator and structural material, and helium as the coolant
- Uranium enriched to 4-8% for PBMR, 19.9% for GT-MHR
- Long annular core geometries
- Control and shutdown absorbers located in graphite reflector region

➢ Similar code modeling and validation issues for PBMR and GT-MHR

Reactor Systems Analysis for Advanced Reactors

Nuclear Analysis

HTGRs – GT-MHR and PBMR

Nuclear Analysis Issues:

- Temperature coefficients of reactivity
- Worth of reactivity control and shutdown absorbers
- Moisture ingress reactivity
- Reactivity transients
- Little or no in-core instrumentation
- Graphite annealing heat sources
- For GT-MHR: fertile and fissile particles, burnable poisons, fuel & poison zoning for power shaping
- For PBMR: pebble burnup measurements and discharge criteria, hot spots, analytical treatments of quasi-random local mixing of pebbles with different burnups, fission powers and decay heat powers

Reactor Systems Analysis for Advanced Reactors

Nuclear Analysis

HTGRs – GT-MHR and PBMR

Nuclear Analysis Issues for TRISO Fuel Testing:

- **Reactivity Transients**: For accident testing of TRISO fuels, define the maximum power transients (e.g., prompt pulses) that can arise from credible reactivity accidents in a given HTGR design.
- **Out-of-Pile Accident Testing**: Evaluate how radionuclide decay and other physical changes that occur in the fuel before out-of-pile accident testing can affect fuel performance in slow heatup tests and rapid transient tests.
- **Irradiation in Test Reactors versus HTGRs**: Evaluate how nuclide inventories and fuel performance can be affected by irradiation in test reactors with accelerated burnup rates and nonprototypic fuel temperature histories, neutron fluences, and neutron energy spectra.

Reactor Systems Analysis for Advanced Reactors

Nuclear Analysis

HTGRs – GT-MHR and PBMR

BNL

- Preparing modern nuclear data libraries (from ENDF/B-VI)
- Starting scoping studies for core neutronics and decay heat analysis
- Initiated PARCS code modifications to incorporate r-theta-z geometry
- Envision PIRT exercises to identify and prioritize data and modeling needs
- Planning cooperation with MIT on core depletion analysis tool
- Pursuing opportunities for HTGR-related domestic and international cooperation (IAEA, EC, NEA – Physics benchmark data from Japan, China, Russia, Switzerland, France, Germany, U.S., U.K.)

Reactor Systems Analysis for Advanced Reactors
Thermal-Hydraulics Analysis

Advanced Light Water Reactors – AP1000

- For AP1000, test programs conducted in support of AP600 remain valid for many of T/H processes that are important to AP1000

- Some T/H phenomena are not well represented by previous tests for conditions expected during hypothetical accident in AP1000, i.e., T/H processes that strongly depend on higher core steam production rate (e.g., entrainment from horizontal stratified flow, upper plenum pool entrainment and de-entrainment)

Reactor Systems Analysis for Advanced Reactors
Thermal-Hydraulics Analysis

Advanced Light Water Reactors – AP1000

- Experiments at Oregon State University to address entrainment from horizontal flow, upper plenum pool entrainment and de-entrainment

- Experiments at the PUMA facility (Purdue University) to address low pressure critical flow.

- Experimental data to resolve ECC bypass for direct vessel injection is being addressed in activities related to the Korean Advanced Reactor.

Reactor Systems Analysis for Advanced Reactors

Thermal-Hydraulics Analysis

Advanced Light Water Reactors – IRIS

- Modular light water reactor with a power of 335 MWe
- Steam generator, pressurizer, and coolant pumps are located internally in RPV
- T/H issues are – two-phase flow and heat transfer in helical tubes, two-phase natural circulation, containment-RCS interaction, and parallel channel flow instabilities
- Need integral and separate effects tests data to validate T/H codes

Reactor Systems Analysis for Advanced Reactors

Thermal-Hydraulics Analysis

HTGRs – GT-MHR and PBMR

- Analytical tool needed to model fluid flow and heat transfer in HTGRs
- Porous and solid structure
- Spherical fuel element for PBMR
- Need to model turbo-machinery and passive decay heat removal systems.
- Envision the use of TRAC-M and FLUENT

Reactor Systems Analysis for Advanced Reactors

Thermal-Hydraulics Analysis

HTGRs – GT-MHR and PBMR

- Initiate the development TRAC-M for HTGR analysis. Development of TRAC-M will build upon HTGR analysis codes GRSAC and THATCH
- Test data from HTGR domestic and international research are being evaluated for their applicability for current designs
- Use PIRT process to develop needs for code development and data for assessment
- Will capitalize on domestic and international collaborations

Reactor Systems Analysis for Advanced Reactors
Severe Accident and Source Term Analysis

Advanced Light Water Reactors – AP1000

- Evolution of severe accidents and source terms will be similar to AP600.

- In-vessel melt retention feasibility is to be examined for AP1000.

- If in-vessel melt retention cannot be assured and in the event of reactor vessel breach, ex-vessel severe accident phenomena will be assessed for AP1000.

Reactor Systems Analysis for Advanced Reactors
Severe Accident and Source Term Analysis

Advanced Light Water Reactors – IRIS

- Evaluate the evolution of severe accidents and source terms for IRIS.

- MELCOR modeling of FP transport throughout the the RCS has to account for unique features of the design (e.g., helical tubes).

- Envision the use of PIRT to identify data and modeling needs.

Reactor Systems Analysis for Advanced Reactors
Severe Accident and Source Term Analysis

HTGRs – GT-MHR and PBMR

- Types of sequences and FP release and transport in HTGRs are expected to be different.
- Different fuel design (spherical, block/prismatic) and reactor internal structure
- Initiated MELCOR development and modeling for HTGR. Use of GRSAC to support this effort.
- Initiated TRISO fuel PIRT.
- Examining past research (Germany, Japan, IAEA)
- Planning to participate in European Commission new initiative on HTGR research ($16M, 4-year program)

Reactor Systems Analysis for Advanced Reactors

Summary

- Capitalize on International Programs and Activities
- Build on LWR tools (e.g., PARCS, TRAC-M, MELCOR)
- Expand infrastructure to address advanced reactor technologies (e.g., graphite, helium, higher burnup)

ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE

Advanced Reactor Research Plan - Materials Analysis

July 8, 2002
Rockville, MD

Dr. Joseph Muscara, (301) 415-5844
Senior Technical Advisor for Materials Engineering Issues
US NRC Office of Nuclear Regulatory Research
Division of Engineering Technology

Materials

- Background
- Metals Issues and Research
- Graphite Issues and Research
- International Research Cooperation
- Summary

Background

- Behavior of metallic and graphite components is a key research area important to primary system integrity
- A sound technical basis is needed for evaluating integrity and failure modes
  - Integrity of components is necessary to avoid air, water, or steam ingress into the pressure boundary and maintain core geometry
  - Defense-in-depth barrier to release of radioactivity from primary system coolant
- Information from the materials research area is needed for conducting probabilistic risk assessments (PRA)
  - Failure probability data is not available from experience, therefore large uncertainty
  - Information may be developed from research to identify and quantify degradation processes

Issues for High Temperature Metallic Components

- Availability and applicability of national codes and Standards
- Lack of appropriate data bases for calculating fatigue, creep, and creep-fatigue lifetimes
- Effects of impurities including oxygen on degradation
- Aging behavior of alloys
- Sensitization of austenitic alloys
- Degradation by carburization, decarburization, and oxidation
- Treatment of connecting pipe as a vessel
- Inspection of HTGR and ALWR components

Design Codes - Metals

- Lack of design codes and standards
  - ASME Code Cases N-499, N-201, and subsection NH for application to high temperature materials design
  - Based on studies conducted in the '70s and '80s for LMFBRs
- Data of the '90s led to improvements in correlations for creep and creep-fatigue
- Effect of helium coolant with impurities (oxygen) on reduction in strength, fatigue life, and creep not considered
  - Experience and research for LWRs has shown potential detrimental effects of the environment

Environmental Effects on Fatigue, Creep, and SCC

- Lack of data bases on fatigue, creep, and stress corrosion cracking (SCC) for evaluating lifetime design
  - Temperature, stress, strain rate, and impurities such as oxygen reduce fatigue and creep life and increase susceptibility to SCC
  - Increase in crack growth rate for fatigue, stress corrosion, and crevice corrosion
  - Research will be conducted on fatigue, creep, SCC, and crevice corrosion cracking
    - Oxygen, chloride, temperature, strain rate & range, stress
  - Confirm and quantify any enhancement of degradation
  - Provide a database to update codes and standards as necessary

Connecting Pipe

- Consideration of connecting pipe as a vessel
  - Designed, fabricated, and inspected to the same rules as a RPV
    - Double-ended break is not considered as a design basis
    - No mitigating systems are incorporated in the design
  - Pipe as a vessel is not realistic
    - Much thinner wall than a RPV for the same pressure
    - Any cracking would propagate through wall rapidly

Carburization, Decarburization, and Oxidation

- Dependent on composition of coolant, and/or the presence of particulates
- Carburization
  - Ferritic: hard, brittle layer
  - Austenitic: chromium depletion
- Decarburization
  - Soft layer, reduced fatigue and creep life
- Study carburization, decarburization, and oxidization as a function of time and temperature in helium gas with impurities including oxygen
  - Metallographic studies and mechanical testing
  - Characterize conditions under which these phenomena occur

Aging Behavior and Sensitization of Austenitic Steels

- Aging and sensitization
  - Embrittlement
  - Susceptible to SCC
  - Low-temperature sensitization
- Thermal aging and sensitization research
  - As-received and welded condition
  - Mechanical and microscopic studies
  - Quantify time and temperature for different levels of sensitization and embrittlement
  - Identify the potential and the degree of these solid state reactions
  - Provide database for safety evaluations and improvements to codes

Components Removed From Service

- Opportunity to validate degradation mechanisms
- Components that have history and analysis available
  - Microstructural studies
    - SCC, fatigue, creep cracks, solid state transformations carburization, decarburization, oxidation
  - Mechanical tests such as tensile, fracture, fatigue, creep
    - Creep and fatigue tests will determine remaining life
    - Evaluate the predictive capability of current design codes

Inservice Inspection and Continuous Monitoring

- Long operating periods between short-duration outages
  - ISI intervals may be long and the amount of inspection limited
  - Effectiveness of ISI programs
  - Continuous on-line, monitoring may be required
- Evaluation of ISI programs using risk informed methodology
  - Inspection frequency, inspection reliability, number of components, accessibility, and degradation mechanisms
  - Acoustic emission monitoring for
    - Fatigue, SCC, creep, and leak detection
    - Validation on operating reactor

Issues for High Temperature Materials, Graphite

- Lack of data on high levels of irradiation for current graphites
- Lack of predictive capability of irradiated properties from non-irradiated properties
- Lack of data on oxidation kinetics
- Applicability of graphite sleeve properties to large block graphite properties
- Lack of codes & standards for nuclear-grade graphite

Graphite Performance Under High Levels of Irradiation

- Current data - on old graphites
- Irradiation degrades, physical, thermal and mechanical properties
- These changes cause significant stresses and distortion during operation
- Loss of structural integrity, loss of core geometry, and potential problems with insertion of control rods
- Beyond 'turn-around', graphite will experience total loss of integrity.
- Study property changes under varying levels of irradiation and temperature
  - Review of available high dose irradiation data
  - Irradiation in a high flux test reactor
  - Microstructure evaluations, dimensional, mechanical, thermal and physical property measurements

Prediction of Irradiated Graphite Properties From Non-irradiated Properties

- Correlations are needed for predicting irradiated properties from the non-irradiated properties
- Non-irradiated and irradiated graphite properties depend strongly on the raw materials and manufacturing processes
- Some data available from 'old' graphites
- Development of new data is expensive and time consuming; reactor designers have proposed to use the data from 'old' graphites
- NRC needs to confirm that these data are realistic for a new graphite

Prediction of Irradiated Graphite Properties From Non-irradiated Properties (*Continued*)

- Conduct studies to determine irradiated graphite properties from non-irradiated graphite properties
  - Parametric studies with carefully controlled parameters will be conducted
    - Raw materials and processing parameters
    - Temperatures
    - Irradiation levels
  - Characterize the non-irradiated and post-irradiated properties
    - Mechanical, thermal, and physical properties
    - Anisotropy
    - Dimensional changes

Oxidation Kinetics of New Graphites

- Needed for evaluating heat generation, structural integrity and core geometry during normal operating and accident conditions
- Lack of data - especially for new graphites
  - Air ingress leads to corrosion and oxidation of graphite
    - Exothermic reaction
    - Loss of material and structural integrity
    - Reduction in fracture toughness and strength
    - Changes in thermal conductivity
  - Rates vary for block, fuel matrix, and dust graphites, and with the level of impurities
- Study oxidation on different graphites
  - Temperature, various levels of oxidants and irradiation
  - Oxidation rate, heat generation rate, material loss, mechanical and physical property changes, and microstructure

Variability in Large Block Graphite

- Applicability of graphite properties from 'thin' sections to large blocks
  - Designers may use graphite properties and experience from 'thin' section components, such as AGR-type, fuel sleeves
  - Mechanical and physical properties may vary through the block thickness
  - Irradiated properties may also vary through the thickness
- Conduct study and assess uniformity of properties through the block thickness
  - Strength, fracture toughness, density, thermal conductivity, coefficient of thermal expansion, level of chemical impurities, isotropy, and absorption cross-section
  - From these studies assess whether large block bulk properties would vary under irradiation conditions

Lack of Codes and Standards for Nuclear Grade Graphite

- Lifetime design codes
  - Creep, fatigue, strength, fracture toughness
- Materials specification that establishes minimum mechanical, physical, and chemical requirements
  - Limit elements detrimental to irradiation properties, or can cause degradation of other components
- NRC staff and contractors will work with national codes and standards organizations
  - NRC staff and contractor from ORNL are participating in development of an ASTM materials specification
  - Review and evaluate available design methods from different countries and make recommendations for development of a national design code
  - Staff assignee to work with graphite experts in UK to outline key aspects and requirements for a materials specification and a component design code

International Cooperation

- European Community and Japan have considerable research on high temperature materials for HTGRs
- The high temperature materials research plan has been shared with the international community
  - The EC has agreed with the importance and need for the research
  - Welcomes NRC participation in their high temperature materials research (HTR-M) program
  - Participation is through the exchange of research results, and not funds
- Much of the research described is addressed in the EC's current and future program
  - Pressure vessel steel containing 9% Cr
  - Turbine blade materials
  - Inservice inspection
  - Graphite

International Cooperation (*Continued*)

- Research possibly not fully addressed by the EC:
  - Coolant impurities on degradation of materials
  - Effectiveness of inservice inspection programs
  - Correlations of non-irradiated graphite properties to post-irradiation properties
- Exchange of NRC research results in these areas could be used for cooperation with the EC HTR-M program
- Cooperation with JAERI will provide access to their high temperature corrosion data and design codes for HTGRs.

Summary

- Key technical issues
- International Cooperation
- Infrastructure (Expertise)

Advanced Reactor Research Plan: HTGR Fuel Analysis

July 8, 2002

Stuart D. Rubin
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Fuel Safety Objective
- Fuel Safety Research Issues
- Purpose of Fuel Safety Research
- Scope of Planned Research
  - Irradiation Testing
  - Accident Condition Testing
  - Analytical Codes and Methods Development
  - Fabrication Knowledge and Information
- Expected Regulatory Applications

HTGR Fuel Safety Objective

The fuel safety objective is to reliably contain and retain the radiologically important fission products within the TRISO coated fuel particles during all reactor conditions within the licensing basis. These conditions are: (1) normal operation; (2) design-basis accidents; and (3) potential severe accidents beyond the design-basis.

Fuel Safety Research Issues

- Completeness of previous **irradiation test** conditions (margins)
- Completeness of previous **accident condition test** conditions (margins)
  - Core heat-up performance and limits
  - Prompt supercritical reactivity pulse behavior and limits
  - Chemical attack performance behavior and limits
- **Fabrication** process to achieve quality and performance
  - Key process variables and acceptable ranges
  - QA product characterizations and statistical analysis methods
- Applicability of historical **testing methods**
  - Accelerated vs real-time irradiation testing
  - Ramp-hold vs actual temperature history accident condition testing
- Applicability of **performance models and methods**
  - Availability of irradiated materials property data
  - Particle failure and FP transport for mechanistic source term
- Prediction of maximum operating/accident temperatures
- Applicability of previous performance data to new fuel and plant designs
  - Q/A used for previous testing programs

Purpose of Fuel Research

- Explore limits (i.e., margins) of TRISO coated fuel particle performance and fission product retention capability
- Independently assess applicant claims of fuel performance and fission product release
- Develop information to support the review of applicant fuel qualification test plans and methods
- Develop independent tools to predict fuel fission product release and TRISO particle failure for licensing basis conditions
- Understand the effects of fuel fabrication on fuel performance

Scope of HTGR Fuel Research

- Irradiation testing
- Accident condition testing
- Analytical models and methods
- Fabrication process knowledge and information
- Staff technical expertise and information

Objectives for HTGR Fuel Irradiation and Accident Condition Testing

- Explore the limits/margins of fuel performance and fission product retention capability during irradiation and accidents

- Support evaluation of applicant fuel qualification program irradiation and accident condition test plans, methods and results

- Support development of independent analytical tools to predict fuel performance during operations and accidents

  - TRISO coated particle failure models
  - Fission Product transport/release models (source-term)

Potential Fuel Performance-Limiting Factors

environmental

- Pressure induced ("Pressure Vessel") particle failure (E)
- Fission product diffusion through coatings and matrix graphite (E)
- Coating defects arising from manufacture (e.g., "weak fuel") (M) Manufacture
- Kernel/coating interactions (fuel "kernel migration") (E)
- SiC disassociation, increased porosity (at high temperature) (E)
- Fission product chemical interaction with SiC (e.g., Pd attack) (E)
- Matrix graphite interactions with coated particles (E) outer
- Heavy metal contamination of the graphite matrix or OpyC (M)
- Chemical attack (e.g., oxygen) of silicon carbide layer (E)
- Large energy deposition (reactivity pulse) (E)

Explore the Limits/Margins of Fuel Performance and Fission Product Retention Capability

<u>Irradiation Conditions Beyond the Expected Design Basis</u>:

- Irradiation Temperature
- Burn-up
- Fast Fluence
- Coated Fuel Particle Power level

Explore Limits/Margins of Fuel Performance and Fission Product Retention Capability (Cont.)

<u>Monitor Fission Gas Release During Irradiations:</u>

- Diffusion through intact coating particles and matrix
- Release from failed coated particles

<u>Conduct Post-Irradiation Examinations:</u>

- Characterize fuel condition, particle failure mechanism(s)

Evaluate Applicant Fuel Irradiation Test Methods

- Accelerated vs Real-Time Irradiation Testing
- Obtain Knowledge/Experience in Irradiation Testing

Explore the Limits/Margins of Fuel Performance and Fission Product Retention Capability

<u>Accident Conditions: Beyond the Expected Licensing Basis</u>:

- Heatup Events:
  Fuel Irradiated Beyond Design Conditions
  Temperatures Beyond the Design-Basis

- Reactivity Events:
  Bounding Supercritical Reactivity Pulse

- Chemical Attack Events:
  Fuel Irradiated Beyond Design Conditions
  Oxidation Beyond the Licensing-Basis

Explore Limits/Margins of Fuel Performance and Fission Product Retention Capability (Cont.)

Monitor Fission Product Release During Accident Simulations:

- Diffusion through intact coated particles and matrix
- Release from failed coated particles

Conduct Post-Accident Simulation Examinations:

- Characterize fuel condition, particle failure mechanism(s)

Evaluate Applicant Fuel Qualification Accident Condition Testing Methods

- Accident condition heat up testing method:
  - Ramp-up and hold at constant temperature
  - Temperature vs time accident simulation

- Obtain knowledge/experience in accident condition testing

HTGR Fuel Testing Strategy

Leverage Resources with Cooperative Agreements and Technical Information Exchange:

- Cooperative Agreement with DOE
- Cooperative Agreement with the EC HTR-F
- Participate in IAEA Coordinated Research Project No. 6
- Cooperative Agreement with JAERI

  Exchange
- Information Exchange with INET

  information

Objectives for Fuel Performance Analysis Tool Development

Provide NRC staff with an independent capability to predict HTGR fuel performance:

- CFP Behavior/Failure During Normal Operation and Accident Conditions

- Fuel Fission Product Release During Normal Operation and Accident Conditions

Fuel Performance Analysis Tool Development Issues

- Coated particle irradiation and accident behavior (failure) depends on design, manufacture, irradiation environment
- PyC irradiated material properties data (e.g., dimensional change, creep, thermal expansion, Young's modulus) have uncertainties
- Important failure mechanisms require 3-D modeling (SiC surface flaws, layer de-bonding)
- Statistical variations of key properties associated with manufacture require Monte Carlo analysis
- Chemical interaction effects need to be included (e.g., SiC palladium attack)

Strategy to Develop HTGR Fuel Performance Analysis Tools

Establish cooperative agreements with organizations currently developing HTGR fuel performance analysis tools

- INEEL PARFUME Code
- MIT Fuel Performance Code
- EC HTR-F Fuel Performance Code
- Use Data from Cooperative Fuel Testing Agreements

Fuel Performance Analysis Tool Applications

- Assess applicant's in-core fuel particle integrity calculations (supplements empirical test data used in safety analyses)

- Assess applicant's predictions of in-core fuel fission product release calculations (for source term)

- Assess causes of in-reactor fuel performance anomalies and corrective actions

- Calculate fission gas release for input to NRC reactor accident and consequence analyses

Objectives for Fuel Fabrication Knowledge and Information

Provide NRC staff with in-depth knowledge of the key factors of fuel fabrication that ensure quality and performance of fuel over the plant (fuel supply) lifetime:

- Fuel fabrication *process* factors
- Fuel *product* factors (kernel, coated particle and element)
- Fuel process and product *quality control* measures

Strategy to Acquire Fuel Fabrication Knowledge and Information

- Cooperative agreement with the EC (HTR-F)
- Information exchange with DOE and ORNL on particle coating technology development
- Information exchange with INET (China) and JAERI (Japan)
- HTGR Pre-application review activities

Applications for Fuel Fabrication Knowledge and Information

- Input to policy decision on regulatory approach to ensure fuel quality and performance over the life of a plant
- Input to possible risk-informed performance-based fuel fabrication process/product technical specifications
- Input to risk-informed performance-based fuel fabrication inspection procedures
- Input to fuel fabrication facility inspector training

Research Products and Applications

- Review of fuel qualification programs
- Policy decision on fuel fabrication technical specifications
- Fuel fabrication facility inspection procedures
- Fuel safety limits and limiting conditions for operation
- Fuel condition on-line monitoring system evaluation
- Independent analysis and evaluation of fuel safety performance (licensing, operating experience)
- Fuel design and fuel process change evaluations
- Staff training on fuel technology

Summary and Conclusions

- Develops infrastructure of NRC analytical tools and data
- Explores TRISO fuel safety margins and performance
- Increases staff knowledge of key elements of fuel fabrication
- Builds on international knowledge and experience
- Centers on technical issues and research needs
- Reduces NRC resources and time by cooperative research
- Enhances NRC's capability to conduct HTGR COL reviews

"Framework" for Advanced Reactors

ACRS Subcommittee

Mary Drouin
Office of Nuclear Regulatory Research
July 8, 2002

OUTLINE

- Background
- Structure/Framework
- Plan
- Approach
- Issues
- Status

BACKGROUND

- Current regulatory structure/framework has limited applicability
- Need to address unique design and operational issues associated with advanced reactors
- Incorporate PRA results and insights into new framework
- Develop an approach applicable to all advanced reactor concepts under consideration

BACKGROUND (cont'd)

- Link framework for advanced reactors to "coherence" plan
  - SRM for current reactors
  - "provide plan for moving forward with risk-informed regulation to address regulatory structure convergence with our risk-informed processes"
- Development of plan for advanced reactor framework just started
  - Soliciting input on proposed approach

STRUCUTRE/FRAMEWORK

- Established at various levels
  - Generic – applicable to all currently envisioned advanced designs
  - Design Specific – applicable to one design or a group of similar designs
  - Combination of the above

- A combination of qualitative and quantitative criteria

STRUCTURE/FRAMEWORK (cont'd)

- PRA will be an integral part of future license applications
- Focus regulations on high risk areas
- Maintain basic principles
  - Defense in depth
  - Safety Margins

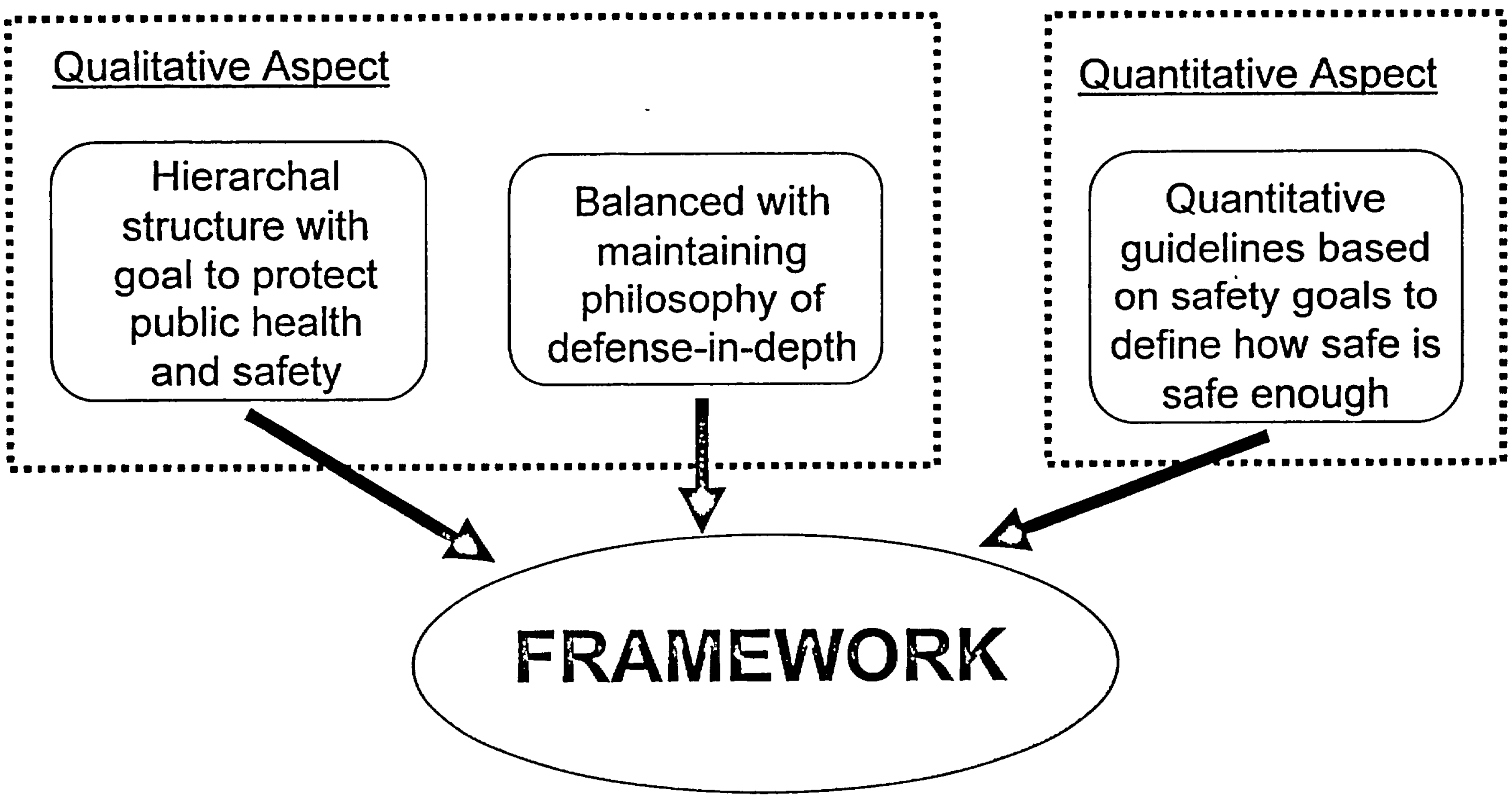
PLAN

- Start with Framework developed for risk-informing Part 50
- Use experience gained from risk-informing current LWRs
- Both policy and technical issues will need to be resolved

PLAN (cont'd)

- Outline a path for generating ~~decision-making~~ Framework criteria that:
  - are suitable for developing design and operating requirements for advanced reactors in a consistent, systematic and structured manner.
  - allow direct linkage of advanced reactor regulations to high level safety goals and principles
  - can be used to demonstrated that the safety goals are met (or exceeded)
  - are performance based.

CURRENT REGULATORY FRAMEWORK



CURRENT REGULATORY FRAMEWORK

GOAL

Protect Public Health and Safety

⇩

CORNERSTONES

Safe Nuclear Power Plant Operation

⇩

STRATEGIES

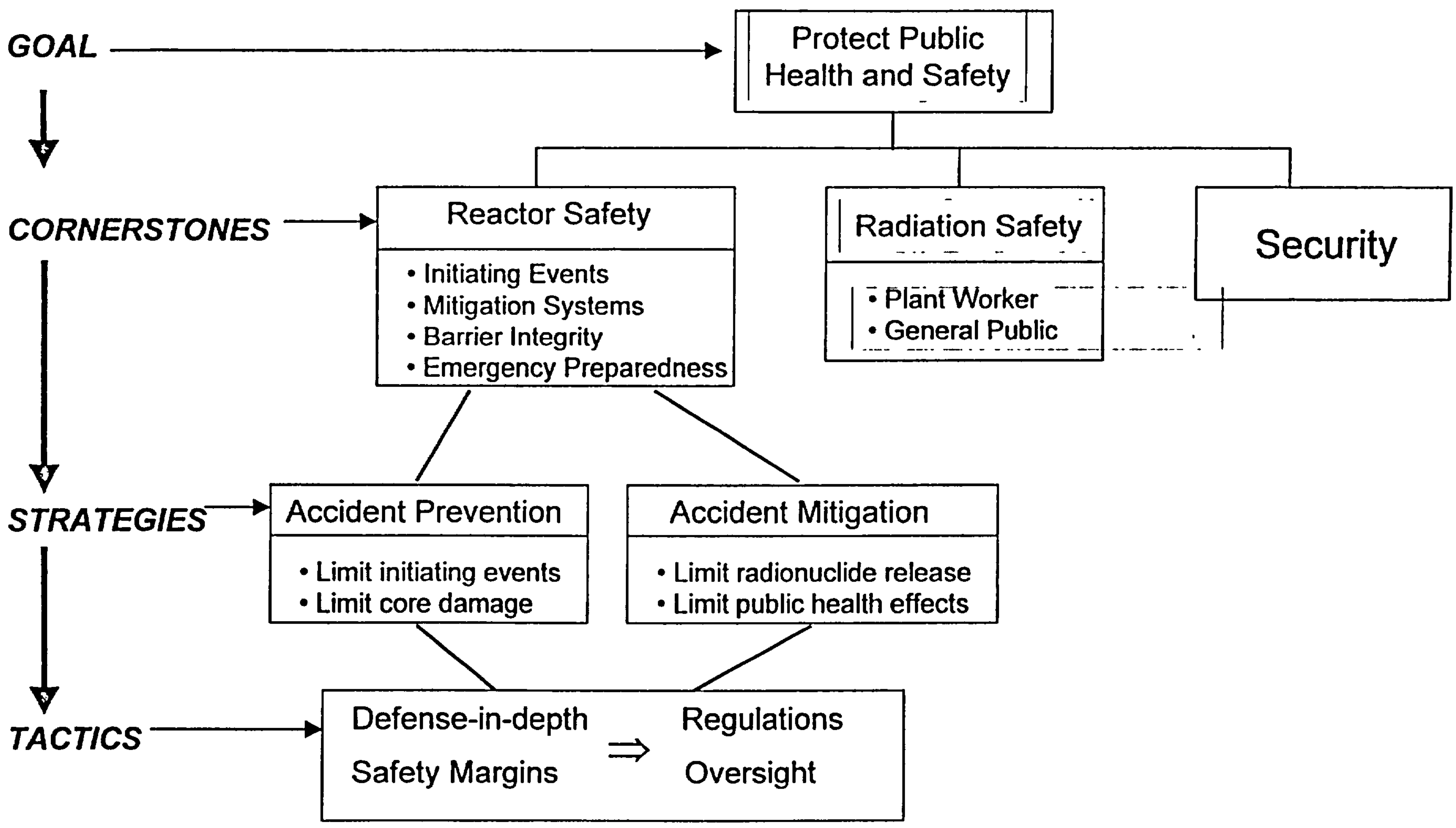
Accident Prevention and Accident Mitigation

⇩

TACTICS

Defense-in-depth, Safety Margins ⇒ Regulations and Plant Oversight

CURRENT REGULATORY FRAMEWORK



APPROACH

- Safety Goals
  - Applicable?
  - Additional goals needed?
- Cornerstones
  - Appropriate cornerstones identified, if not, modify
  - Additional cornerstones appropriate?
- Strategies
  - Appropriate qualitative strategies identified?
  - Appropriate quantitative guidelines?
- Tactics
  - Appropriate tactics identified (e.g., defense-in-depth, margins)?
  - Level of tactics (e.g., defense-in-depth) appropriate?

EXAMPLE OF POLICY ISSUES

- Should additional cornerstones (besides reactor safety) be included – radiation safety (worker), security, safeguards
- Should environmental risk metrics (land contamination) be considered
- Should level of safety be raised for new plants (explicitly/implicitly)
- Should criteria apply to single units or entire sites (what about mixed sites – current reactors and advanced reactors on same site)

EXAMPLES OF TECHNICAL ISSUES

- Evaluate if core damage frequency (CDF) and large early release frequency (LERF) are sufficient surrogates
- For the surrogates chosen, determine the appropriate quantitative guidelines
- Define what is the appropriate level of defense-in- depth, safety margin, etc.
  - Considerations will differ from those of current reactors where margins, defense-in-depth layers are well established

STATUS

- Only just started
- Will interact with stakeholders to solicit their input
- Preliminary plan – September 2002

ACRS Subcommittee
on Future Plant Designs

Advanced Reactor Research Plan

July 8, 2002

John H. Flack
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Introduction
- Discussion of Key Areas:

  - Framework
  - Fuel Analysis
  - Materials Analysis
  - Reactor System Analysis

- Other Technical Areas
- Future Plans

Meeting Objectives

- Interact with ACRS Subcommittee
- Comments and Feedback
- Discuss Future Plans

NRC Advanced Reactor Research Plan (Scope)

Plan Includes:

- Pebble Bed Modular Reactor (PBMR)
- Gas Turbine-Modular Helium Reactor (GT-MHR)
- International Reactor Innovative and Secure (IRIS)
- Westinghouse AP-1000

Expected Increase in Scope:

- ES Boiling Water Reactor (ESBWR),
- Boiling Water Reactor (SWR-1000),
- Atomic Energy of Canada Limited ACR-700.
- Generation IV

NRC Advanced Reactor Research Plan (Structure)

Structured around 9 Key Research Areas:

1. **Framework*** (tools)
2. Accident Analysis (PRA, human factors, instrumentation & control)
3. **Reactor Systems Analysis*** (T/H, nuclear analysis, severe accident analysis)
4. **Fuels Analysis*** (fabrication and performance)
5. **Materials Analysis*** (high-temperature metals, graphite)
6. Structural Analysis (external events, concrete performance)
7. Consequence Analysis (environmental impact)
8. Nuclear Materials and Waste Safety (enrichment process, fabrication)
9. Safeguards and Security

ACNW

* major research areas

Presentation on Specific Technical Areas:

- Framework
- Fuels Analysis
- Materials Analysis
- Reactor Systems Analysis

Probabilistic Risk Assessment (PRA)

- Support risk-informed performance-based regulatory process

  - policy issues and rulemakings
  - safety issues
  - uncertainties, defense-in-depth, and safety margins

- Review of Applicant's PRA, results, and insights
- Support research programs and activities

Technical Issues:

- Initiating event identification for advanced designs.
- Modeling different systems and structures (e.g, confinement).
- Modeling of passive systems.
- Applicability of data to advanced reactors.
- Modeling human performance (for multi-modular designs) and I&C.

Human Factors

Role of operator:

- Normal operations (e.g., configuration control).
- Accident initiation and response.

Technical Issues:

- Reliance on I&C and automatic systems.
- Staffing levels and multi-modular designs.
- Operator response to slowly evolving events.
- Models to support PRA applications.

Instrumentation and Control (I&C)

Application and reliance on advanced I&C for process controls in multiple modular facilities.

Technical Issues:

- Models and data to address new I&C reliability issues.
- Models to support PRA applications.
- Evolution and application of new technology.

Structural Analysis

Integrity of the reactor vessel support and confinement building structures.

Technical Issues:

- Concrete aging and performance at elevated temperature.
- Applicability of current industry codes/standards to modular HTGR design and construction features.
- Seismic response of (connected) vessels and graphite structures
- Soil structure interaction effects.
- Risk-Informed in-service inspection criteria and guidance.

Consequence Analysis

Treatment of radionuclides and chemical forms that may be different for advanced reactors

Technical Issues:

- User input into MACCS.
- Biological factors.
- Link to emergency planning.

Future Action

- Consideration of ESBWR, AECL ACR-700, SWR-1000.
- Additional stakeholder interactions.
- Transmit plan to Commission in Fall 2002.
- Implement and maintain living.