Developing Technical Insights on ACR-700

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Meeting with AECL on the ACR-700 Plant Design
September 26, 2002
Rockville, MD

Discussion Objective

- Start identifying areas for early NRC staff interactions with AECL on the technology-specific expertise, analysis capabilities, and data needed for assessing the ACR-700 design

Advanced reactors and their fuel cycles give rise to potential technical safety issues concerning:

Nuclear Materials Safety

Nuclear Reactor Safety

Nuclear Waste Safety

Safeguards

Major Areas in the Reactor Safety Arena:

- Regulatory Framework Development
  Risk-informed and performance-based decision-making criteria

- Accident Analysis
  Probabilistic risk assessment, human factors, instrumentation & control

- Reactor Systems Analysis
  Neutronics, thermal hydraulics, fuel behavior, severe accidents, source terms

- Materials Analysis
  Material properties, degradation, codes & standards

- Structural Analysis
  Containment/confinement performance, external challenges

- Consequence Analysis
  Dose calculations, environmental impact studies

Use of PIRT to Assess Reactor Systems Analysis Needs

PIRT Process: (<u>P</u>henomena <u>I</u>dentification and <u>R</u>anking <u>T</u>ables)

➜ 1. Design

➜ 2. Representative Scenarios

➜ 3. Important Phenomena

➜ 4. Desired Data and Models

➜ 5. Available Data and Models

➜ 6. Gaps in Available Data and Models

Potential Topics for ACR Reactor Systems Analysis (1)

Nuclear Analysis

Modeling and Data for Predicting:
- Reactivity Feedback Effects - Coolant, fuel, moderator
- Reactivity Control and Shutdown - Stability, vertical rods, horizontal injectors
- Power Distribution - On-line fueling, axial water gaps between bundles
- Point/Spatial Kinetics - Neutron lifetime, photoneutrons, spatial coupling
- Decay Heat Power - HWR versus LWR algorithms

Nuclear Analysis Codes:
- Lattice physics depletion code and models for generating ACR few-group nodal data
- Core management code and models for ACR static power distributions
- Spatial neutron kinetics, coupled with thermal hydraulics, for ACR transient analysis

ACR Code and Model Validation:
- Existing and planned database applicable for validating ACR code predictions
- Evaluation and treatment of code biases and uncertainties in safety analysis

Potential Topics for ACR Reactor Systems Analysis (2)

Thermal Hydraulic Analysis

Modeling and Data for Predicting:

- Quenching of the horizontal fuel bundles
- Flow rates between headers and tubes
- Pressure tube (PT) sagging heat transfer to calandria tube
- Thermal-hydraulic phenomena in calandria vessel after PT rupture
- Natural circulation flow and heat transfer in primary and secondary

ACR Thermal Hydraulic Validation:

- Two-phase flow in headers and feeders
- Steam generator performance in natural circulation, after small breaks
- Full-scale two-phase pump performance
- Flow resistance in deformed fuel channels

Potential Topics for ACR Reactor Systems Analysis (3)

Fuel Behavior Analysis

Modeling and Data for Predicting:
- Steady state ACR fuel rod behavior
- Transient ACR fuel rod behavior
- Transient behavior of ACR multi-rod bundles

Validation Data on ACR fuel behavior:
- Fuel rod bursting
- Pellet-cladding interaction

Potential Topics for ACR Reactor Systems Analysis (4)

Severe Accident Analysis

Modeling and Data for Predicting:
- Pressure/calandria tube failure
- Fuel failure propagation
- Debris/melt progression
- Calandria and shield tank failure

Validation Data on ACR Phenomena:
- Rapid energy insertion effects
- Heat transfer to moderator from sagging fuel and pressure tubes
- Melt relocation stages in ACR specific geometry

Potential Topics for ACR Reactor Systems Analysis (5)

Fission Product Transport Analysis

Modeling and Data for Predicting:

- Energetic scenarios like fuel-coolant interactions or reactivity excursions
- Specific models for ACR systems configurations, materials, and accident environments

Validation Data for ACR Phenomena:

- Direct release of fission products and actinides from fuel fragmentation and dispersal
- Release from fuel-coolant interactions with molten debris or dispersed fuel particulates

Approach to Developing Necessary Insights on ACR-700

Build on staff's LWR experience and relevant insights gained from CANDU 3 research and review:

- Familiarization with ACR-700 design, safety analysis, and data

- ACR-700 systems failure analysis

- PIRT process to identify and prioritize ACR-700 data and modeling needs for NRC reactor systems analysis

- Analysis to support the staff's initial understanding of phenomena in ACR-700

- Planning and implementation of activities for independent confirmatory analysis