ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

April 11, 2003

Overview

M. V. Bonaca
ACRS Chairman

Overview

- 500th Meeting Celebration
- Quadripartite Meeting
- License Renewal Activities
- Core Power Uprates
- Future ACRS Activities
- Sunset Activities

Quadripartite Meeting

Participants:

Germany, France, Japan, and U.S.

Observers:

Sweden and Switzerland

Topics:

- **Safety Culture**
- **Probabilistic Safety Assessments**
- **Thermal-Hydraulic (T/H) Codes**
- **Stress Corrosion Cracking**

ACNW Members participated in the discussion of waste management issues

License Renewal

- Reviewed three applications since July 2002
- Plan to review five applications in 2003
- Improvements to generic license renewal guidance - July 2003
- Future inspection of commitments
- Streamlined review of license renewal applications – from 2 subcommittee and 2 full committee meetings to 1 subcommittee and 1 full committee meetings

Core Power Uprates

- **Extended Power Uprate Review Standard**
  - ***Plan to review the draft final Standard after reconciliation of public comments***
- **Expect to review seven extended power uprate applications in 2004**
- **Plan to revisit the need for ACRS to review all power uprate applications after review criteria are established by the staff and the process is stabilized**

Future ACRS Activities

- Advanced Reactor Reviews
  - Early site permit process/ applications
  - Pre-application documents
- Thermal-Hydraulic Codes
- Risk-informed Regulation
- Reactor Oversight Process
- PRA quality

Future Activities (Cont'd)

- Vessel head penetration cracking and degradation
- Mixed oxide fuel fabrication facility
- Safeguards and Security matters
- American Nuclear Society Standard on low-power and shutdown risk

Sunset Activities

- Process in place to ensure that the Commission and EDO priorities are adequately considered in prioritizing the ACRS work.

Sunset Activities (Cont'd)

- ACRS Planning and Procedures Subcommittee Reviews NRC Staff Requests and Assesses:
  - Value-Added from ACRS Review
  - Previous ACRS Related Reviews
  - Significance to NRC's Regulatory Process
  - Timing of Committee's Review
  - Committee's Current Workload

ADVANCED REACTOR DESIGNS

T. S. Kress

Recent ACRS Reviews Associated With Advanced Reactors

I. Early Site Permit process (ESP)

II. Options for resolving policy issues

III. AP1000 review activities

Early Site Permit Activities

Full Committee Meeting
November 7, 2002

- NEI's approach for ESP
- Staff's approach for a review standard
- Briefing only, no report

Early Site Permit Activities (Cont'd)

Full Committee Meeting

March 7, 2003

- Reviewed a draft of the proposed review standard
- ACRS Report March 12, 2003

ACRS March 12, 2003 Report

The Review Standard

- **Is appropriate for reviewing ESP applications**
- **Will accommodate industry's proposed use of plant parameter envelope concept**

Policy Issues

Staff identified 7 policy issues

- **Expectations for enhanced safety**
- **Defense-in-depth**
- **International safety standards and requirements**
- **Event selection and safety classification**
- **Source term**
- **Containment vs. Confinement**
- **Emergency preparedness**

ACRS Report December 13, 2002

- We agreed that the Key Technical Issues (KTIs) identified by the staff needed resolution before certification reviews
- The preferred options to address the KTIs were consistent with opinions we had previously expressed

AP1000 Review Activities

- **Phase 1 – Establish goals and estimate for pre-licensing review**

  Completed - Letter 6/21/00

- **Phase 2 – Develop positions on 4 key issues identified in Phase 1**

  Completed - Report 3/14/02

AP1000 (Cont’d)

Phase 2 - Report 3/14/02

- Agreed with staff position on key issues
- Raised flag on appropriate range of PI-group values for scaling

AP1000 (Cont'd)

Phase 3 (Design Certification) -

In progress

- **Westinghouse/ACRS meeting 11/7/02**
- **ACRS PRA Subcommittee 1/23-24/03**
  - **- Reliability of ADS-4 squib valves questioned**

AP1000 (Cont'd)

T/H Subcommittee 3/19-20/03

- Entrainment of liquid at ADS-4 and top of core still an issue
- Potential for Boron precipitation
- Sump strainer design

AP1000 (Cont'd)

- Future Plant Designs and T/H Subcommittees 7/03

  (Containment structural design, materials, regulatory treatment of non-safety systems, shutdown maintenance, open items)
- Full Committee Interim Report/DSER 9/03
- Full Committee Final Report/FSER 7/04

Pressurized Thermal Shock (PTS) Reevaluation Project

W. J. Shack

Current PTS Rule

- 10 CFR 50.61 provides assurance that reactor vessels will have a low likelihood of failure due to PTS
  - Only a few plants will approach current screening criteria during the initial 40 year license period
  - About 10 plants will approach the current criteria during an additional 20 year extended operation

Technical Bases for PTS Rule

Estimation of the frequency of vessel failure requires:

- **Identification of sequences that could lead to rapid cooling of the vessel**
- **Knowledge of the pressure, temperature, and heat transfer coefficient adjacent to the embrittled portion of the vessel**
- **Determination of the thermal stress, fracture toughness and flaw distributions in the vessel**
- **Probabilistic fracture mechanics analyses**

Current Reevaluation Studies

- More complete description of sequences leading in to PTS
- More realistic distributions for flaw density and geometry
- Use of improved probabilistic fracture mechanics code, FAVOR

Current Reevaluation Studies (Cont’d)

- Systematic consideration of uncertainties in:
  - Frequency of initiating events
  - Fracture toughness
  - Thermal-hydraulic conditions

Plant-Specific Studies (Three Plants)

- **Current PTS screening criteria are very conservative**
  - **At current screening limits mean value of failure frequency is about 1 x 10^{-8}/year**
  - **Distribution of vessel failure frequencies ranges over three orders of magnitude**
  - **For plant lifetimes of 60-80 years, failure frequencies range from 5x10^{-10}/year to 5 x 10^{-8}/year**

Current Reevaluation Studies

ACRS Conclusions:

- **An outstanding multidisciplinary study**
- **Demonstrates utility of systematic uncertainty analyses to reach defensible conclusions in the presence of large uncertainties**

Studies (Cont'd)

- **Support staff plans for an external peer review of importance of conclusions and technical work**
- **Need to complete and improve documentation to address ACRS concerns and support peer review**

ACRS 2003 Report on NRC Safety Research

F. P. Ford

Comments on RES assessment of issues associated with Nuclear Reactor Safety for:

AP1000
ESBWR
ACR-700
GT-MHR
PBMR
IRIS

Overall Conclusions

The Infrastructure Assessment :

- **Is timely**
- **Identifies the technical issues comprehensively**
- **Defines RES-specific activities for FY03**

Long-Term RES Activities

We concur with Long-Term RES activities in the areas of:

Probabilistic Risk Assessment,
Instrumentation & Control,
Materials Analysis, Structural Analysis,
Consequence Analysis,
PIRT Process, and Implementation Issues

Long-Term RES Activities (Cont'd)

Specific comments on:

Generic Regulatory Framework, Human Factors, Thermal-Hydraulic Analysis, Neutronic Analysis, Fuel Analysis, Severe Accident & Source Term, and Advanced Computing Capabilities

Generic Regulatory Framework

Option 3 Framework is a reasonable starting point. However some concerns:

- **Need for additional risk metrics e.g., late containment failure**
- **Regulatory objectives vs. frequency/ consequences**
- **Balance between prevention and mitigation vs. uncertainties**

Human Factors Considerations

- **Plant staffing is an issue that NRC will need to address for advanced reactor plants**
- **Technical basis for judging adequacy of staffing levels must be firmly established**

Thermal-Hydraulic Analysis

- **The timely qualification and use of TRAC-M code essential to support certification decisions**
- **Significant challenges in developing confirmatory data and/or subcodes**
- **Quantification of epistemic uncertainties in thermal-hydraulic codes**

Neutronic Analysis

- **Maintain ability to conduct independent analyses**
- **Coupling of TRAC-M code with 3-D PARCS neutronics code essential for passive reactor designs**
- **Modifications to analysis methods to account for the different features of ACR-700 should be initiated now to facilitate anticipated certification review**

Severe Accident and Source Term

- **Passive ALWR covered by modified MELCOR code:**
  - **-PHEBUS-FP for high burnup fuel**
  - **-MASCA for core retention**
- **Limited NRC data and analysis to cover ACR-700 configuration**
- **Limited NRC experience in accident analysis and fission product release for HTGRs**

Fuel Analysis

- **Continue research on high burnup fuels (62 GWd/t), and extend to higher values**
- **Little NRC experience for reviewing coated-particle fuels. Initiate long-term efforts to develop capabilities using analysis methods and data available overseas**

Impact of Advanced Computer Capabilities

- **Consider the impact the increase in computer capabilities that are occurring might have on NRC efficiency and effectiveness**

ABBREVIATIONS

ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic depressurization system
ALWR	Advanced Light Water Reactor
AP1000	Advanced Passive Reactor 1000
DSER	Draft safety evaluation report
ESBWR	European Simplified Boiling Water Reactor
ESP	Early site permit
FAVOR	Probabilistic fracture mechanics code
FSER	Final safety evaluation report
FY	Fiscal Year
GT-MHR	Gas Turbine Modular Helium Reactor
GWd/t	Gigawatt day/ton
HTGR	High Temperature Gas-Cooled Reactor
IRIS	International Reactor Innovative & Safe
KTIs	Key Technical Issues

ABBREVIATIONS (Cont'd)

MASCA	Organization for Economic Cooperation and Development (OECD) experimental program for severe accident research
MELCOR	Melting of Core Program
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PHEBUS-FP	International severe accident fission product research program
PIRT	Phenomena Identification & Ranking Table
PI-groups	Symbols used in scaling analysis
PRA	Probabilistic Risk Assessment
RES	Office of Nuclear Regulatory Research
T/H	Thermal Hydraulic
TRAC-M	Transient Reactor Analysis Code-Modernized
U.S.	United States of America