ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

G. E. Apostolakis

July 10, 2002

Overview

- Core Power Uprates
- License Renewals
- Future Committee Activities

Core Power Uprates

- Recommended 5 approvals:
  - Duane Arnold Energy Center (15.3%)
  - Dresden Units 2 & 3/Quad Cities Units 1 & 2 (17%/17.8%)
  - Arkansas Nuclear One Unit 2 (7.5%)
  - Clinton Power Station Unit 1 (20%)
  - Brunswick Steam Electric Plant Units 1 & 2 (14.3%)

Uprates (Cont’d)

- **ACRS reviewed GE Topical Report** “Constant Pressure Power Uprate” **(CPPU) (4/02)**
  - **CPPU methodology applied to BWR uprates up to 20% nominal power**
  - **Committee found CPPU methodology acceptable**

Uprates (Cont'd)

- **Committee anticipates review of 4-5 uprate applications each in 2003 & 2004. Several other licensees are evaluating the feasibility of uprate applications.**

Uprates (Cont'd)

- **Committee Review Issues:**
  - **Lack of adequate documentation in staff safety evaluation reports – issue is being addressed via steadily improving documents**
  - **Need for staff guidance document on future uprate reviews–pursuant to SRM, staff is developing proposed "Review Standard"**

Uprates (Cont'd)

- Core reload safety analyses – NRR performing audits to confirm appropriate use of approved methodology
- Need for staff audit calculations/detailed T/H Models - Staff to consider as part of "Review Standard" development and related activities.

License Renewal Activities

- **Current Status**
  - **Turkey Point review complete**
  - **Reviews completed for at least one plant from each vendor**
  - **Interim letters only as needed**

License Renewal (Cont'd)

- Upcoming Reviews
  - McGuire and Catawba (1st Ice Condensers)
  - Fort Calhoun (1st Using Generic Guidance Documents)
  - North Anna/Surry/Peach Bottom/ St. Lucie
- Two License Renewal Subcommittees

Future Committee Activities

- Risk-Informed Performance-Based Regulations
- Reactor Operations (including Reactor Oversight Process, Plant Operating Events)
- Safety Research (focus on Advanced Reactors)
- Reactor Fuel (High-Burnup & MOX)

Activities (Cont'd)

- Safeguards/Security
- Fire Protection
- Transient & Accident Code Reviews
- Human Factors
- Safety Culture
- Naval Reactors

Briefing Topics

- **Advanced Reactors - T. S. Kress**

- **Risk-Informing Special Treatment**
- **Requirements of Part 50-**
  G. E. Apostolakis

- **Pressurized Thermal Shock Technical Basis Re-evaluation Project- F. P. Ford**

ADVANCED REACTORS:
A STATUS REPORT

T. S. Kress

July 10, 2002

ACRS Activities

- Two members participated in the NRC workshop on high temperature gas-cooled reactor safety and research issues
- Main topic at the 2001 ACRS retreat
- ACRS sponsored a workshop on future reactors
  - Identified 24 potential technical issues

ACRS Activities (Cont'd)

- Developing a task action plan to focus Committee review
- Advanced reactors; a main area in the next ACRS research report
- Completed review of policy and technical issues identified by Office of Regulatory Research

Overarching Policy Issues:

- Implementation of Commission's "expectation" that advanced reactors will provide enhanced margins of safety
- Relationship of NRC safety requirements to international safety requirements

Staff Technical Issues

- Event selection and safety classification
- Fuel performance and qualification
- Source term
- Containment versus confinement
- Emergency evacuation

Possible Impediments

- Lack of high-level risk-acceptance criteria other than Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)
- Lack of criteria for selecting design basis accidents
- The appropriate role of defense in depth

Current Activities

- **Priority is AP1000**
- **Working to resolve potential impediments**
- **Planning to develop “strawman” positions on various issues**

Risk-Informing Special Treatment Requirements of 10 CFR Part 50

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July 10, 2002

Risk-Informed Categorization Scheme

	Safety Related	Non-Safety Related
Safety Significant	RISC-1	RISC-2
Not Safety Significant	RISC-3	RISC-4

Previous Comments October 12, 1999 Report

- Terminology of “safety-related” Systems Structures and Components (SSCs) should be preserved
- Significance of importance measures and their limitations
- Recommended guidance to the expert panel

March 19, 2002 Report

Reviewed NEI 00-04/Rev. B

Option 2 Implementation Guideline

Recommendations

- **The criteria used by Integrated Decision-making Panel (IDP) should be explicit and include risk metrics that supplement CDF and LERF (late containment failure; inadvertent radionuclide release)**

March Report (Cont'd)

- A more complete set of risk metrics may allow the elimination of special treatment requirements for class RISC-3.
  - Difficulty in treatment of RISC-3 because risk concerns cannot be completely addressed by CDF and LERF
  - Materials degradation should be considered by IDP

March Report (Cont'd)

- Guidance to IDP should include:

  – Whether SSC acts as barrier to fission product release during severe accidents

  – Whether the SSC is relied upon in Emergency Operating Procedures or Severe Accident Mitigation guidelines

March Report (Cont'd)

– Whether failure of SSC results in an inadvertent radionuclide release

March Report (Cont'd)

- **Treatment of uncertainties in PRA results should be made consistent with the current capabilities of PRA software and data.**
- **When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods**

March Report (Cont’d)

- Use of risk information in regulations is still viewed with skepticism by some groups
- Rigor would contribute to building confidence
- Substituting “sensitivity” analysis for uncertainty analysis does not contribute to confidence building

March Report (Cont'd)

- Assessing the impact on CDF and LERF of changing the failure rates by factors ranging from 2 to 5 (in lieu of the South Texas Project factor of 10) needs better justification

PRESSURIZED THERMAL SHOCK (PTS) RULE (10CFR50.61) RE-EVALUATION

F. Peter Ford
July 10, 2002

PTS Re-evaluation

- **Need For Re-evaluation:**
  - Less frequent/better Operator performance
  - Tougher reactor vessel
  - Smaller cracks
  - Original criterion overly conservative

PTS Re-evaluation (Cont'd)

- Integrated Approach
  - Probabilistic Risk Assessment (PRA)
  - Thermal Hydraulics (T–H)
  - Probabilistic Fracture Mechanics (PFM)

PTS Re-evaluation (Cont'd)

- Application of integrated analytical process
  - Oconee Unit 1
  - Beaver Valley Unit 1
  - Palisades
  - Calvert Cliffs Unit 1

PTS Re-evaluation (Cont'd)

- **Current process versus 1980's analysis**
  - **Latest PRA/human reliability data**
  - **More refined binning**
  - **Operator action/Acts of commission**
  - **External events**
  - **More T–H sequences modeled**

PTS Re-evaluation (Cont’d)

- **Current versus 1980's analysis (continued)**
  - **Conservative bias in toughness model removed**
  - **Spatial variation influence**
  - **Smaller embedded flaws**
  - **Non−conservatisms removed**

PTS Re-evaluation (Cont’d)

- Observations
  - Primary system LOCAs dominant
  - Realistic operator action
  - Main steamline/steam generator tube rupture no longer dominant
  - Safety relief valve closure time

PTS Re-evaluation (Cont'd)

- **Ongoing work**
  - **Complete internals events analysis**
  - **External Events**
  - **Containment Integrity**
  - **Source Terms**

PTS Re-evaluation (Cont'd)

- **ACRS Conclusions**
  - **Extensive/technically sound project**
  - **Preliminary results of Oconee reactor pressure vessel (RPV) analysis indicate that the current PTS screening criterion may be overly conservative.**