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United States Nuclear Regulatory Commission

Protecting People and the Environment

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

June 9, 2010

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Overview

Said Abdel-Khalik

Accomplishments

- Since our last meeting with the Commission on December 4, 2009, we issued 16 Reports:
- Topics:
  - Draft Staff Guidance for the Use of Containment Accident Pressure
  - Status of Rulemaking for Depleted Uranium and Other Unique Waste Streams
  - Safety Research Program
  - License Renewal Application for Prairie Island Units 1 and 2

- Selected Chapters of SER with Open Items Associated with the EPR Design Certification Application
- Topical Reports
  - Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel
- Interim Staff Guidances
  - Digital I&C Systems at Fuel Cycle Facilities
  - Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) – Loss of large areas of the plant due to explosions or fires from a beyond-design-basis event

- Regulatory Guides
  - Instrument Sensing Lines
  - Risk-Informed Performance-Based Fire Protection
  - Assessment of Beyond-Design-Basis Aircraft Impacts
  - Containment Isolation Provisions
  - Terrestrial Environmental Studies
  - Manual Initiation of Protective Actions
- Standard Review Plans
  - Fuel Cycle Facility License Applications
  - Spent Fuel Dry Storage Systems

Solicitation for New Members

- Solicitation closed on April 13, 2010
- Interviews ongoing

New Plant Activities

- Reviewing design certification applications and SERs with open items associated with the US EPR and US APWR designs
- Reviewing design certification and Final SER associated with the ESBWR design

- Reviewing amendments to the AP1000 and ABWR Design Control Documents
- Reviewing the Reference COL Applications for the AP1000, ABWR, ESBWR, and US EPR designs
- Continuing to complete reviews of available material promptly

License Renewal

- Completed review of Prairie Island License Renewal Application
- Completed interim reviews of 2 applications (Cooper and Duane Arnold)
- Will perform interim reviews of 5 applications in CY 2010 (Kewaunee, Crystal River, Palo Verde, Hope Creek, and Salem)

- Will perform final reviews of 3 applications in CY 2010 (Cooper, Duane Arnold, and Kewaunee)

- Will review updates to the Generic Aging Lessons Learned (GALL) Report

Power Uprates

- Reviewed Draft Guidance for the Use of Containment Accident Pressure in Determining Available Net Positive Suction Head
- Will review the Nine Mile Point and Point Beach Extended Power Uprate Applications in CY 2010

Other Ongoing/Future Activities

- Digital I&C / Cyber Security
- Safety Culture
- Risk Metrics for New Reactors
- SOARCA
- GSI-191
- 10 CFR 50.46a
- Radiation Protection and Materials Issues
- MOX Fuel Fabrication Facility

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Risk-Informed Performance-Based Fire Protection (RG 1.205)

John W. Stetkar

10 CFR 50.48(c)

- 10 CFR 50.48(c), approved in 2004, allows licensees to adopt a performance-based Fire Protection Plan that meets the requirements of NFPA Standard 805 (2001 Edition)
- Alternative to 10 CFR 50.48(b) or the plant-specific fire protection license conditions

Plants That Do Not Adopt NFPA 805

- RG 1.189, “Fire Protection for Nuclear Power Plants,” Revision 2, issued November 2009
- Operating and new reactors
- Concepts of “safe shutdown” and “important to safety”
- Evaluation of fire-induced multiple spurious actuations

Regulatory Guide 1.205

- RG 1.205, "Risk-Informed, Performance-Based Fire Protection For Existing Light-Water Nuclear Power Plants," Revision 1, issued December 2009
- Endorses portions of Nuclear Energy Institute (NEI) 04-02, Revision 2
- Clarifications and exceptions to NEI 04-02 guidance

NEI 04-02

- "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Revision 2, issued 2008
- Transition from current Fire Protection Plan to one based on NFPA 805
- Programmatic changes
- Fire analysis guidance

Other Guidance in RG 1.205

- NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," 2005
- NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 1, 2005

ACRS Review of RG 1.205

- Selection of deterministic vs. probabilistic methods for specific fire areas
- Definitions of “manual actions,” “recovery actions,” and treatment of previously approved operator actions
- Definition of primary control station

ACRS Review of RG 1.205

- Evaluation of fire-induced multiple spurious actuations
- Application of RG 1.174 during transition to NFPA 805
- Risk-informed changes after transition to NFPA 805

ACRS Review of RG 1.205

- The staff was responsive to our comments and made several changes to clarify the risk evaluations
- Recommended issuance of RG 1.205 and associated SRP Section 9.5.1.2
- RG 1.205, Revision 1, issued December 2009

Pilot Applications of NFPA 805

- Presentations to ACRS by the pilot plants, Shearon Harris Nuclear Power Plant (Progress Energy) and Oconee Nuclear Station (Duke Energy) were very informative
- Pilot applications will improve industry and staff experience with practical implementation

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NRC SAFETY RESEARCH PROGRAM

Dana A. Powers

ACRS Review of NRC Research

- Research that supports regulatory actions brought to ACRS
- Quality reviews of selected RES projects
- Biennial review of NRC's research program

Biennial Research Program Review

- Advanced Reactors
- Digital I&C
- Fire Safety
- Reactor Fuel
- Human Factors
- Materials & Metallurgy
- Neutronics/Criticality
- Operational Experience
- PRA
- Radiation Protection
- Nuclear Mat's & Waste
- Seismic
- Severe Accidents & Source Terms
- Thermal Hydraulics
- Life Beyond 60

General Observations

- The current safety research program is working very well
  - Productive
  - Line organizations supportive
  - Enthusiastic research staff
  - Outreach to larger technical and international community

Some Areas of Note

- TRACE code is being integrated into the regulatory process
- Progress in human reliability modeling
- Seismic research has been greatly revitalized
- Fire safety research has made major strides in integrating modeling and experimental studies
- Steam Generator Action Plan completed

Some Needs

- Improving PRA methods
- Common approach to uncertainty analysis / expert opinion elicitation
- Proactive Materials Degradation Assessment initiative seems to have lost its momentum
- Follow-on to NUREG-1150

On the Horizon

- DOE initiative to apply high fidelity computer simulation to existing nuclear power plants
  - How will products of massively parallel computation interface with the regulatory process?
- Impressive research plan for the gas-cooled reactor
- Safety of reactor fuel reprocessing

Caution

- Continued degradation of nuclear safety experimental capabilities in the USA
  - Test reactors and hot cells particularly limiting
- NRC needs to consider when results of ever more complex computer code calculations must be substantiated by tests

Life Beyond 60

- Research focused on known areas
  - Vessel integrity and surveillance
  - Cable aging
  - Buried pipe
- Proactive Materials Degradation Assessment Program

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Crediting Containment Accident Pressure in the NPSH Calculations

William J. Shack

NPSH Margin

- Since 1970, NRC regulatory position has been that emergency core cooling and containment heat removal systems should be designed so that adequate NPSH is provided to system pumps assuming no increase in containment pressure from an accident
- Most reactors meet this position

Defense in Depth / Additional Safety Margin

- For defense in depth, ECCS function should not depend on containment integrity, so that an unexpected loss of containment integrity or strainer blockage would not lead automatically to core melt

Extended Power Uprates

- For some plants, demonstrating adequate NPSH for EPU operation would require:
  - Credit for all of the predicted containment accident pressure
  - Reliance on operator action to maintain NPSH
  - Reliance on CAP credit for long duration

- In some cases, pump cavitation is expected even after crediting all of the predicted accident pressure

ACRS Position on CAP Credit

- NRC should seek to maintain independence of containment function and accident mitigation and additional margin for NPSH
- Deterministic conservative calculated CAP credit for DBA should be “short and small”

ACRS Position on CAP Credit

- If hardware modifications are impractical, defense-in-depth margins can be relaxed only if associated increase in risk is small

ACRS May 19, 2010 Letter

- Addresses voluntary requests for a change in the licensing basis
- Licensees must first demonstrate that it is impractical to make plant modifications that eliminate need for CAP credit
- Plant-specific demonstration

- Staff draft guidance provides an improved framework for assessment of CAP credit. Focused on deterministic analysis of licensing-basis events. Should be complemented by plant-specific PRAs
- Support reassessment of the potential problems with operation of pumps at low NPSH

- If no CAP credit is needed for the special events licensing-basis analyses, and 95/95 statistical lower bound for LOCAs, then CAP credit is small enough to be acceptable

- Staff PRAs provide important insights. Include order-of-magnitude estimate of seismic risk, no estimate of fire risk or risk associated with operator actions to maintain CAP. Need plant-specific PRAs to address.
- Staff reluctant to request plant-specific PRA information for non-risk-informed applications (SRP 19.2 Appendix D)

- ACRS position is that CAP credit violates defense-in-depth principle of independence of barriers and 40 year old regulatory position and thus is a “Special Circumstance” that warrants request for risk information

Conclusion

- Our May 19, 2010 letter is consistent with long-standing ACRS position
- Consistent with NRC defense-in-depth philosophy that need for defense in depth is associated with uncertainty in risk

Status of Rulemaking for Disposal of Depleted Uranium

Michael T. Ryan

- In October 2005, the Commission directed the staff to consider whether DU in wastes from uranium enrichment facilities warrant amending 10 CFR 61.55(a)(6) or Tables 61.55(a) on waste classification

- The staff concluded that near-surface disposal of large quantities of DU can be appropriate, but not at all sites
- Staff recommended a limited rulemaking to revise 10 CFR 61 to require site-specific analyses that address site characteristics, proposed waste forms, and disposal methods

- In 2009 the staff held workshops in Maryland and Utah to inform the public about the rulemaking and related technical issues
- The staff is currently developing interim guidance
- The staff will respond to technical assistance requests from Agreement States

- Staff guidance should focus on key factors for a risk-informed analysis:
  - waste form
  - radionuclide quantity (not concentration)
  - geology, geochemistry, and hydrology
  - climatic conditions
  - depth of disposal
  - cover technologies

- The proximity of potentially exposed members of the public should reflect site-specific conditions, not prescribed bounding conditions
- It should be treated in a risk-informed and probabilistic fashion
- Scenarios to estimate dose to the public should be based on realistic assumptions, exposure scenarios and conditions

- The dose (and the uncertainties) to members of the public and future residents at a disposal site should be estimated over a time frame for specific sites on a case-by-case basis

- The standards by which applications will be reviewed should be clearly articulated

- Staff expectations for data supporting waste disposal requests and the quantification of uncertainties should be provided in guidance

Recommendations

- The staff should continue their efforts to risk-inform regulations for disposal of DU based on site-specific, realistic performance assessments
- Appropriate consideration should be given to uncertainties

Abbreviations

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
AP1000	Advanced Passive 1000
CAP	Containment Accident Pressure
CFR	Code of Federal Regulations
COL	Combined License
CY	Calendar Year
DBA	Design Basis Accident
DOE	Department of Energy
DU	Depleted Uranium
ECCS	Emergency Core Cooling System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
ESBWR	Economic Simplified Boiling Water Reactor
GALL	Generic Aging Lessons Learned
GSI	Generic Safety Issue
I&C	Instrumentation & Control
LOCA	Loss of Coolant Accident
MOX	Mixed Oxide
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
SBO	Station Blackout
SER	Safety Evaluation Report
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRP	Standard Review Plan
TRACE	Thermal-Hydraulic System Analysis Code
US	United States
US-APWR	United States Advanced Pressurized Water Reactor