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ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

April 5, 2018

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Overview

Mike Corradini

Accomplishments

Since our last meeting with the Commission on October 6, 2017, we issued 11 Reports

- **NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability"**
- **Revision 3 to Regulatory Guide 1.174**

Reports

- **State-of-the-Art Reactor Consequence Analysis (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses**
- **Report on the Safety Aspects of the Construction Permit Application for Northwest Medical Isotopes, LLC, Radioisotope Production Facility**
- **Biennial Review and Evaluation of the NRC Safety Research Program**

Reports

- **Safety Evaluation for Topical Report ANP-10300P, Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios”**
- **Safety Evaluation of the NuScale Power, LLC Topical Report TR-0116-20825-P, “Applicability of AREVA Fuel Methodology for the NuScale Design”**

Reports

- **Safety Evaluation for Topical Report APR1400-F-M-TR-13001, Revision 1, "PLUS7 Fuel Design for the APR1400"**
- **Safety Evaluation for ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)"**

Reports

- **Regulatory Guide 1.232: “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors”**
- **Assessment of the Quality of Selected NRC Research Projects**

Ongoing / Future Reviews

- **Design Certification**
  - **APR 1400**
  - **NuScale**
- **Early Site Permit**
  - **Clinch River**
- **Brunswick Units 1 & 2 MELLLA+**

Ongoing / Future Reviews

- **License Renewals**
  - **Seabrook**
  - **Waterford Unit 3**
  - **River Bend**
- **AP1000**
  - **WCAP assessing potential debris generation from AP1000 cables and non-metallic insulation (GSI-191)**

Ongoing / Future Reviews

- **Guidance and Bases**
  - **Draft Regulatory Guide DG-1327, Reactivity-Initiated Accidents**
  - **NUREG on High Burnup Fuel Storage and Transportation**
  - **NUREG/BR-0058**
- **Advanced Reactors**
  - **Licensing Modernization Framework**
  - **Functional Containment Policy Paper**

Ongoing / Future Reviews

- **Digital I&C**
  - **ISG-06 Revision**
  - **Diversity and Defense-in-Depth against Common Cause Failure**
  - **Integrated Action Plan**
- **Rulemaking**
  - **Emergency Preparedness for SMRs**
  - **Non-Power Production or Utilization Facility**

Ongoing / Future Reviews

- **Thermal-Hydraulic Phenomenology**
  - **GSI-191**
    - **PWR Owners Group In-vessel Debris Test Results**
  - **AREVA's AURORA-B Transient Code Suite: LOCA**
- **Metallurgy and Reactor Fuels**
  - **Consolidation of Dry Cask and Dry Fuel Storage Standard Review Plans**

Ongoing / Future Reviews

- **Reliability and PRA**
  - **Level 3 PRA**
  - **Human Reliability Analysis Method Development**
    - **IDHEAS program**
    - **Control Room Abandonment Risk**

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NuScale Power Exemption Request From 10 CFR Part 50, Appendix A, General Design Criterion 27

Michael Corradini

Background

- **The General Design Criteria are the minimum requirements for principle design criteria for water-cooled nuclear plants to provide reasonable assurance that the facilities can be operated safely**
- **GDC's were based on the licensing of early commercial water-cooled reactor plant designs**
- **Staff has acknowledged that fulfillment of some of the GDC may not be necessary or appropriate for some designs**
- **NuScale reactor is a modular, passive, water-cooled reactor design with innovative design features**

Background

GDC 27, “Combined Reactivity Control Systems Capability”

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained

Background

- **Staff has historically interpreted the intent of GDC 27 to require that the reactor:**
  - **Be reliably controlled in normal operation**
  - **Achieve and maintain a safe shutdown condition, including subcriticality beyond the short-term, using only safety-related equipment following a DBE with margin for stuck rods**
- **Staff informed NuScale that an exemption would be required for its reactor design**

Requested Exemption

- NuScale submitted a request for an exemption to GDC 27
- Staff plans to evaluate whether the NuScale design meets the underlying intent of the GDC and assures public health and safety are maintained based on two criteria:
  - Demonstrate sufficient core cooling
  - DBE sequence of events is not expected to occur during the lifetime of a module

Maintain Long-Term Cooling

- **To assure long-term core cooling, we expect that NuScale will perform an evaluation to ensure SAFDLs are not exceeded for any of the DBE scenarios considered. Analyses would include:**
  - **Consideration of operator actions**
  - **Estimates of the return to power and associated strategies to return to a subcritical condition**
  - **Assurance that the margin does not degrade over the duration of the event**

Low Probability of Return to Power

- **The staff evaluation criteria should be augmented to include:**
  - **An assessment of the incremental risk to public health and safety from the hypothesized situation**
  - **Whether that risk increase is acceptable, considering the entire NuScale facility**

Low Probability of Return to Power

- **Non-safety SSCs that provide boron addition should have certain characteristics**
  - **They should not degrade during plant operations**
  - **They should function reliably when called upon, including operator actions needed for their startup and alignment**

ACRS Conclusion and Recommendation

The proposed criteria are reasonable provided the following recommendations and enhancements outlined in the letter report are addressed:

1. **Evaluate the overall risk and not just the frequency of the challenge**
2. **Risk considerations should be based on the facility rather than an individual module**

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Revision 3 to Regulatory Guide 1.174

John Stetkar

Background

- **Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"**
- **Describes key principles and guidance for the use of risk information in regulatory decisions**
- **Primary intent of Revision 3 to clarify guidance for considering defense-in-depth**

ACRS Engagement

- **SRM for SECY-15-0168 directed staff to issue Revision 3 expeditiously**
- **Four Subcommittee meetings from May 2016 to August 2017**
- **ACRS previously reviewed evolution and interpretation of the defense-in-depth philosophy (NUREG/KM-0009) during the staff's evaluation of issues for implementation of a proposed Risk Management Regulatory Framework**

ACRS Recommendation

- **Revision 3 of Regulatory Guide 1.174 should be issued***
  - **Substantially expands and clarifies the guidance for consideration of defense-in-depth and its integration with the other risk-informed decision-making principles**
  - **Clarifies the staff's intent for determining acceptability of a PRA for use in risk-informed decisions**
  - **Enhances the guidance on evaluation and treatment of uncertainties**

*** Issued in January 2018.**

Future Revisions

- **Plans to expand the guidance on integrated decision-making and the use of uncertainty as an input to the decision process**
- **Encourage staff to also consider extending the guidance to address applications of risk information for new reactors, which may have much different risk profiles and lower overall levels of risk than currently operating reactors**

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STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES (SOARCA) PROJECT

SEQUOYAH INTEGRATED DETERMINISTIC AND UNCERTAINTY ANALYSES

John Stetkar

Background

- **Original Peach Bottom and Surry SOARCA studies reported only "point estimate" results, without an evaluation of the uncertainties in those estimates**
- **Subsequent to the original studies, focused uncertainty analyses were performed for selected scenarios at Peach Bottom and Surry**

Background

- **In most cases, the uncertainties were retrofit around the "point estimate" values used in the original studies**
- **These focused studies provided important insights about how consideration of the uncertainties affects understanding and interpretation of the results**

Background

- **Sequoyah study extends the scope of the SOARCA analyses to include a focused evaluation of severe accident response for a PWR with an ice condenser containment**
- **It is intended specifically to examine the effects from hydrogen generation and release, timing and locations of ignition, and containment vulnerability to failure caused by a highly energetic deflagration**

Background

- **Sequoyah study evaluates responses to one short-term and one long-term station blackout scenario**
- **Assumes each scenario is caused by a severe earthquake**
- **Offsite emergency response models account for infrastructure damage**
- **Integrated evaluation of uncertainties for thermal-hydraulic response and offsite consequences for only the short-term blackout scenario**

ACRS Engagement

- Three Subcommittee meetings on Sequoyah SOARCA study between May 2016 and October 2017
- April 2017 joint Subcommittee meeting on changes to thermal-hydraulic models and analyses in MELCOR

ACRS Conclusions and Recommendations 1

- **Sequoyah SOARCA study has significantly advanced the understanding of severe accident progression in a PWR with an ice condenser containment**
- **It demonstrates the importance of an integrated assessment of uncertainties about equipment performance, thermal-hydraulic phenomena, and emergency planning**

ACRS Conclusions and Recommendations 2

- **Study evaluates site-specific conditional consequences from two station blackout scenarios, tailored to examine the effects from hydrogen generation, ignition, and containment failure vulnerability**
- **It does not examine other scenarios that may be important for containment failure or bypass**

ACRS Conclusions and Recommendations 2

- **Study does not account for accident mitigation strategies that have been implemented at Sequoyah**
- **Results from the study should not be extrapolated to other PWRs with ice condenser containments at other sites**

ACRS Conclusions and Recommendations 3

- **Sequoyah SOARCA report should be published after the staff more clearly documents the following issues and research needed for their resolution:**
  - **Potentially important modeling uncertainties**
  - **Justification for safety valve failure rates**
  - **Failures to complete some MELCOR simulations involving an early stuck-open pressurizer safety valve**

ACRS Conclusions and Recommendations 4

- **Staff should examine and resolve the issues regarding safety valve failure rates and MELCOR performance before further enhancements are made to the SOARCA studies**

Staff Responses

- **Report emphasizes that study is specific to Sequoyah**
- **Evaluation of model uncertainty is outside the study scope**
- **Report contains enhanced discussions of safety valve failure rates and insights from incomplete MELCOR runs**
- **Will address safety valve failure rates in updated Surry uncertainty analyses**

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Report on the Safety Aspects of the Construction Permit Application for Northwest Medical Isotopes, LLC, Radioisotope Production Facility

Dana Powers

Background

- **Northwest Medical Isotopes, LLC (NWMI) submitted a preliminary design for a facility that addresses hazards associated with the extraction of ^{99}Mo from irradiated targets and the fabrication of targets for irradiation**

Background

- **ACRS reviewed the preliminary safety analysis report submitted by NWMI and the draft final safety evaluation report prepared by the NRC staff**

Conclusions and Recommendations

- **Once the design is finalized, the proposed facility can be constructed and licensed for operation with adequate protection of the public health and safety and no undue risk to the environment**

Conclusions and Recommendations

- **A construction permit for the proposed radioisotope production facility can be issued to NWMI**

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Biennial Review and Evaluation of the NRC Safety Research Program

Joy Rempe

Background

- **2018 biennial review addresses 1997 Commission guidance:**
  - **Need, scope, and balance of reactor safety research program**
  - **Progress of ongoing activities**
  - **How well RES anticipates research needs and how it is positioned for changing environment**

Background

- **2018 biennial review also emphasizes:**
  - **Prioritization and identification of new research needs**
  - **Long-term planning**
  - **More succinct ACRS report**

Background

- **2018 biennial report developed using insights from:**
  - **Initial meeting with RES Director to obtain overview of program, plans, priorities, and areas of interest**
  - **Three working group meetings to discuss research conducted by each RES division: Division of Risk Analysis (DRA), Division of System Analysis (DSA), and Division of Engineering (DE)**
  - **Other ACRS activities**

DRA Review Findings

- **Division-specific:**
  - **Level 3 PRA**
  - **IDHEAS**

DRA Review Findings

- **General:**
  - **It is not clear how research priorities account for integrated consideration of 'enterprise risk', which addresses factors such as safety and security, emerging issues, innovative technologies and associated uncertainties, preservation of core competencies, and development and maintenance of analysis methods and tools**

DSA Review Findings

- **Division-specific:**
  - **The agency must have an independent reactor safety analysis capability**
  - **Consequence analysis should be a core competency**

DSA Review Findings

- **General:**
  - **Difficult strategic choices needed to maintain current computational capabilities and core competencies and to anticipate and adapt to future regulatory needs**

DE Review Findings

- **Division-specific:**
  - **Material performance computer code development**
  - **Codes and standards review**
  - **Risk evaluation needed prior to embarking on spent fuel dry storage cask research**

DE Review Findings

- **General:**
  - **Rather than developing data independently, efforts should focus on identifying data that licensees or applicants must provide**
  - **An effective process should be developed for terminating ongoing research that ceases to be high priority**

Conclusion and Recommendations

- **NRC safety research program appears to be meeting near-term agency needs satisfactorily**
- **Current process to prioritize agency research could be improved by performing a systematic assessment that emphasizes 'enterprise risk' in research project selection, evaluation, and termination**

Conclusion and Recommendations

- **RES should develop long-term strategies to address emerging technical issues, support development and maintenance of needed analytical tools and data bases, emphasize activities that improve regulatory efficiency, and identify and preserve needed core competencies**

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
CFR	*Code of Federal Regulations*
CRDA	Control Rod Drop Accident
DBE	Design Basis Event
DG	Draft Regulatory Guide
GDC	General Design Criteria
GSI	Generic Safety Issue
I&C	Instrumentation and Control
IDHEAS	Integrated Human Event Analysis System
ISG	Interim Staff Guidance
LOCA	Loss-of-Coolant Accident
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
^{99}Mo	Molybdenum 99
NWMI	Northwest Medical Isotopes, LLC
PRA	Probabilistic Risk Assessment
PWR	Pressurized-Water Reactor
RES	Office of Nuclear Regulatory Research
SAFDL	Specified Acceptable Final Design Limit
SECY	Secretary of the Commission
SER	Safety Evaluation Report
SMR	Small Modular Reactor
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRM	Staff Requirement Memorandum
SSC	Structure, System, and Component