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United States Nuclear Regulatory Commission

Protecting People and the Environment

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

October 6, 2016

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Overview

Dennis C. Bley

Accomplishments

Since our last meeting with the Commission on March 4, 2016, we issued 15 Reports

- **Non-Power Production or Utilization Facilities License Renewal Rulemaking**
- **Fukushima: Interim Staff Guidance, JLD-ISG-2016-01, "Guidance for Activities Related to Near-Term Task Force Recommendation 2.1, Flooding Hazard Reevaluation; Focused Evaluation and Integrated Assessment"**

Reports

- **NuScale Licensing Topical Report, "Risk Significance Determination"**
- **Draft Final Regulatory Guide 1.230, "Regulatory Guidance on the Alternative Pressurized Thermal Shock Rule," and Draft Final Report NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternative Pressurized Thermal Shock Rule"**

Reports

- **COLAs**
  - **Turkey Point Units 6 and 7**
  - **Exemptions to the AP1000 Certified Design Included in the Levy Nuclear Plant Units 1 and 2 Combined License Application**
- **License Renewal Applications**
  - **LaSalle County Station Units 1 and 2**
  - **Fermi 2**

Reports

- **Fukushima**
  - **Closure of Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation**
  - **Updated Assessment of Tier 2 Recommendations Related to Evaluation of Natural Hazards Other Than Seismic and Flooding**

Reports

- **Guidance and Bases**
  - **Regulatory Guide 1.229, "Risk-Informed Approach for Addressing the Effects of Debris on Post Accident Long-Term Core Cooling"**
  - **NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel"**

Reports

- **Topical Report WCAP-16996-P, "Realistic Loss-of-Coolant Accident Evaluation Methodology Applied to the Full Spectrum of Break Sizes"**
- **Diablo Canyon Power Plant Units 1 and 2 Digital Replacement of the Process Protection System**
- **Biennial Review and Evaluation of the NRC Safety Research Program**

Visits

- Site and Region Visit
  - Vogtle Units 3&4
  - Vogtle Units 1&2
  - Region II
- AREVA Fuel Fabrication Facility

Ongoing / Future Reviews

- **Fukushima**
  - **Evaluations of Natural Hazards other than Seismic and Flooding**
  - **Mitigation of Beyond-Design-Basis Events Rulemaking**
- **Radiation Protection**
  - **10 CFR Part 61 Rulemaking**

Ongoing / Future Reviews

- **Design Certification**
  - **APR 1400**
- **COLA**
  - **North Anna (ESBWR)**
- **NuScale Safety-Focused Review**
- **License Renewal**
  - **Grand Gulf**
  - **South Texas Project Units 1 and 2**

Ongoing / Future Reviews

- **GSI-191**
  - **WCAP Related to GSI-191 Debris Issues**
  - **PWR Owners Group In-vessel Debris Test Results**
  - **South Texas Project Risk-Informed License Amendment Request**

Ongoing / Future Reviews

- **Digital I&C**
  - **SECY Paper on Cyber Security for Fuel Cycle Facilities**
  - **10 CFR 50.59 Guidance**
- **Reliability and PRA**
  - **Level 3 PRA**
  - **Human Reliability Analysis Methods**

Ongoing / Future Reviews

- **Metallurgy and Reactor Fuels**
  - **Consequential Steam Generator Tube Rupture**
  - **Consolidation of Dry Cask and Dry Fuel Storage Standard Review Plans**
- **Thermal-Hydraulic Phenomenology**
  - **AREVA Extended Flow Window (Monticello)**
  - **Supplement to Topical Report on BISON code**

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Non-power Production or Utilization Facility (NPUF) License Renewal Rulemaking

Dana A. Powers

Class 104 a, c Reactors

- **Research reactors and Test Facilities**
- **31 operating facilities**
  - **Most in universities (25)**
  - **Often the distance to the ‘public’ is small**
- **Typically**
  - **Low radionuclide inventory**
  - **Unpressurized**
  - **Natural cooling**

Low Power Reactors

- 4 < 1kW
- 1kW < 12 < 1 MW
- 1 MW < 10 < 2 MW
- 5 > 2 MW

Low Usage

- **4 used a few hours per year**
- **16 used a few hours per week**
- **7 used for 20-40 hours per week**
- **4 have high usage level – 24/7**

- **Aging of facilities is very slow**
- **Few design changes**

Accorded Special Consideration by Atomic Energy Act

- **Minimal regulation consistent with Commission obligations to protect public health and safety**
- **20 year license period**

Novel Approach from Staff

- **Licenses for research reactors don’t expire**
- **Updated final safety analysis report submitted every five years**
- **Continued program of inspection and monitoring**

ACRS Concluded

- **Non-expiring license would not degrade safety**
- **Similar conclusion on other changes**
  - **Accident dose criterion increased to 1 rem consistent with Protective Action Guidelines**
  - **10 CFR 50.59 applicable regardless of decommissioning status**
  - **Timing for submission of license renewal applications for test facilities and irradiation facilities**

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Guidance for Flooding Hazard Reevaluation; Focused Evaluation and Integrated Assessment

John W. Stetkar

COMSECY-15-0019

- **Focused evaluations confirm that key safety functions are protected by existing barriers and equipment or by plant modifications**
- **Integrated assessments evaluate plant-specific protection and mitigation strategies**
- **Revised integrated assessment of local intense precipitation (LIP) is not required**

FLEX Strategies

- **Industry developed guidance for assessing FLEX strategies**
- **Licensee may consider alternate or targeted mitigating strategy to compensate for limitations**
- **JLD-ISG-2016-01 endorses NEI 16-05**
  - **Paths 1-3: Focused Evaluations**
  - **Path 4: Effective Mitigation**
  - **Path 5: Scenario-Based**

Integrated Assessments

Evaluation Options

- **Path 1: Refined analysis of flooding parameters; bounded by licensing basis**
- **Path 2: Demonstrate adequate physical margin for protection of key safety functions**
- **Path 3: Applies only to LIP; protection of key safety functions or mitigation of damage**
- **Path 4: Strategies to mitigate flooding damage; primarily consider flooding severity**
- **Path 5: Strategies to mitigate flooding damage; consider scenario-specific flooding frequency and severity**

Conclusions and Recommendations

- **Graded approach provides an appropriate evaluation framework**
  - **Focused evaluations emphasize protection against flooding damage**
  - **Mitigation strategies examined only if protection cannot be assured**
  - **Supports defense-in-depth approach to safety**

Conclusions and Recommendations

- **Treatment of LIP**
  - **If mitigation strategies are needed for flooding caused by LIP, the staff should review those evaluations in the same manner as the integrated assessments that are performed for other flooding mechanisms**

Conclusions and Recommendations

- **Reliability of mitigation strategies**
- **Path 4 and higher-frequency Path 5 assessments**
  - **Guidance for equipment is very good**
  - **Guidance for personnel performance is weak, by comparison**
  - **Staff should better specify expectations for assurance of reliable personnel performance**

Conclusions and Recommendations

- **Evaluation of seismically-caused floods**
  - **Strong seismic event that causes damage to site and nearby dams**
  - **Strategies that are targeted to only one hazard could be compromised**
  - **Staff should develop guidance to ensure evaluation of coupled seismic and flooding scenarios**

Conclusions and Recommendations

- Independent peer reviews
  - Staff recommended an independent peer review be performed for integrated assessments
  - Conducting these reviews would be challenging
- Guidance has been revised; detailed peer reviews are not needed for all assessments

Continuing Engagement

- Fukushima Subcommittee briefed on draft guidance for Phase 2 regulatory decision-making (August 17, 2016)
- Requested future briefings on selected site-specific evaluations

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NuScale Licensing Topical Report, “Risk Significance Determination”

Michael Corradini

Background

- **NuScale Design Certification Application expected in December 2016**
- **Lower risk profile of NuScale iPWR design than current LWRs**
- **Estimated CDF and LRF values are much lower than current operating NPPs.**

Background

- **A component or system is risk significant if an assumed failure causes a notable increase in CDF**
- **Current risk significance criteria in RG 1.200 would overstate the importance of SSCs for a plant with low risk**
- **For NuScale, this would result in categorizing a majority of NuScale equipment modeled in the PRA as risk-significant**

Background

- **NuScale Approach**
  - **Alternative approach to RG 1.200 for identifying SSCs as candidates for risk-significance follows a framework similar to RG 1.174**
- **NuScale Risk Significance Determination Methodology**
  - **Criteria for candidate SSC risk significance – a fixed contribution to CDF and LRF**

Background

- **ACRS reviewed NuScale Licensing Topical Report and issued letter in May 2016**

ACRS Conclusions and Recommendations

- **Criteria for determining risk significance in a case-by-case manner can lead to inconsistencies in regulatory positions**
- **Staff should develop a consistent approach by adopting a continuous scale to determine quantitative risk significance criteria, with more margin allowed for plants with lower risk**

ACRS Conclusions and Recommendations

- **NuScale approach is reasonable provided CDF or LRF remains consistent with their current estimates**
- **Staff will need to address multi-module aspects of NuScale design that could alter CDF and LRF risk estimates and associated SSCs classification**

Staff Response to ACRS Recommendations

- **Staff agrees that generic numerical criteria for determining risk significance would be advantageous rather than case-by-case criteria**
- **Staff intends to pursue revision of quantitative risk significance criteria to make them consistent with a broad spectrum of designs and absolute levels of overall plant risk**

Staff Response to ACRS Recommendations

- **Numerical criteria will be scalable based on applicable base risk metrics (i.e., CDF, LRF, and LERF)**
- **Numerical criteria will be anchored to thresholds for risk significance that conform with acceptable risk increase guidelines in RG 1.174**
- **Criteria would complement existing criteria in RG 1.200 being used by current operating plants**

Staff Response to ACRS Recommendations

- **Staff will draft a single guidance document for using PRA to rank SSCs by risk**
- **Staff will consider revising existing guidance documents as resources permit**

Staff Response to ACRS Recommendations

- **Staff agrees with ACRS recommendation on multi-module aspects of NuScale design**
- **Staff will consider impact of multi-module aspects of NuScale design on CDF and LRF and on categorization of SSCs**
- **Staff will consider this as part of its review of NuScale design certification application, Section 17.4, “Reliability Assurance Program”**

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Guidance on the Alternative Pressurized Thermal Shock Rule

Ronald Ballinger

Background

- **Original rule (10 CFR 50.61) contains screening limits for prevention of RPV failure due to thermal shock during LOCA event**
- **Alternative rule (10 CFR 50.61a) was issued in 2010 and provides alternative limits based on probabilistic fracture mechanics (PFM) analysis (frequency of vessel failure < 10^{-6} per year)**
- **NUREG-2163 and Regulatory Guide 1.230 provide guidance on use of alternative rule**

10 CFR 50.61a

- **Less restrictive reference temperature (embrittlement) screening criteria enable longer operations**
- **Criteria must be satisfied to use the alternative rule**
  - **Evaluation of plant-specific surveillance data**
  - **Evaluation of inservice inspection data**

Motivation

- **Original screening criteria resulted in unnecessary burden without improving overall plant safety**
- **Conservative bias in toughness resulted in artificial impediment to license renewal**
- **Plant specific analysis was an option if original screening criteria could not be met but was found to be impractical**

Improvements in Technical Understanding

- **Spatial variation in fluence recognized**
- **Most flaws now recognized as embedded rather than on the surface**
- **Spatially dependent embrittlement properties**

10 CFR 50.61a

	10 CFR 50.61	10 CFR 50.61a Voluntary
Reference Temperature Screening Criteria	More restrictive	Better informed, Less restrictive
Plant-specific surveillance data check	Required 1 test	Required 3 tests
Plant-specific flaw inspection	Not required	Required

Plant-Specific Surveillance Checks

- **Ensures that surveillance data for the plant being assessed is well represented by the embrittlement trend equation used in PFM analysis**

Guidance on Plant-Specific Flaw Inspections - NDE

- **Assures that actual flaw distribution is bounded by data base used in PFM model**
  - **Qualified examination – ASME Code, Section XI**
  - **Verification that flaws at the clad/base metal interface do not open to the RPV inside surface**
  - **NDE uncertainty addressed**
  - **Flaws closer to the ID are assessed more stringently**

Recommendation

- **Regulatory Guide 1.230 and NUREG-2163 should be issued**

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
CDF	Core Damage Frequency
COLA	Combined Operating License Application
CFR	Code of Federal Regulations
GSI	Generic Safety Issue
I&C	Instrumentation & Control
ID	Internal Diameter
iPWR	Integral Pressurized Water Reactor
ISG	Interim Staff Guidance
kW	Kilowatt
LERF	Large Early Release Fraction
LIP	Local Intense Precipitation
LOCA	Loss of Coolant Accident
LRF	Large Release Frequency
LWR	Light Water Reactor
MW	Megawatt
NEI	Nuclear Energy Institute
NDE	Non-destructive Examination
NPP	Nuclear Power Plant
NPUF	Non-Power Production or Utilization Facility
NRC	Nuclear Regulatory Commission
PFM	Probabilistic Fracture Mechanics
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SECY	Office of the Secretary
SSC	Structure, System or Component