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United States Nuclear Regulatory Commission

Protecting People and the Environment

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

July 11, 2013

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Overview

J. Sam Armijo

Accomplishments

- *Since our last meeting with the Commission on December 6, 2012, we issued 16 Reports*

- *Topics:*
  - *Draft Design Specific Review Standard for mPower iPWR Chapter 7 Instrumentation and Control Systems*

- ***Topics (cont.):***
  - ***Station Blackout Mitigation Strategies Rulemaking***
  - ***Next Generation Nuclear Plant Key Licensing Issues***
  - ***Draft NUREG-2125, "Spent Fuel Transportation Risk Assessment"***
  - ***Construction Reactor Oversight Process Program and Pilot Program Results***

- ***Topics (cont.):***
  - ***Draft Revision 1 to NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-informed Decisionmaking"***
  - ***Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel***

- ***Topics (cont.):***
  - ***Selected Chapters of the Safety Evaluation Report with Open Items for the US-APWR Design Certification and Safety Evaluations of Supporting Technical Reports***
  - ***Selected Chapters of the Safety Evaluation Report with Open Items for the Comanche Peak Nuclear Power Plant, Units 3 and 4, US-APWR Reference Combined License Application***

- ***Topics (cont.):***
  - ***WCAP-17116-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 5 – Application to ABWR"***
  - ***Report on the Safety Aspects of the License Renewal Application for the Limerick Generating Station***

- ***Topics (cont.):***
  - ***Revision 1 to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program"***
  - ***Regulatory Guide 4.22, "Decommissioning Planning During Operations"***
  - ***Regulatory Guides 1.168 – 1.173, Software Processes for Digital Computers in Safety Systems of Nuclear Power Plants***

Ongoing/Future Reviews

New Plants:

- Design Certification applications and SERs associated with the EPR and US-APWR designs
- Adequacy of Long-Term Core Cooling Approach for the US-APWR and EPR
- Reference COLAs for ABWR, ESBWR, US-APWR, and EPR
- Subsequent COLAs for AP1000

Ongoing/Future Reviews

License Renewals :

- ***Interim and final reviews for Grand Gulf, South Texas, Callaway, and Sequoyah***

- ***Final reviews for Diablo Canyon, Seabrook, and Davis Besse***

Ongoing/Future Reviews

Extended Power Uprates:

- Browns Ferry 1, 2, & 3
- Peach Bottom 2 & 3
- Monticello

Ongoing/Future Reviews

Other:

- Spent Fuel Pool Study
- Fukushima Longer-Term Efforts (e.g., Recommendation 1, Station Blackout Rule, Tier 3 Recommendations)
- Revisions to 10 CFR Part 61
- Uncertainties in SOARCA Analysis
- Watts Bar 2
- Fire Modeling Applications
- Naval Reactors: Gerald Ford Class

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Draft Design Specific Review Standard for mPower iPWR Chapter 7 Instrumentation and Control Systems

Charles H. Brown, Jr.

Fundamental Principals of Instrumentation Safety and Reliability + 1

- ***Redundancy***
- ***Independence***
- ***Determinancy***
- ***Defense and Depth/Diversity***

  +

- ***Simplicity***

Fundamental Principles (cont.)

- ***Nuclear Plants are being designed with computer based DI&C systems and networks as the backbone for protection, control, alarm, display, and monitoring***

Fundamental Principles (cont.)

- **Computer based systems allow enhanced performance but:**
  - **result in a higher degree of functional integration and**
  - **have new design and failure issues; e.g., less inherent inter-division communication independence, non-inherently deterministic processing, software complexity and V&V**

Fundamental Principles (cont.)

- ***Also, networks are used for communication between plant systems and control spaces and to external site and corporate networks resulting in potential compromised control of access from external plant networks***

Fundamental Principles (cont.)

- ***Thus, use of computer based systems need new design features that ensure:***
  - ***the fundamental principles are embodied and captured in the DI&C architecture particularly Independence,***
  - ***that one-way non-software based hardware for data transmission to external networks is part of the basic DI&C architecture, and***
  - ***both are detailed during the licensing phase***

mPower DSRS Chapter 7

- ***The Office of New Reactors has begun to develop Design Specific Review Standards for small modular integral PWR designs starting with mPower to streamline and improve review quality and efficiency***

mPower DSRS Chapter 7 (cont.)

- ***Licensing reviews of digital-based I&C systems have been a significant challenge***
- ***The DSRS Chapter 7 goal is to apply lessons from recent reviews of DI&C systems and develop a review standard for the mPower SMR design that enhances the focus on Fundamental Safety Principles***

mPower DSRS Chapter 7 (cont.)

- ***The DSRS reorganizes the existing standard review plan from a bottom-up system-by-system approach, where regulatory requirements and principles are repeated multiple times, to a top-down approach which focuses on ensuring the basic architecture of the DI&C systems:***
  - ***meets the Fundamental Design Principles,***
  - ***provides guidance on the Fundamental Design Principles, and***
  - ***then assesses design characteristics and regulatory requirements within each system***

ACRS Comments and Recommendations

- ***The Control of Access section of the DSRS should be revised to ensure non-software based one-way external communication is part of the basic hardware architecture***
- ***With the above exception, the DSRS Chapter 7 is a significant and innovative approach to revising the Standard Review Plan for future I&C designs***

ACRS Comments and Recommendations (cont.)

- ***Although an mPower pilot initiative, the DSRS is likely applicable to large reactor designs as well as other SMRs***
- ***We are working with staff to resolve our recommendations***

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Station Blackout Mitigation Strategies Rulemaking

W. J. Shack

ACRS Reviews

- ***Subcommittee Meetings on December 5, 2012, and April 23, 2013***
- ***Committee completed review during June 2013 meeting***

Background

- ***Station blackout involves loss of all offsite and onsite ac power. (dc power is assumed available).***
- ***Current station blackout rule (10 CFR 50.63) requires that all plants be able to cope and recover from station blackout***
- ***Most plants can cope for 4 to 8 hours***

Background (cont.)

Station Blackout Rule:

- ***Scope is limited to switchyard, grid, weather related events***
- ***External events (e.g., fire, flood, seismic) not specifically addressed by the rule***
- ***Alternate ac source can be credited for coping with station blackout***

Background (cont.)

- ***Fukushima accident demonstrated that other aspects need to be considered:***
  - ***External events beyond the design basis***
  - ***Extended station blackout conditions***
  - ***Impact on multiple units at the same site***
  - ***Failure of the alternate ac source***
  - ***Spent fuel pool cooling during extended SBO***
  - ***Station blackouts that occur during any mode of operation***

SBO Mitigation Strategies Rule

- ***Proposed approach is consistent with Order EA-12-049***
- ***Provides significant increase in defense-in-depth beyond the current SBO rule 10 CFR 50.63***
- ***Would address limitations in the current rule***

ACRS Conclusions and Recommendations

- ***Sufficient regulatory basis for mitigation strategies rulemaking***
- ***Robust supplemental ac power source should be explored further***
- ***Guidance needed for evaluating feasibility and reliability of manual actions***

ACRS Conclusions and Recommendations (cont.)

- ***Staff should consider the results from the ongoing integrated assessments of external hazards to determine if the available margins for these hazards are adequate for the development of mitigating strategies***

ACRS Conclusions and Recommendations (cont.)

- ***Loss of decay heat removal as a separate condition and not just as a consequence of extended loss of ac power should be considered in efforts on NTTF Recommendation 1 and the RMTF program development***

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Next Generation Nuclear Plant (NGNP) Key Licensing Issues

Dennis Bley

Background

- ***Energy Policy Act of 2005 directed DOE to establish and manage NGNP project with INL as lead laboratory***
- ***The Act stipulated that NRC has licensing and regulatory authority for any reactor developed by the project***

Background (cont.)

- ***NGNP reactor technology: DOE selected a high-temperature gas-cooled reactor***
- ***NGNP Licensing Strategy Report: joint DOE/NRC submission to Congress described four licensing options***
  - ***Option 2: risk-informed, performance-based using engineering judgment and analysis to establish licensing basis and technical requirements***

Background (cont.)

- ***Option 2***
  - ***Design-specific PRA to be used to help select LBEs and guide special treatment of SSCs***
  - ***Adapt current regulations and guidance, as needed***

Background (cont.)

- ***INL white papers address key issues highlighted in the joint report***
- ***Staff reviewed white papers and issued draft assessments and summary report on licensing issues***
  - ***NGNP approach generally reasonable with several caveats***

Staff caveats

- ***Lack of detailed design information and incomplete testing preclude firm conclusions***
- ***Expect issues could be resolved during pre-application interactions, as information becomes available***
- ***Not clear that all service conditions during possible accidents have been considered***
- ***Additional fuel testing is needed and is in progress***

INL Approach to Option 2

- Design-specific PRA to be used to select LBEs (AEs, DBEs, BDBEs)
- DBAs derived from DBEs
- DBAs must meet deterministic criteria similar to current practice
- PRA guides special treatment of SSCs
- Adapt current regulations and guidance, as needed

ACRS Letter

- **Conclusion**
  - Staff assessment of white papers is appropriate

- **Recommendations**
  1. Staff assessment documents should be revised to provide clear links to RAIs and responses

ACRS Letter (cont.)

- ***Recommendations (cont.)***
  2. ***Staff LBE selection assessment should point out need to clarify definition of event sequences and event sequence families to ensure consistency in developing LBEs and DBAs***

ACRS Letter (cont.)

- ***Recommendations (cont.)***
  3. ***Staff suggestion that final selection of DBAs include postulated deterministic event sequences is inconsistent with risk-informed framework proposed for NGNP and other ongoing NRC activities encouraged by the Commission; any such sequences should be included in PRA so they be considered for inclusion as DBAs***

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DRAFT NUREG-2125, "SPENT FUEL TRANSPORTATION RISK ASSESSMENT"

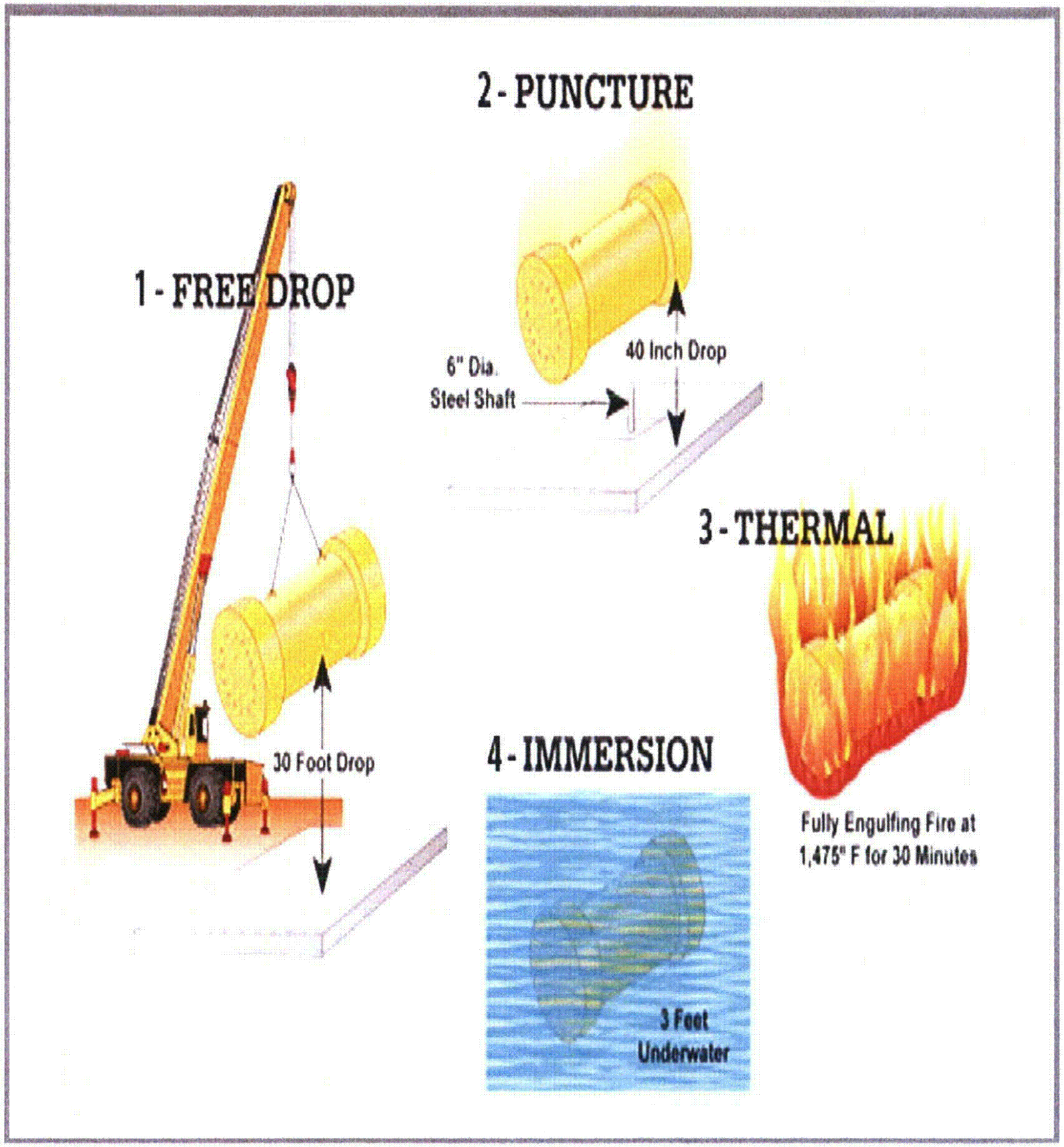
Michael T. Ryan

BACKGROUND

- ***The staff has conducted and published a series of studies evaluating the risks associated with the transportation of SNF in casks***
- ***NUREG-2125 documents the evaluations of risks associated with SNF shipments by rail or highway***

USE OF CERTIFIED CASKS

- ***Prior generic risk assessments have used generic casks***
- ***This assessment uses casks that have been certified to meet the requirements of 10 CFR Part 71***



The Study

- ***Improved event trees were used to estimate the probabilities of accident conditions***
- ***Fire scenarios were extended to consider very low frequency events***
- ***The flammable pool area is conservatively sized***

The Study (cont.)

- ***Finite element analyses were performed to analyze how the casks responded to impact and thermal challenges under accident conditions***
- ***NUREG-2125 also includes an assessment of consequences involving criticality***

The Study found

- ***The collective doses from routine SNF shipments were 10,000 to 100,000 times less than collective background radiation doses***
- ***Little variation in the risks per kilometer of transport distance over the routes analyzed***

The Study found (cont.)

- ***No release of radioactive material in any of the accident scenarios with welded stainless steel canisters***
- ***Accidents involving rail casks without inner welded canisters could result in release of very small amounts of radioactive material***

ROUTINE TRANSPORTATION SUMMARY

- ***Individual and collective doses are calculated for a single shipment and are very small***
- ***Maximum individual doses are comparable to background doses***
- ***Collective doses from routine transportation are orders of magnitude less than the collective background dose***

SFTRA CONCLUSIONS

- ***This study reconfirms that estimated radiological risks from spent fuel transportation conducted in compliance with NRC regulations are low, in fact generally less than previous estimates, which were already low.***

ACRS RECOMMENDATION AND CONCLUSIONS

1. ***Despite the lack of a systematic assessment of a broader range of phenomena that could occur in accidents, the results in NUREG-2125 continue to support the conclusion that risks from accidents involving SNF casks certified under the current regulatory framework are very low.***

ACRS RECOMMENDATION AND CONCLUSIONS (cont.)

2. ***NUREG-2125 provides a more complete and realistic assessment than earlier SNF transportation risk studies.***
3. ***NUREG-2125 should be published after the responses to our comments are incorporated***

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DRAFT NUREG-1855, REVISION 1, "GUIDANCE ON THE TREATMENT OF UNCERTAINTIES ASSOCIATED WITH PRAs IN RISK-INFORMED DECISIONMAKING"

John W. Stetkar

NUREG-1855

- ***Originally issued for use March 2009***
- ***Regulatory Guides refer to NUREG-1855 methods***
  - ***RG 1.174 – "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"***
  - ***RG 1.200 – "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"***

ACRS Reviews of Revision 1

- ***Subcommittee Meetings on June 19, 2012, and October 19, 2012***
- ***Committee completed review during December 2012 meeting***
- ***Letter Report issued January 2, 2013***

Revision 1 Highlights

- ***Refinements in response to comments received from public workshops, performance of a test case, and user experience***
- ***Reorganized to more closely follow the sequence of activities that are performed to prepare a risk-informed licensing application***

Revision 1 Highlights (cont.)

- ***Specifies levels of analyses needed to meet technical capability requirements in the ASME/ANS PRA Standard***
- ***Guidance and expectations for NRC staff reviews of the treatment of completeness, parameter, and model uncertainties in risk-informed applications***

ACRS Recommendation

- ***NUREG-1855, Revision 1, provides valuable guidance for the treatment of uncertainties in risk-informed decision making. It should be issued for public comments.***

Margins to Acceptance Criteria

- ***NUREG-1855 emphasizes need for enhanced attention to evaluation and review of uncertainties when the point-estimate results are close to challenging or exceeding regulatory acceptance guidelines***
- ***Indicates that comprehensive assessment of uncertainties is less important when results are not close to the acceptance criteria***

Margins to Acceptance Criteria

- ***Assessment of uncertainties provides information about the degree of confidence in the available margins***
- ***Can affect decisions to implement one option vs. another***
- ***Evaluation of uncertainties can also identify sources of optimism***

ACRS Recommendation

- ***Staff should consider revising the guidance to note that assessment and review of uncertainties is important for all risk-informed applications, even when the point-estimate results are well below the nominal acceptance criteria***

Implementation Examples

- ***February 2009 ACRS letter critical of implementation examples in draft NUREG-1855 Appendix A***
- ***NUREG-1855 was issued without Appendix A***
- ***Revision 1 refers to two EPRI reports that contain the examples in practical guidance for a risk-informed licensing submittal***

Implementation Example Issues

- ***Emphasize sensitivity analyses, in lieu of characterization and quantification of uncertainty***
- ***Screening and sensitivity examples presuppose conservatism in point-estimate values***
- ***Sensitivity analyses are not organized to inform a complete evaluation of uncertainties***

ACRS Conclusion

- ***The guidance in NUREG-1855, Revision 1, provides an appropriate framework for the identification and quantification of uncertainties***
- ***Examples in the referenced EPRI reports do not clearly demonstrate appropriate applications of the guidance***

ACRS Recommendation

- ***Staff should initiate efforts to ensure that the principles of uncertainty analysis in NUREG-1855 are applied more consistently throughout the NRC***

Abbreviations

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
AEs	anticipated events
ANS	American Nuclear Society
APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
AP1000	Advanced Passive 1000
BDBEs	beyond design basis events
BWR	boiling water reactor
CFR	Code of Federal Regulations
COLA	combined license application
DBAs	design basis accidents
DBEs	design basis events
DI&C	digital instrumentation and control
DOE	U.S. Department of Energy
DSRS	design specific review standard
EA	enforcement action
ECCS	emergency core cooling system
EPR	Evolutionary Power Reactor
ESBWR	Economic Simplified Boiling Water Reactor
EPRI	Electric Power Research Institute
I&C	instrumentation & control
INL	Idaho National Laboratory
iPWR	integrated pressurized water reactor
LBEs	licensing basis events
NGNP	next generation nuclear plant
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RAIs	requests for additional information
RG	Regulatory Guide
RMTF	risk management task force
SBO	station blackout
SER	safety evaluation report
SMR	small modular reactor
SNF	spent nuclear fuel
SOARCA	State-of-the-Art Reactor Consequence Analyses
SFTRA	spent fuel transportation risk assessment
SSCs	structures, systems and components
V&V	verification and validation