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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 23, 2006

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT - 535th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 7-8, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 535th meeting, September 7-8, 2006, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memoranda:

REPORT:

Report to Dale E. Klein, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the Monticello Nuclear Generating Plant, dated September 19, 2006

LETTERS:

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

- Proposed Direct Final Rule to Amend 10 CFR 50.68, "Criticality Accident Requirements," dated September 21, 2006

- Lessons Learned from the Review of Early Site Permit Applications, dated September 22, 2006

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," dated September 13, 2006

R/1

- Proposed Revision to Standard Review Plan, NUREG-0800, Section 6.1.1, "Engineering Safety Features Materials," dated September 13, 2006

- Proposed Revision to Regulatory Guide 1.23 (DG-1164), "Meteorological Monitoring Programs for Nuclear Power Plants," dated September 13, 2006

- Draft Final Revision to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," dated September 13, 2006

- Questions Raised by Members of the Public During the ACRS Subcommittee Meeting on Palisades Nuclear Plant License Renewal Application, dated September 13, 2006

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant

The Committee met with representatives of the NRC staff and the Nuclear Management Company, LLC (NMC) to discuss the license renewal application for the Monticello Nuclear Generating Plant (MNGP) and the final Safety Evaluation Report (SER) prepared by the NRC staff. The operating license for MNGP expires on September 8, 2010. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date. The applicant discussed operating experience; major equipment replacements and repairs; major exceptions to the Generic Aging Lessons Learned Report; and the commitment tracking system. The staff discussed the results of its evaluation of the Monticello license renewal application as well as the results of the inspection and audit. In the final SER, the staff concluded that the requirements of 10 CFR 54.29(a) have been met.

Committee Action

The Committee issued a report to the NRC Chairman on this matter, dated September 19, 2006, recommending that the NMC application for renewal of the operating license for MNGP be approved.

2. Lessons Learned from the Review of the Early Site Permit Applications

The Committee met with representatives of the NRC staff and applicants to discuss any lessons that may have been learned from the preparation, evaluation, and review of the North Anna, Grand Gulf, and Clinton ESP applications. The staff and applicants agreed that there should be better communications and guidance related to the information contained in applications. Specific areas that would benefit from clearer guidance include: guidance for the electronic submission of applications, guidance on the treatment of the high frequency component of seismic ground motion, guidance for computing the probable maximum flood at proposed sites, and guidance for assuring the integrity of data posted on the Internet. Some issues that consumed a lot of time during the preparation and review of the first three ESP applications, such as the development of the "plant parameter envelope" and the review of specific major features of an emergency plan, are unlikely to require the same level of attention in the future.

Inside NRC

Volume 28 / Number 17 / August 21, 2006

Eight units feel impact of MSPI with indicator, column changes

Implementation of the new risk-informed performance index in NRC's reactor oversight process (ROP) triggered eight reactors to change from a green to a white indicator, which was slightly fewer than the 10-12 anticipated from completing a pilot but a significant jump from the three white indicators reported for first-quarter 2006 under the previous set of performance indicators (Pis).

The color changes to the indicator also caused the eight reactors to move into a performance column in the ROP action matrix that would increase the level of NRC oversight (see story, page 9).

A handful of units teetered on the green-white threshold, while the three units that previously had been white PI turned green. Under NRC's oversight process, a green indicator means performance met the objectives of the mitigating systems cornerstone. White indicates that performance was outside an expected range of utility performance, while yellow and red indicators correspond to minimal and significant reductions in safety margin.

The first quarterly results of the mitigating system performance index (MSPI) were sent to the NRC in mid-July and released on the agency's web site on August 4 (http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/pi_summary.html).

The MSPI, which consists of five monitored systems and support systems, replaced the safety system unavailability PIs, which tracked four systems. While the industry considers the MSPI an improvement over the previous indicator because it is plant-specific and based on risk-significant functions, it is more complex.

NRC posted this explanation of the indicator on its web site: "In simple terms, the MSPI reflects the composite averaged performance of important components and trains within a monitored system over a 12 quarter (three-year) period. Licensees will report two values for each of the five monitored systems, a UAI, or unavailability index number, and a URI, or unreliability index number. NRC will then add the two together for a total MSPI index value for the system."

Lessons learned

Both industry and NRC staffers acknowledged the complexity of the indicator at an August 16 meeting during a discussion of the four years it took to revise the PI. John Butler of the Nuclear Energy Institute said there could be value in reviewing the lengthy change process but that it was unlikely there would be any kind of similar type of change in near future.

John Thompson, senior reactor operations engineer in NRC's performance assessment branch, said the recommendations from the staff were general and could be applied to future PI changes.

He said the staff had suggested that future PI changes be kept as simple as possible. "We thought we were keeping it simple (for MSPI) but it mushroomed over the years," he said. By minimizing the complexity, "you lose some accuracy, but you gain clarity," Thompson said. "It's a trade-off."

Looking back, the absence of a rule on probabilistic risk assessment (PRA) standards also complicated efforts to change the indicator, Thompson said of the staff's assessment. He said the staff recommended that the NRC staffindustry working group involve the "right people" earlier in the process. For future PI changes, there should be more of an effort by both sides to identify "showstoppers," or critical issues, sooner. Another lesson learned was to keep the commission, and executive director for operations, informed of the progress — or lack of it. "We should have briefed them earlier," Thompson said.

The industry agreed that the MSPI should have been simplified. "We started with a more modest version," NEI's Butler said. But in the end, "we bit off a lot," he said. Butler said the industry now believes that there would have been benefits to have re-piloted the index change. A six-month pilot involving 20 units was conducted in 2002. But many changes were made to the MSPI following the pilot. One industry representative said that the final MSPI, in fact, had "little resemblance to what was piloted." "We probably would have been receptive to that if you weren't so entrenched in changing it," Thompson said.

The industry representative said there wasn't "adequate input" from across the industry. Butler agreed that PRA experts and owners groups should have been involved earlier. Another generic lesson was to keep a more accurate master list of issues — small and large — that needed to be addressed. Some issues got dropped, others languished, and others still didn't make it to closure, industry and NRC staff said.

Butler said the industry now recognizes that there should have been greater industry involvement during the pilot. He said there should have been participants representing a range of plant designs and vintages.

Like NRC staff, the industry also recommended improving communications on its end. Butler said licensees should have been kept better informed of the changes and discussion. One industry representative said some plant employees were "trying to kill MSPI" even after their chief nuclear officer had approved it.

MSPI changes

The industry and NRC staff also are scrutinizing how the MSPI has been put into practice. At the August 16 meeting, the group discussed a few generic and plant-specific issues, including a situation at Brunswick. The industry said that a motor/pump coupling broke on a service water pump at Brunswick, causing a failure and "unplanned unavailability." Plant employees determined that the coupling failed because of corrosion and wanted to check the condition of the remaining nine pumps by removing them from service for about 3-4 days each.

Under the MSPI, Brunswick would have to report "unavailability" even though the maintenance was "planned" unavailability, the industry said. That is because much of the work would fall outside of the plant's baseline for planned maintenance. The industry asked at the meeting for Brunswick to be allowed to adjust its baseline in mid-quarter rather than allow it to go into effect in the following quarter.

"All of the unavailability will take place this quarter so making the change effective next quarter will be after the fact and, in all likelihood, after the PI changes from Green to White because of the significant amount of unavailability incurred," the industry said in a request for a deviation from the MSPI guidance.

Mark Tonacci, a reactor operations engineer in NRC's performance assessment branch, questioned why the staff should allow the change. "I think this PI is measuring exactly what we expected," he said.

But an industry representative argued that a licensee would be "penalized" for doing what it needs to do to ensure reliable performance at the plant. Another industry official said the company would be punished for "doing the right thing." A third meeting participant said Brunswick would feel the hit financially because of the increase in inspections. In addition, he said, the financial community would be monitoring the situation since it could result in one or two white indicators.

But Thompson said the indicator was now risk-informed and "allows a balance between reliability and unavailability. To flip-flop is not what we want to do." Later, Thompson said he was "not sympathetic to the argument" that prohibiting the baseline change during the current quarter would "drive industry to do the wrong thing." He noted that the industry had previously said that a PI color change would not come into consideration when work needed to be done.
With the industry and staff at an impasse, the two sides decided to revisit the issue at next month's meeting, which will be held for the first time outside of NRC headquarters. The ROP meeting is scheduled for September 14 in Fort Worth, Texas, the day after the Region IV utility group meeting.—
Jenny Weil, Washington

NRC staff commends, comments on ANS draft fire PRA standard

The American Nuclear Society's draft standard for fire probabilistic risk assessment methodologies is "for the most part well-written and addresses important issues," NRC staff said in its August 10 comments.

Development and issuance of the ANS standard, BSR/ANS-58.23, is a critical element in NRC's effort to provide a performance-based, voluntary alternative to current fire protection regulations in the form of the National Fire Protection Association's standard, NFPA 805, which requires that plants have a fire PRA, something few of them currently have (INRC, 6 Feb., 1).

About 37 units so far have notified the agency of their intent to transition to NFPA 805 over the next few years (INRC, 23 Jan., 1).

NRC staff's comments "relate to scope, clarification, the relationship with the ASME (American Society of Mechanical Engineers) internal events PRA standard, interpretation, and strengthening of requirements in this standard," Farouk Eltawila of the Office of Nuclear Regulatory Research said in an August 10 letter to ANS standards coordinator Patricia Schroeder.

The letter and NRC staff comments are available on NRC's Adams document system under accession number ML062120524.

Eltawila said "the major concern expressed" in the comments "is in regard to the relationship of the fire PRA standard with the ASME PRA standard," which "is not clear in the fire PRA standard."

He said that NRC staff believes "this issue is more efficiently and effectively resolved via an integrated standard.

Therefore, we recommend that the fire PRA standard should be issued as part of the ASME/ANS integrated PRA standard, in lieu of a separate, stand-alone standard, if this can be done without significantly affecting the schedule."

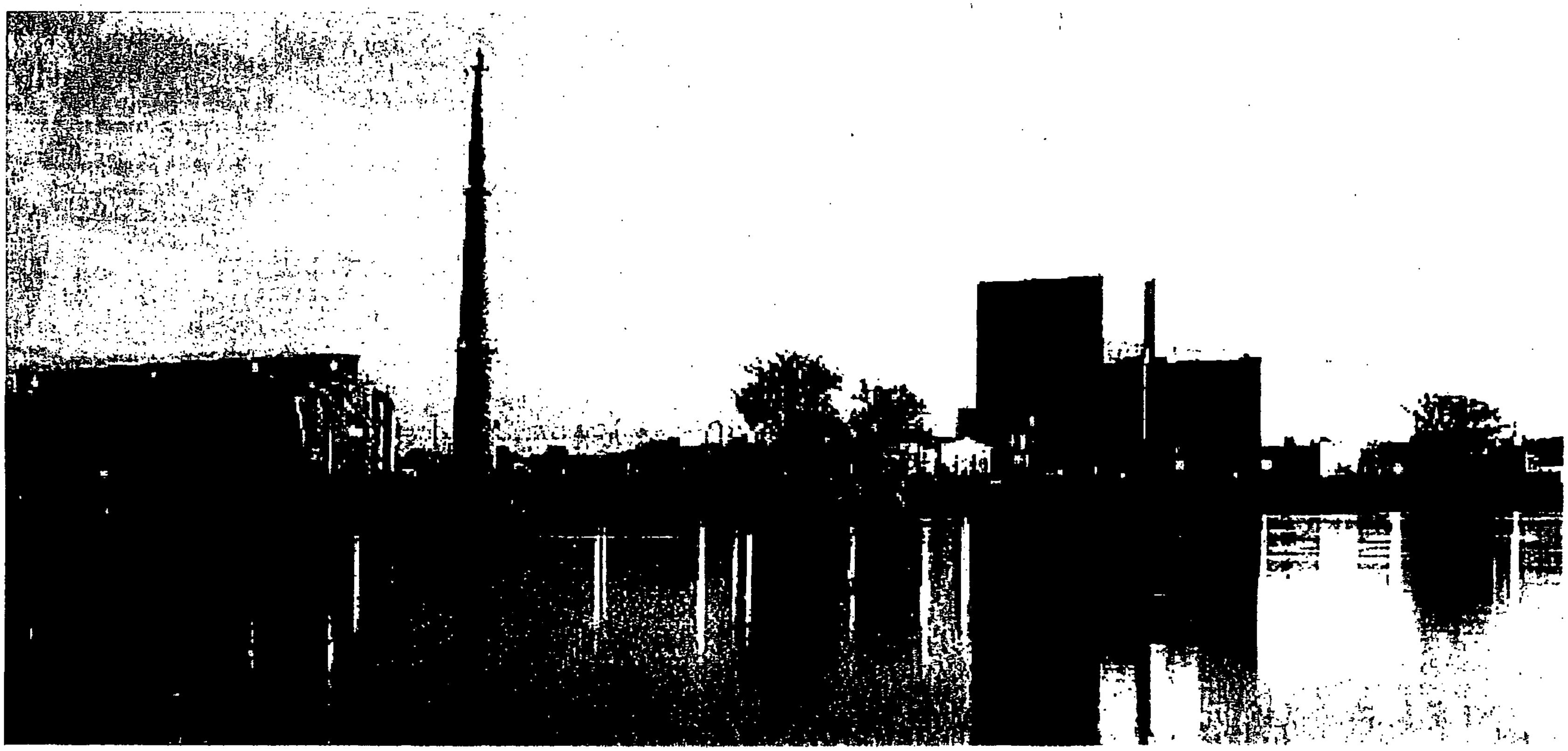
Schroeder said in an interview last month that the comment period has been extended to August 19. She expects to receive in the range of 300 sets of comments on the standard, which will take "several months to resolve." Voting on the final standard will then take place, and the standard will have to be rewritten if too many negative ballots are received, Schroeder said. Thus, it could take anywhere from two months to a year to finalize the standard after voting closes.

NRC posts information on the fire protection program on its web site (http://www.nrc.gov/reactors/operating/opsexperience/fire-protection. html).—***Steven Dolley, Washington***

Monticello Nuclear Generating Plant

ACRS License Renewal Presentation

September 7, 2006



Xcel Energy

NMC
Committed to Nuclear Excellence

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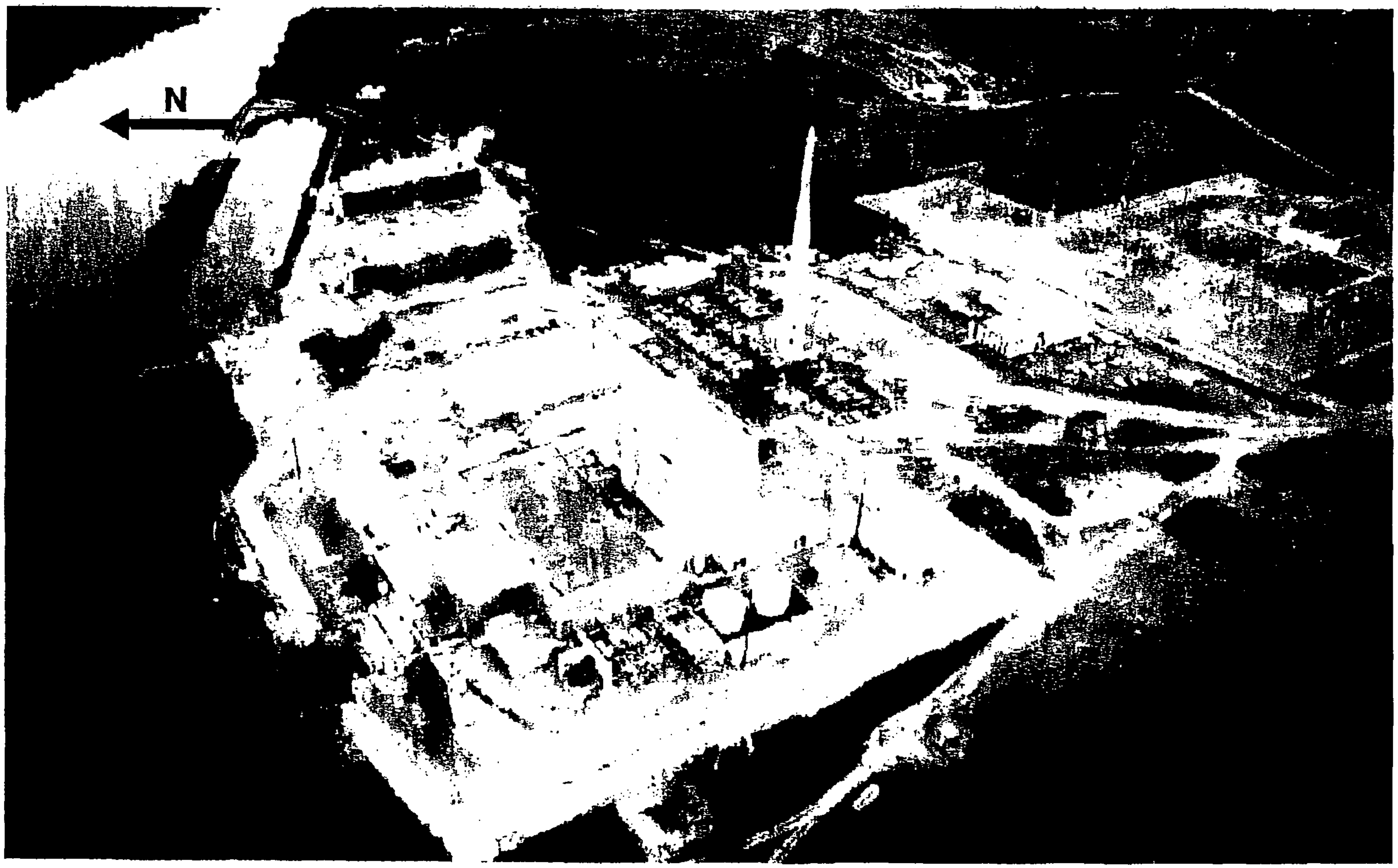
- [illegible] of Engineering
- Pat Burke - Manager of Projects
- Joe Pairitz - LR Project Manager/Mechanical Lead
- Ray Dennis - LR Civil/Structural Lead
- Ron Siepel - LR Electrical Lead
- Jim Rootes - LR Programs Lead
- Mike Aleksey – TLAA Coordinator
- Dave Potter - Engr. Supervisor of Inspections/Materials

Agenda

- ➢ Description of the Monticello Nuclear Generating Plant (MNGP)
- ➢ Major Plant Enhancements
- ➢ Project/Application Background
- ➢ ACRS Subcommittee Follow-up Items
  - Shroud Neutron Fluence
  - Drywell Shell Integrity
- ➢ Commitment Tracking/Implementation Status

Monticello Nuclear Generating Plant



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Major Plant Enhancements

- 1984: Replaced all recirculation piping with low carbon stainless steel resistant to Intergranular Stress Corrosion Cracking (IGSCC)
  - Risers, supply headers, suction piping, and safe-ends replaced
  - Number of welds significantly reduced
  - Induction heating stress improvement and electro-polishing applied to new pipe
- 1986: Core Spray safe-ends and piping replaced with IGSCC-resistant material

Major Plant Enhancements

- 1989: Moderate Hydrogen Water Chemistry (HWC-M) initiated
- 1997: Replaced Emergency Core Cooling System (ECCS) suction strainers in suppression pool (Torus)
  - Strainer design and surface area significantly improved
- 1998: Both condensate pumps replaced with more efficient models; one pump motor also replaced
- 2005: #11 Recirc pump motor & rotating assembly replaced
- 2005: 24-Month Fuel Cycle License Amendment approved
- Future Life Cycle Management projects (e.g. replacement of FW heaters, #12 recirc pump motor & rotating assembly, SW pumps, transformers, generator rewind, etc.)

Project/Application Background

- Core Team NMC Employees
  - 4 with previous SRO or SRO certifications at MNGP
  - Experienced, multi-discipline MNGP personnel
- Supplemented by LR experienced on-site contractor support
- Team retained to support audits/inspections and implementation activities
- Contract with GE for RPV & Internals TLAAs & AMRs
- Plant/Site personnel involved with AMR & AMP development

ACRS Subcommittee Follow-Up Items

- Shroud Neutron Fluence
  - Magnitude Increase Calculated for LR
- Drywell Shell Integrity
  - Location of sand pocket drains with respect to previous drywell floor excavation
  - Configuration of sand pocket area

Shroud Neutron Fluence

- Explanation for relative magnitude difference between 54 EFPY and 32 EFPY values
- Maximum 54 EFPY shroud fluence: 3.84×10^{21} n/cm2
  - Calculated using Reg Guide 1.190 methodology
- Previous 32 EFPY shroud fluence: 2.7×10^{20} n/cm2
  - From APED-5460, Design and Performance of General Electric Boiling Water Reactor Jet Pumps
- Primary difference is water gap geometry
  - Approx. 1.8 inch min. (MNGP) vs. 6.7 inches (APED-5460)

Drywell Shell Integrity

➢ <u>MNGP Design Features</u>

- Three separate drain paths prevent water accumulation

- Sealed sheet metal barrier over the sand pocket area

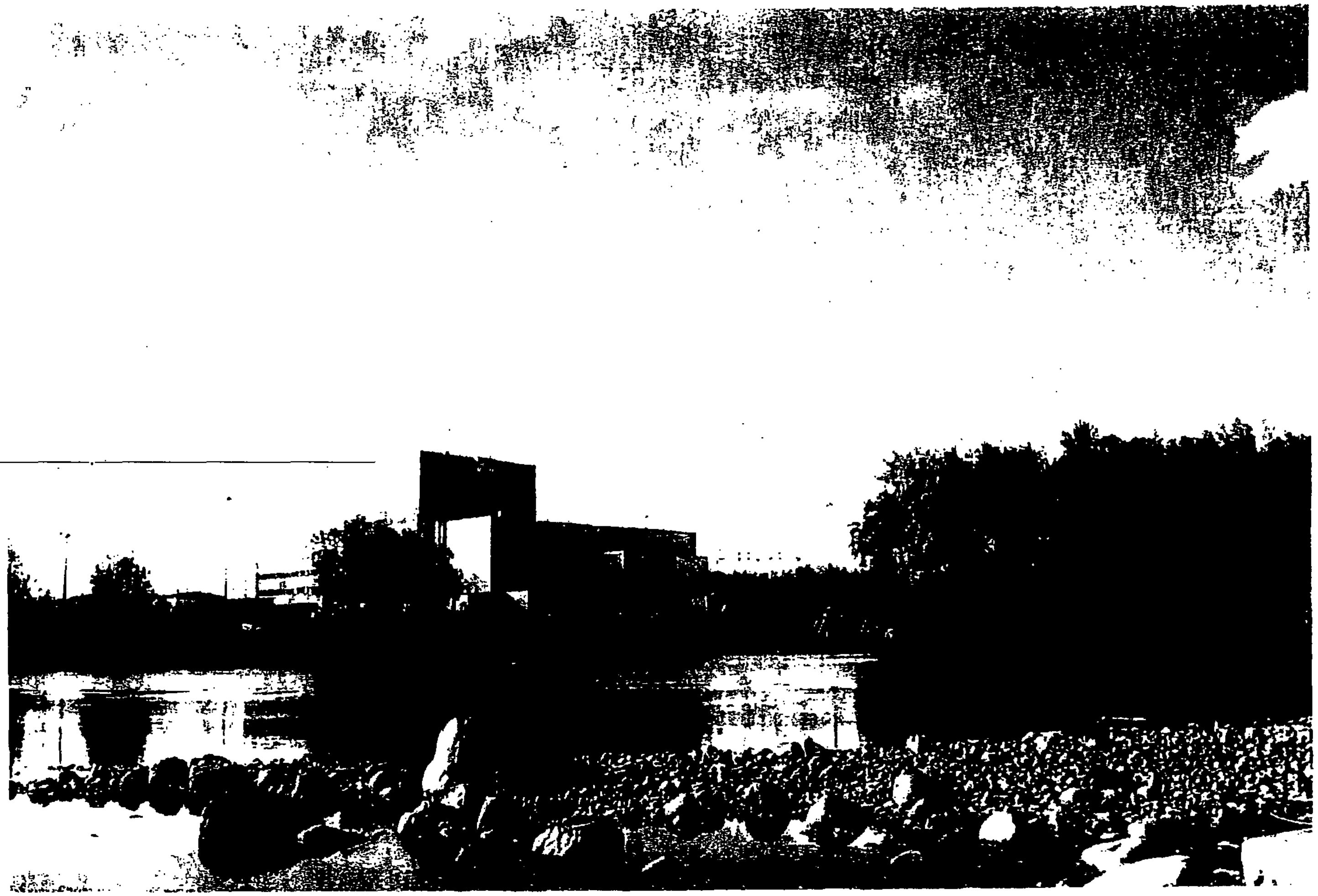
➢ <u>Excavation of Drywell Floor for UT Measurements</u>

- Extensive GL 87-05 UT inspections on drywell shell

Commitment Tracking/Implementation Status

- 60 Commitments made to enhance aging management at MNGP
- Commitments are described in the MNGP License Renewal Updated Safety Analysis Report Supplement
- All commitments are entered in the MNGP Corrective Action Program
  - Assures an owner and a due date
- Implementation Status

Questions?



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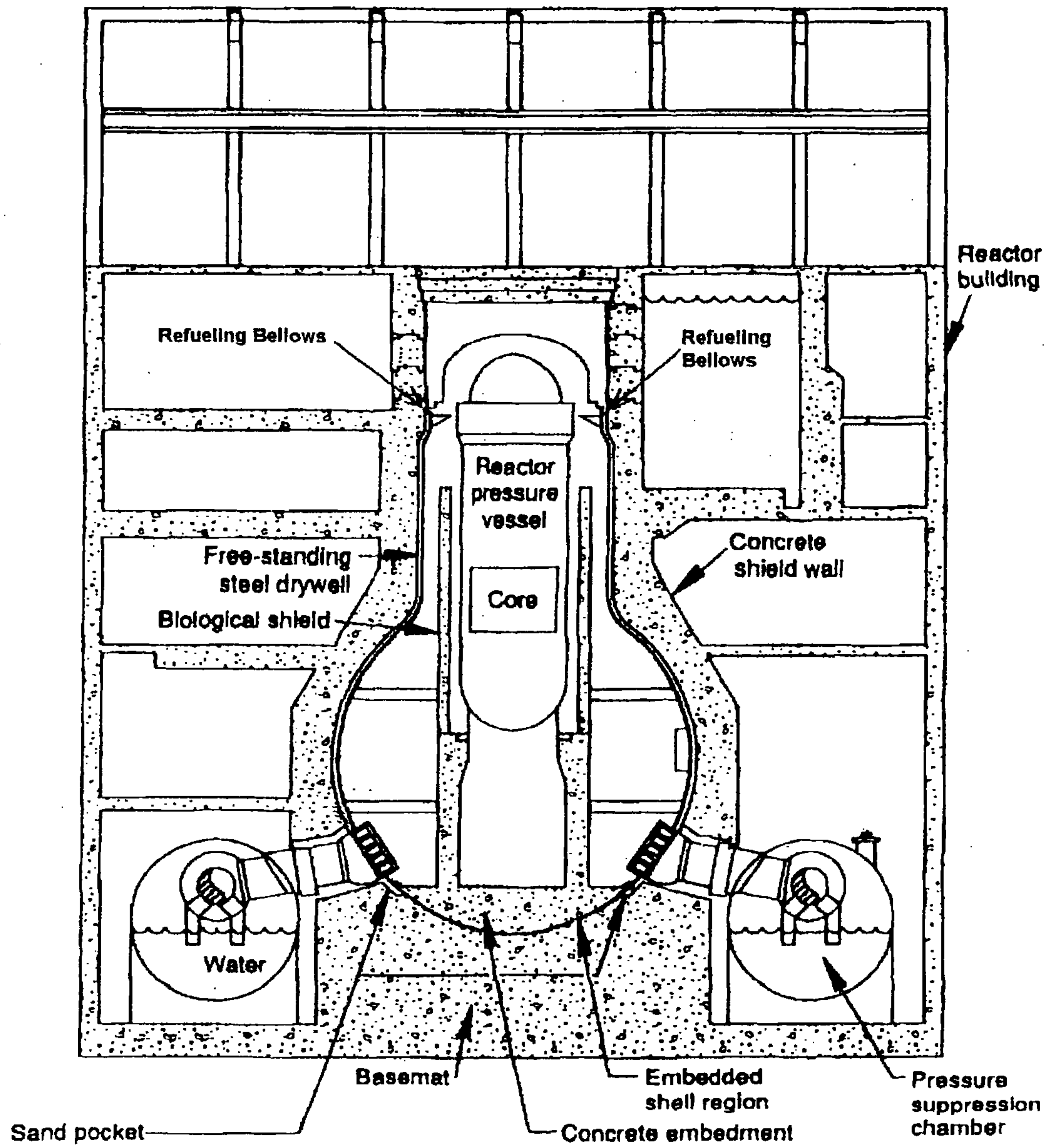
MNGP Mark I Primary Containment



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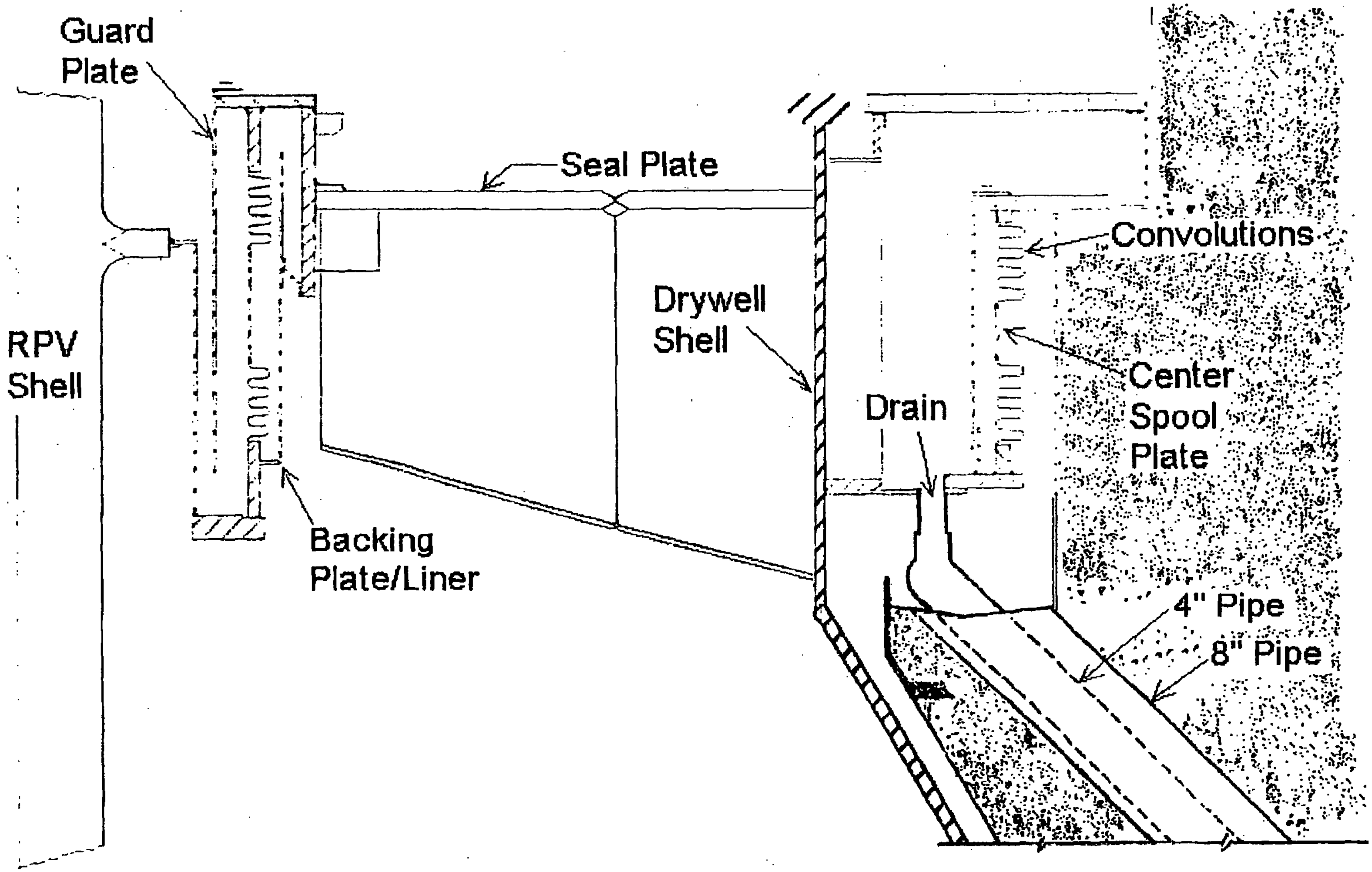
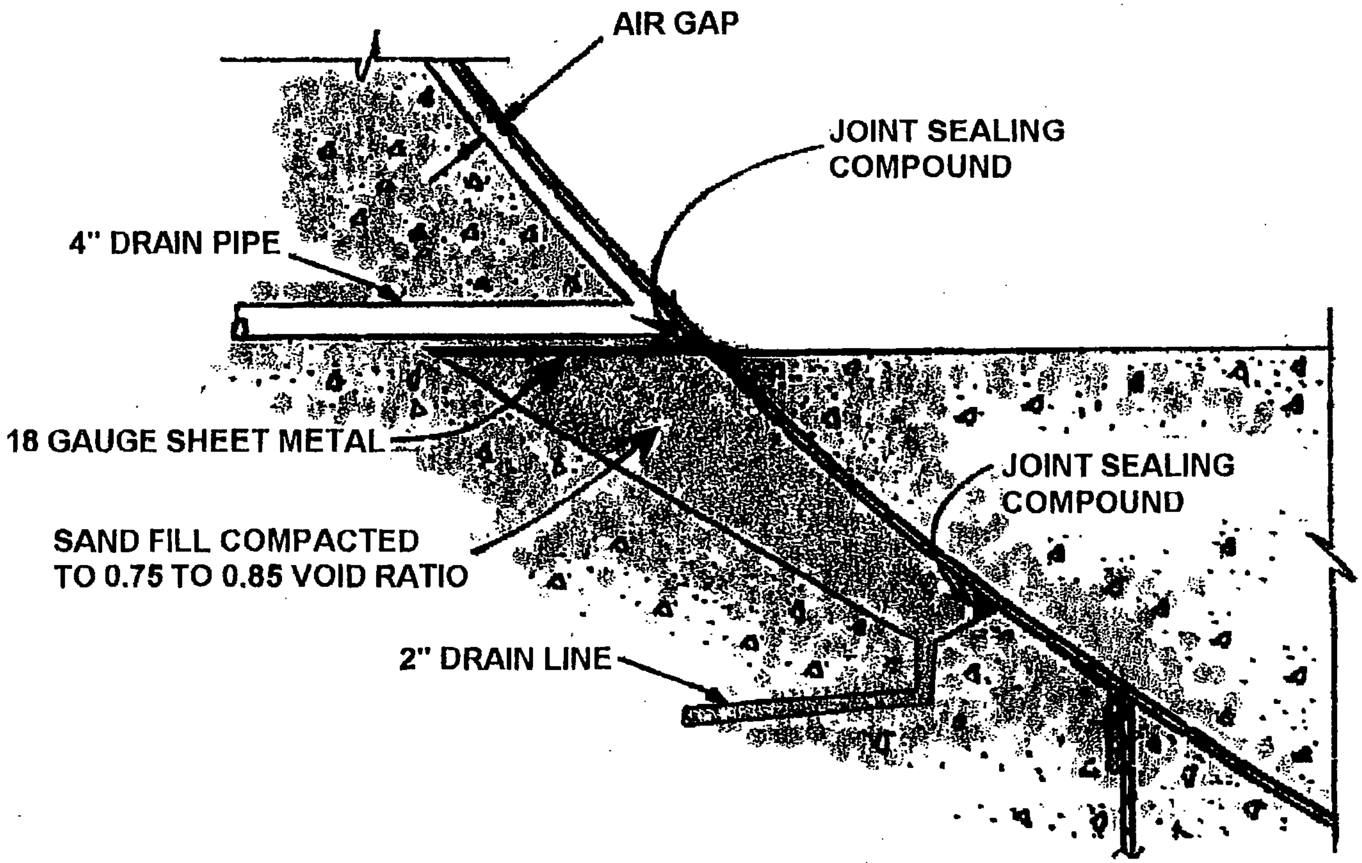
Refueling Bellows



Sand Pocket Area



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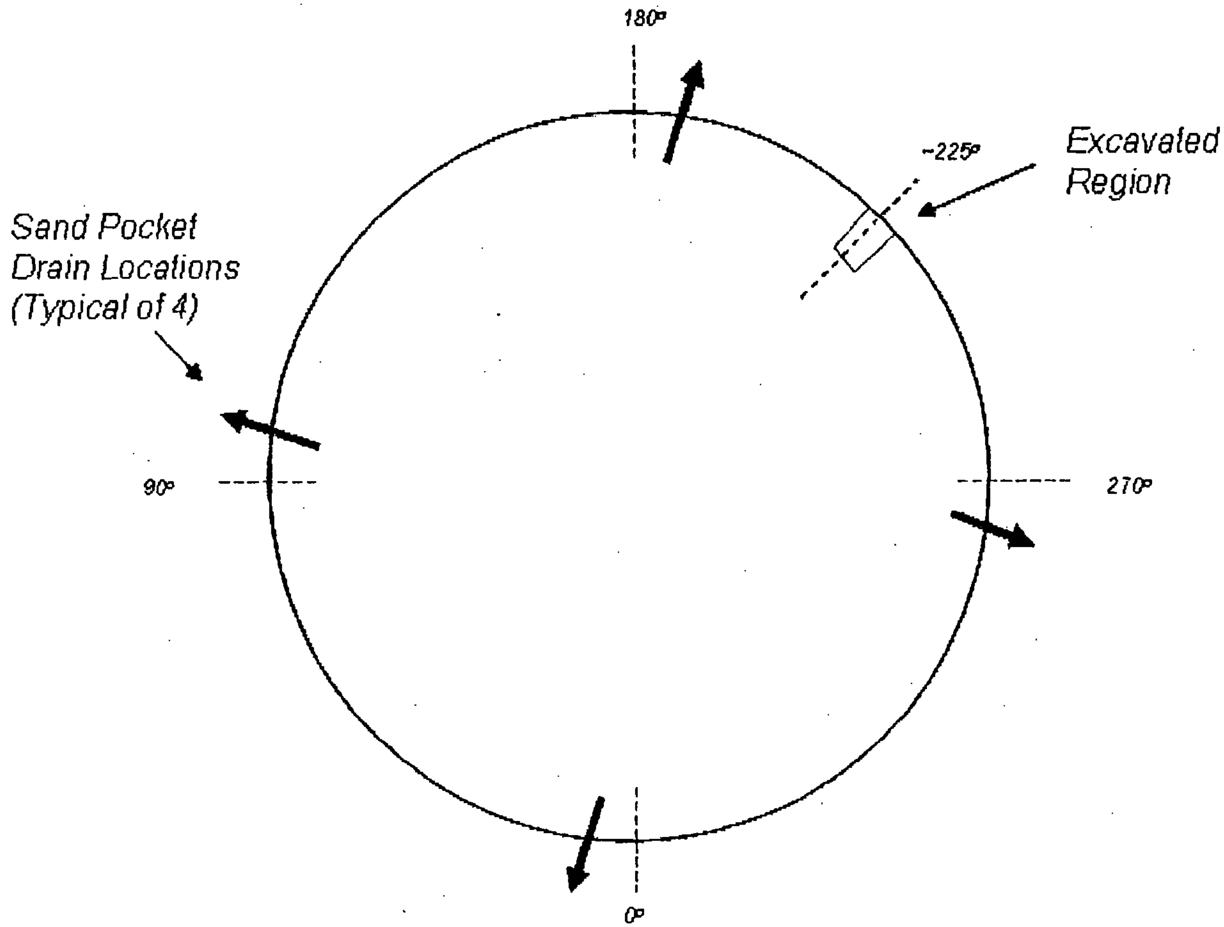

Plan View of Monticello Drywell at Elevation 920'

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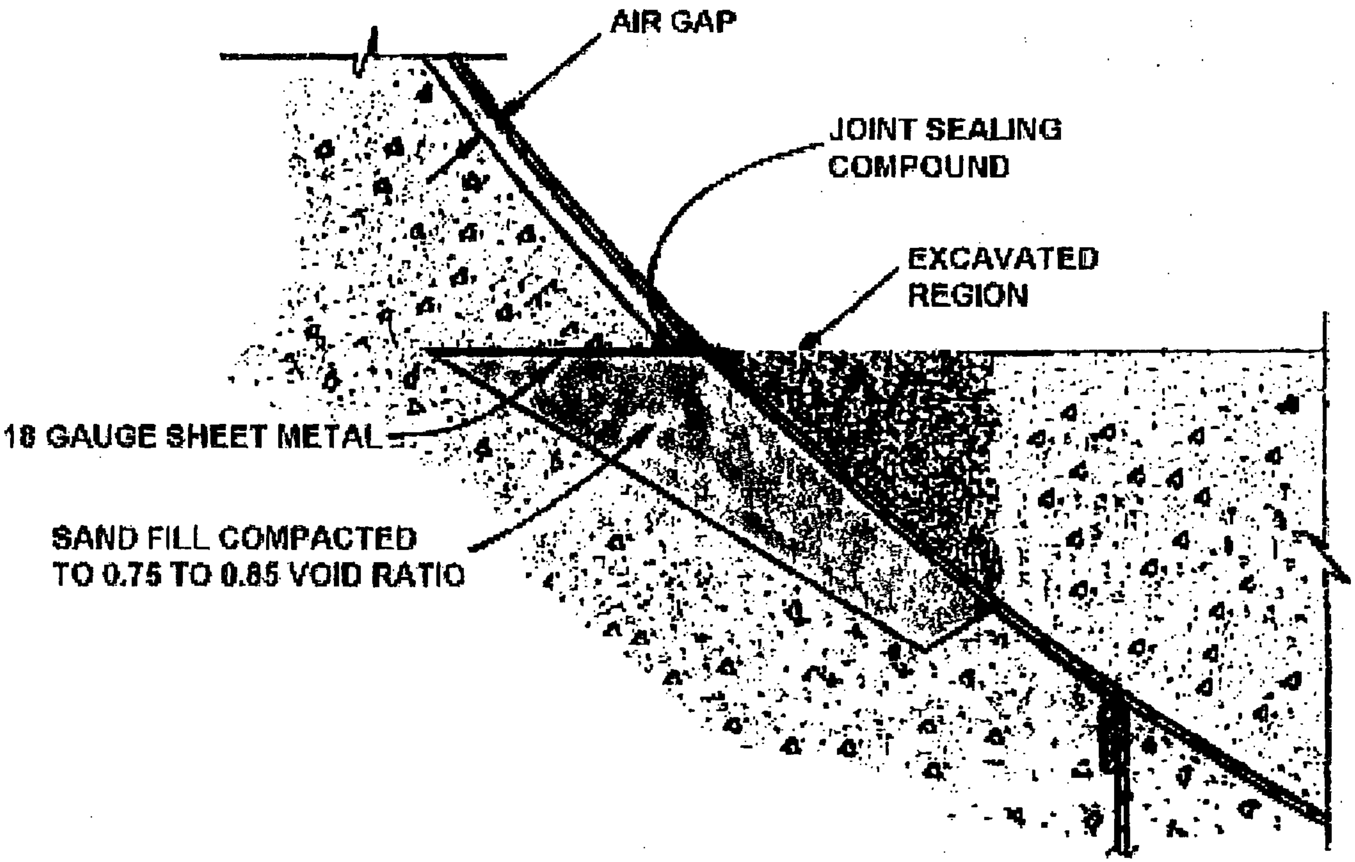
SAND POCKET AREA AT 225° AZIMUTH



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Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report

Staff Presentation to the ACRS

Daniel J. Merzke, Project Manager
Office of Nuclear Reactor Regulation

September 7, 2006

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Introduction

- Overview
- Highlights of the Review
- Time-Limited Aging Analyses (TLAAs)
- Conclusion

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Overview

- LRA submitted by letter, dated March 16, 2005
- GE BWR-3, Mark I steel containment
- 1775 MWth, 600 MWe – includes 6.3% power uprate in 1998
- Operating License expires September 8, 2010
- MNGP located 30 miles NW of Minneapolis, MN

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Overview

- AMP GALL Audit
  - June 13 -17, 2005
- Scoping and Screening Methodology Audit
  - June 20 – 24, 2005
- AMR GALL Audit
  - July 25 – 29, 2005
- Regional Inspections
  - January 23 – 27, 2006
  - February 6 – 10, 2006

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Overview

- Initial SER issued April 26, 2006
  - No Open or Confirmatory items
- 113 RAIs issued
- 95% consistent with GALL, Revision 1
- Final SER issued July 28, 2006
  - 60 commitments
  - 3 license conditions

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Highlights of Review

- Three (3) license conditions
  - USAR to be updated following issuance of the renewed license
  - Commitments completed in accordance with the schedule in Appendix A of the SER
  - Reactor Vessel Surveillance Program
    - All capsules placed in storage must be maintained for future insertion
    - Any changes to storage requirements must be approved by the NRC

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Highlights of Review

- The applicant's scoping methodology meets the requirements of 10 CFR Part 54
- Scoping and screening results, as amended, included all SSCs within the scope of license renewal and subject to AMR
- Items brought into scope and subject to AMR
  - Stored steel plates/hatch covers
  - HVAC piping and steam trap
  - Floor drain piping

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Highlights of Review

- Commitment #57 - NMC will continue to follow applicable BWRVIP inspection guidelines and recommendations throughout the period of extended operation
  - BWRVIP-139 – Steam Dryer inspection
  - BWRVIP-26 – Top Guide inspection
    - Commitment 22 – increased sample size in high fluence region

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Section 3.5: Aging Management – Drywell

- ASME Section XI, Subsection IWE
- Primary Containment In-Service Inspection AMP amended to include requirements of LR-ISG 2006-01
  - UT of sand-pocket region performed in 1986 and 1987, no degradation detected
  - Water leakage monitoring program (each refueling) by procedure
    - refueling seal bellows
    - drywell air gap drains
    - sand pocket drains
  - If leakage detected, augmented inspections will be performed IAW ASME Section XI, Subsection IWE
- Staff concluded the program is acceptable to manage aging.

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Aging Management of In-Scope Inaccessible Concrete

	Acceptance Criteria	MNGP
pH	>5.5	>7.0
Chlorides	<500 ppm	<100 ppm
Sulfates	<1500 ppm	<100 ppm

- Below-grade environment is non-aggressive
- Periodic testing of ground water will be performed for Structures Monitoring Program

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Reactor Vessel Upper Shelf Energy (USE) – Analysis Summary

RV Beltline Component	Acceptance Criterion	MNGP Value at 54 EFPY	Acceptable Y/N
C2220-2 Limiting Plate	>50 ft-lbs	57.5 ft-lbs	Y pursuant to 54.21(c)(1)(ii)
Welds – shielded metal arc	>50 ft-lbs	68 ft-lbs	Y pursuant to 54.21(c)(1)(ii)
N2 Nozzle - forging	>50 ft-lbs	52 ft-lbs	Y pursuant to 54.21(c)(1)(ii)

Conclusions

- On the basis of its evaluation of the license renewal application, the NRC staff has concluded that the requirements of 10 CFR 54.29(a) have been met.

Criticality Accident Requirements
10 CFR 50.68 Rulemaking

ACRS Briefing
September 7, 2006

George Tartal
Project Manager
Regulatory Analysis, Policy and Rulemaking Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Tom Martin
Division Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

Meraj Rahimi
Sr. Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards

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Overview

- Criticality accidents are prevented or controlled
  - 10 CFR 50.68 or 70.24 for fuel in spent fuel pools
  - 10 CFR 71.55 and 71.59 for fuel in transportation packages
  - 10 CFR 72.124 for fuel in dry storage casks
- NRC determined that 10 CFR 50.68-compliant licensees loading casks must meet **both** criticality requirements of 10 CFR 50.68 and 10 CFR 72.124 for fuel within a cask in a spent fuel pool
- NRC did not intend to create overlapping requirements

Overview (cont)

- To comply with 10 CFR 50.68, licensees must perform an additional criticality analysis for fuel within a cask, already licensed under Part 72, in the spent fuel pool and either amend their Technical Specifications or receive an exemption from 10 CFR 50.68
- NRC staff position is that the additional criticality analysis is unnecessary to protect public health and safety

Overview (cont)

- Licensee cost to comply with this requirement is considerable
- Solution is to change 10 CFR 50.68
- The 10 CFR 50.68 rulemaking clarifies the regulatory boundary between 10 CFR Part 50 and 10 CFR Part 71 or 72 for criticality accident considerations

10 CFR 50
Spent Fuel Pools

10 CFR Part 50 Fuel Storage Regulations

- 10 CFR 50.68: Criticality analysis, monitoring, and procedural controls for fuel handling and storage in a spent fuel pool

- GDC 62: Fuel storage criticality prevention based on physical systems or processes, preferably geometrically safe configurations

- GDC 63: Monitoring requirements for fuel storage

10 CFR 50
Spent Fuel Pools

10 CFR 50.68 - Criticality Safety Requirements

- Subcritical in unborated, maximum moderation condition
- Analysis considers:
  - Fuel assembly reactivity based on initial enrichment/design and operational history of the fuel (fuel burnup)
  - Licensees have detailed information on fuel assembly burnup
  - Licensee information supports crediting fuel burnup in analysis
- Soluble boron provides defense-in-depth for prevention of criticality and subsequent fuel damage in PWRs

10 CFR 71
Transportation Packages

10 CFR 71.55 and 71.59 – Criticality Safety Requirements

- Transportation package criticality requirements for single and array of packages under normal and accident conditions
- Design, construct, and package non-site specific fissile material (fuel) to remain subcritical if water leaks in
- Under 10 CFR 71.55(b) analysis must assume moderation by water to the most reactive credible extent (no credit for soluble boron)
  - Consistent with 10 CFR 50.68

10 CFR 72
Dry Storage Casks

10 CFR 72.124 – Criticality Safety Requirements

- Criticality design, fuel handling, control, and monitoring requirements for storage of fuel in dry storage casks
  - Design based on geometry or fixed neutron poisons (or both)
  - Designed to remain subcritical
  - Two unlikely, independent changes before criticality can occur
  - Margins required for uncertainties in data and models
  - Criticality monitors required for handling, using, or storing fuel
- When dry - substantial margins to criticality ($K_{eff} < 0.50$)

Package or Cask in the SFP

- Loading done in water filled SFP for shielding
  - Increased reactivity due to moderation by water
  - Decreased margins to criticality
- Package/casks licensed for broad range of fuel types
  - Generic information is used
  - Burnup credit available to the extent that data is available for cask environment
- Boron dilution event highly unlikely during cask loading
- Reliance on soluble boron to maintain subcriticality for storage-only casks with no poison plates or high-density geometry

RIS-2005-05

- Addresses criticality analyses for SFPs and ISFSIs
- Advises licensees that they must meet criticality requirements of Parts 50 and 72 during storage cask loading in SFPs
- NEI letter dated 7/25/2005
  - Implementation of the RIS would "create an unnecessary burden for both industry and the NRC with no associated safety benefit for public" since Part 72 generic criticality analysis already approved

Rulemaking Purpose and Scope

- Reduce the regulatory burden imposed by compliance with both 10 CFR 50.68 and 10 CFR Part 71 or 72 requirements, as applicable
- The requirements of 10 CFR 50.68 would not apply to fuel that has entered the physical boundary of the cask or package located in the SFP
- The requirements of 10 CFR Part 71 or 72, respectively, would apply to fuel that has entered the physical boundary of a package or cask in the SFP

Technical Evaluation

- Soluble boron used for criticality control
- Potential for boron dilution to cause fuel damage evaluated
  - Slow boron dilution due to injection from an unborated water source
  - Rapid SFP draindown and subsequent reflood of SFP with unborated water
- Controls in 10 CFR Part 71 or 72, as applicable, are sufficient to preclude fuel damage and to protect public health and safety

Slow Boron Dilution

- Scenario
  - Inleakage into SFP from SFP cooling system, fire suppression system or intentional injection of unborated water into package or cask
- Detection of Slow Boron Dilution Event
  - Licensed operator performs fuel movement
  - Periodic sampling required
  - Criticality monitors required
- Fuel damage is highly unlikely

Rapid Draindown

- Scenario
  - Catastrophic failure of SFP
  - Reflood SFP and fill package or cask loaded with fuel with unborated water
- Catastrophic failure of SFP highly unlikely
  - Concurrent with package or cask loading
  - Fuel in cask may be covered with borated water
- Criticality monitors required
- Fuel damage is highly unlikely

Summary

- Criticality controls of 10 CFR Parts 71 and 72 provide assurance fuel damage is prevented by an accidental criticality during package or cask loading
- Requiring separate criticality analyses is not justified based on low risk and reasonable assurance of safety
- Staff plans to issue revised 10 CFR 50.68

Criticality Analysis Comparison

Part 50

- Actinides reactivity credit
- Fission product reactivity credit
- Plant-specific analyses
- Full credit for fixed neutron absorbers with surveillance program

➔ Soluble boron provides defense-in-depth to prevent criticality

Part 71/72

- Actinides reactivity credit
- No Fission product reactivity credit yet
- Generic analyses
- 75% to 90% credit for fixed neutron absorbers no surveillance program required

➔ Soluble boron relied on as a control to maintain subcriticality during fuel handling (Part 72 only)

Why the Differences?

- Full vs. partial burnup credit
  - Under Part 71/72, K_{eff} for casks with non-site specific spent fuels need to be calculated by quantifying the biases and uncertainties with higher accuracy because casks:
    - In an open environment
    - Susceptible to fresh water in-leakage
    - No soluble boron available for defense in-depth

Why the Differences? (cont.)

- Site-specific vs. generic analyses
  - Under part 50, site-specific fuel depletion history is available for storage racks burnup credit analyses
  - Under Part 71/72, generic fuel depletion analyses are used for generic cask designs
  - Under Part 50, site-specific reactor restarts can be used to confirm or fine-tune the criticality computer code predictions over the years
  - Under Part 71/72, there are no confirmation of predicted k_{eff} for storage or transport casks flooded with fresh water

Burnup Credit Actions

- Steps towards full burnup credit for casks
  - Computer code benchmarking data for casks are needed
  - National and international data
  - Burnup credit ANSI 8.27 standards
  - Two applications currently under review by staff with one near completion

Rulemaking Schedule

- Technical basis prepared in April 2006
- Rulemaking was initiated in May 2006
- Direct final rule package prepared in June 2006
- Concurrence in July 2006
- ACRS review in September 2006
- Publish rule in October 2006
- Public comment period through November 2006
- Publish confirmation of effective date in January 2007

State-of-the-Art Reactor Consequence Analyses (SOAR CA)

Presentation to the Advisory Committee on Reactor Safeguards
September 7, 2006

Office of Nuclear Reactor Regulation

Office of Nuclear Regulatory Research

Office of Nuclear Security and Incident Response

5

OUTLINE

- Objectives
- Approach
- Potential Uses
- Improvements/Motivation
- Schedule and Resources
- Scenario Selection
- Accident Progression
- Consequence Analysis
- Internal and External Communications
- Progress To Date
- Next Steps

OBJECTIVES

- Realistic evaluation of severe accident progression, radiological releases and offsite consequences

- Focus on a spectrum of scenarios most likely to contribute to release and subsequent offsite consequences, using a *risk informed* approach

APPROACH

- Use realistic, detailed integral modeling of plant systems, radionuclide transport and deposition, and release pathways (i.e., PRA, MELCOR, MACCS etc.)
- Use updated emergency preparedness modeling assumptions
- Account for plant improvements, including insights from newer, more realistic NRC evaluations
- Account for use of recent mitigation strategies for the delay or prevention of core damage, and further reduction in offsite consequences
- Also develop a faster-than-real-time tool to assist in decision-making in the event of off-normal events

POTENTIAL USES

- Safety-Related Decision Making
- Insights for New Reactor Licensing at New Sites
- Emergency Preparedness and Emergency Response
- Regulatory Analysis Guidelines
- Provide a more accurate assessment of potential offsite consequences for the <u>current</u> state of NPPs
- Communication with the Public, DHS
- Provide insights for future regulatory and research activities

IMPROVEMENTS/MOTIVATION

- Level 1
  - Improved level 1 PRA modeling
  - Improved plant performance
  - Added plant design features (e.g., alternate AC power for SBO)
- Level 2/3
  - Phenomenological experiments → better understanding of source terms
  - MELCOR integrated severe accident analysis code
  - Computing speed
- Net effect
  - More realistic assessment of radiological source term and potential consequences

SCHEDULE & RESOURCES

- Three-year project
  - 1st year: Westinghouse large dry, GE Mark I, and GE Mark III plants
  - 2nd year: GE Mark II, Ice Condenser, Sub-atmospheric plants
  - 3rd year: B&W, CE plants
- Resources
  - NRC staff
  - Contractor - SNL

SCENARIO SELECTION PROCESS

Internal Events Screening (w/ SPAR Models)

- Screen out initiating events with a CDF $\leq 10^{-7}$
- Screen out sequences with a CDF $<10^{-8}$

Sequence Evaluation

- Identify/evaluate the dominant cutsets for the remaining sequences (90% of initiator CDF) including:
  - Equipment unavailabilities
  - Speed of the accident sequence

Scenario Grouping

- Group sequences together that have similar times to core damage and equipment unavailabilities

Scenario Evaluation

- Evaluate scenarios with CDF $\geq 10^{-6}$ to determine if release can be mitigated:
  - Status of containment systems
  - Equipment recovery (e.g. EDGs)
  - Other mitigation measures
- *Also evaluate containment bypass scenarios with CDF $\geq 10^{-7}$.*

Additional Considerations

- Review IPEEEs for dominant external event scenarios (e.g., fire, seismic, flooding) that may exceed 10^{-6} release frequency threshold.
- Review IPEs when enhanced SPAR models are not *available*.

Scenario Selection

- Determine the scenarios that are likely ($\geq 10^{-6}$ release frequency per year) to lead to a *"significant"* radiological release.
  - Also consider high consequence scenarios that are close to the 10^{-6} release frequency threshold.

TECHNICAL ISSUES

- Evaluation of external event scenarios
  - IPEEE data conservatism and limitations
  - Treatment of external event risk numbers versus internal event numbers (data and modeling maturity)
- Mitigation and release frequency calculations
  - Evaluation of mitigation/recovery actions (HRA) for scenario screening and MELCOR input
  - Methodology to calculate scenario release frequencies and address uncertainty

ACCIDENT PROGRESSION

- RCP seal leakage (PWR)
- SRV operation with no DC power (BWR)
- Containment failure mode/characteristics
  - Size
  - Location

CONSEQUENCE ANALYSIS

- Source terms representative of plant group (reactor/containment type)
- Site-specific factors
  - Emergency response
  - Population distribution (2000 census data)
  - Weather data (site weather monitoring program – Reg. Guide 1.23)
    - Availability of precipitation data?
  - Shielding factors

INTERNAL AND EXTERNAL COMMUNICATIONS

- Steering Committee Meetings
- ACRS Meetings
- Deputy EDO for Materials, Research, State and Compliance Programs Briefings
- Commission Staff briefings
- Commission Updates
- Public
  - Category 2 meeting – September 8, 2006
  - Workshops – dates TBD

PROGRESS TO DATE

- Pilot sites (six) selected
- Preliminary scenario selection for GE4 BWR Mark I and Westinghouse 4 Loop/Large Dry
- MELCOR and MACCS enhancement expert panels meeting concluded (August 21-24, 2006)

NEXT STEPS

- **Prepare input to begin MELCOR runs on first six sites**
  - Investigate external event impacts on scenario selection
  - Investigate post accident operator actions to determine impact on scenario selection
  - Investigate potential credit for available equipment for post accident mitigation
  - Revise MELCOR analysis as necessary
    - Example, External events
- **Continue SPAR model runs to identify accident scenarios for remaining sites**
- **Begin MACCS runs on first six sites**

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
535th	12	12.1

Title: PLANNING & PROCEDURES/ FUTURE ACTIVITIES

Authors
John Larkins

Document Attached

P&P Subcommittee Minutes, September 6, 2006

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Instructions to Preparer	From Staff Person:
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	Sam Duraiswamy

September 8, 2006 (8:38am)
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SUMMARY/MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
September 6, 2006

The ACRS Subcommittee on Planning and Procedures held a meeting on September 6, 2006, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 10:45 a.m. and adjourned at 12:10 p.m.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

ACRS STAFF
J. T. Larkins
S. Duraiswamy
H. Nourbakhsh
M. Afshar-Tous
R. Caruso
J. Flack
E. Thornsbury
M. Junge
D. Fischer
M. Snodderly
J. Gallo

NRC Staff

S. Koenick, NRR

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the September ACRS meeting

Member assignments and priorities for ACRS reports and letters for the September ACRS meeting are attached (pp. 8-9). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the September ACRS meeting be as shown in the attachment (pp. 8-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through November 2006 is attached (pp. 10-13). The objectives are to:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 14-15).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Regulatory Guides and Standard Review Plan Updates in Support of New Reactor Licensing

The staff is in the process of developing and/or updating several Regulatory Guides and Standard Review Plan (SRP) Sections in support of new reactor licensing. This is being done to comply with the requirement in 10 CFR part 52 that all Regulatory Guides and SRP Sections that are applicable to new reactors should be completed six months prior to receiving the first COL application. Also, the Commission directed the staff to complete this task by March 2007.

The staff has identified 28 Regulatory Guides to be completed by March 2007 to comply with the 10 CFR part 52 requirement and the Commission direction. The staff has identified several Regulatory Guides that do not need ACRS review because they either deal with process issues or the changes are minor. In addition, the staff requested that the ACRS hold a special meeting in January 2007 to review about 12 Regulatory Guides. The staff has been informed by the ACRS staff and the ACRS Executive Director that the Committee will not hold a special meeting in January 2007.

An alternate proposal by the ACRS staff is included in the attachment (pp. 16-19) and summarized below. Assuming that the staff will provide the documents by the end of September:

- In October, the ACRS will review one Regulatory Guide, and decide whether to review certain Regulatory Guides.
- In November, the ACRS is tentatively scheduled to review 8 Regulatory Guides.
- In December, 6 Regulatory Guides are tentatively scheduled for review.

Assignments have been made for the members and ACRS staff for reviewing and/or making recommendations on whether to review these Guides. The staff is also revising the SRP Sections applicable to the future plant licensing. Upon receiving information on this matter, they will be scheduled for ACRS review.

RES and NRR staff are scheduled to meet with the ACRS at the October 2006 full committee meeting to provide the staff's views on which Regulatory Guides and SRP sections require ACRS review (pp. 20-54). Based on cognizant member's review and recommendations, the Committee will decide on a course of action.

To complete review of these Guides to accommodate the Agency schedule, the Committee may have to hold 4-day meetings in November and December.

Another option for consideration would be the establishment of an Ad Hoc Subcommittee to review these Guides and SRP Sections in October and November and refer to the full Committee only those Guides and SRP Sections that need to be reviewed by the full Committee. Following the Ad Hoc Subcommittee meeting, the Subcommittee Chairman will prepare one proposed letter commenting on all Regulatory Guides and SRP Sections and submit to the full Committee for consideration. Even with this approach, the Committee may need to hold 4-day meetings in November and December.

RECOMMENDATION

The Subcommittee recommends the following:

- The members should provide feedback on the assignments for reviewing the Regulatory Guides. After receiving the documents, Cognizant members should recommend whether to review the Guides assigned to them.
- The Committee should consider holding 4-day meetings in November and December, as needed.
- The Committee should decide whether to establish an Ad Hoc Subcommittee to review those Guides with significant changes.

4) Quadripartite Meeting Status

In response to the invitation letters sent to NRC Commissioners, the EDO, and selected Program Office Directors, Chairman Klein has agreed to be a keynote speaker for the opening session. Dr. Paul Epstein, M.D. from Harvard University will be the keynote speaker for evening session 1. Commissioner Jaczko has agreed to be a keynote speaker for the opening session of day two. Mr. Dennis Spurgeon, Assistant Secretary of Nuclear Energy, DOE, has agreed to be a keynote speaker for evening session 2. The EDO has agreed to attend the meeting.

During the June 2006 ACRS meeting, the members were reminded that final papers and power point presentation slides are due by Friday, July 28, 2006. So far, with the exception of one member, all members have submitted their papers. Some members still need to provide their presentation slides. Member requiring staff support for papers and/or slides should let Mike Snodderly know as soon as possible. We anticipate receiving the papers from Japan shortly.

Arrangements have been made to visit TMI-1 Nuclear Plant on October 17, 2006. Several meeting attendees from Japan, Germany, and France as well as ACRS members Armijo, Maynard, Sieber, and Wallis will participate in this plant visit.

RECOMMENDATION

The Subcommittee recommends that the members make sure that the papers and presentation slides are completed as soon as possible. The ACRS staff should keep the Committee informed of the arrangements for visiting TMI-1.

5) ACRS Meeting with the NRC Commissioners

The ACRS meeting with the NRC Commissioners is scheduled for Friday, October 20, 2006, between 2:30 and 4:30 p.m.. The following topics have been approved by the Commission:

I. Overview (GBW)
- Accomplishments
- License Renewal
- Power Uprate
- Risk-Informing 10 CFR 50.46
- Ongoing/Future Activities

II. PWR Sump Performance (GBW)

III. Safety Research Program Report (MVB)

IV. Lessons Learned from the Review of Early Site Permit Applications (WJS)

V. Future Plant Design Activities and coordination with the NRC staff on the Master Integrated schedule. [Including 10 CFR Part 52] (TSK)

During September ACRS meeting, the Committee needs to discuss and provide comments on the presentation slides. Following approval by the Committee at the October meeting, the final slides will be sent to the Commission.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the proposed presentation slides.

6) Proposed Revision to the ACRS Subcommittee Structure

A proposed revision to the ACRS Subcommittee structure is attached (pp. 55-72). This revision involves combining certain existing Subcommittees, creation of new Subcommittees to deal with COL applications, and member assignments. It was sent to all members and the ACRS staff engineers in August 2006 for comment. Comments received have been incorporated, as appropriate. Assignment of staff engineers for certain Subcommittees will be made in the near future.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the proposed revision to the ACRS Subcommittee structure.

7) Annual Retreat, visit to a Nuclear Plant, and Meeting with the Regional Administrator

Each year, the members visit a nuclear plant and meet with the Regional Administrator to discuss items of mutual interest. In 2006, the members visited the Limerick Nuclear Plant and met with the Region I Administrator.

In 2007, the Committee will visit a plant in Region IV and meet with Region IV Administrator. During the discussion of Risk Management Technical Specification Initiative 4b, "Use of Configuration Management for Determining Technical Specification Completion Times Related to the use of PRA and Risk-Monitoring Tools," at the April 28, 2006 Reliability and PRA Subcommittee meeting, Dr. Apostolakis suggested that in 2007 the members visit a plant with Risk Monitor. The plants in Region IV that use Risk Monitors are San Onofre, South Texas, and Fort Calhoun. It was also suggested that the 2007 plant visit and meeting with the Regional Administrator be held in January 2007.

During the visit to the Limerick plant, there were some discussions about combining the 2007 ACRS retreat with the plant visit. Please be informed that in January 2007, we anticipate having several Subcommittee meetings.

RECOMMENDATION

The Subcommittee recommends that the Committee not hold a retreat in 2007 to allow time for holding Subcommittee meetings, as needed, and that the members visit San Onofre in June/July 2007 and meet with the Region IV Administrator to discuss items of mutual interest.

8) Meeting with the Nuclear Installations Inspectorate (NII) United Kingdom

During a conversation with Mr. Paul Harvey, Principal Inspector, NII, at the July 26, 2006 meeting with the Region I Administrator, Dr. Wallis expressed some interest in a meeting between NII and members of the ACRS to discuss items of mutual interest. Subsequently, Mr. Harvey sent an e-mail to the NRC Office of International Programs (OIP), stating that NII would like to find out whether Dr. Wallis wants to pursue his interest in meeting with NII and if so when. Dr. Larkins has discussed this matter with the OIP Desk Officer for the U.K. and noted that the Committee has had bilateral exchanges with the U.K. in the past and would get back to OIP shortly.

It should be noted that the Committee met with Mr. Lawrence Williams, Her Majesty's Chief Inspector, NII during the December 5-7, 2002 ACRS meeting to discuss several items of mutual interest, including pre-decisional plans to expand the nuclear program in U.K.

RECOMMENDATION

The Subcommittee recommends that the Committee invite NII representatives to meet with the ACRS to discuss items of mutual interest. If NII agrees to meet with the ACRS, the Committee should propose a list of topics for this meeting.

9) Request by Mr. Herschel Specter to brief ACRS on Indian Point Emergency Planning

In an e-mail to Dr. Kress, dated August 20, 2006 (pp. 73-74), Mr. Herschel Specter states the following:

- There has been a large effort to modernize the emergency plan at the Indian Point Nuclear Plant.
- For about two years, as a consultant to Entergy, the Indian Point plant owner, Mr. Specter has led the technical effort to modernize the emergency plan at Indian Point. This phase of the effort is nearing completion and Entergy and its supporting team would like to present their analyses to the ACRS sometime after Thanksgiving this fall.
- The NRC staff and SNL are also active in modernizing the emergency plan and they may be ready to present their results in a similar timeframe.

ACRS does not normally get involved in reviewing plant-specific emergency plans. We need to discuss with the NRC Chairman whether ACRS should get involved in this matter. In addition, since staff and SNL are involved in modernizing the emergency plan, we should wait until they complete their work. If the Commission, EDO, or the staff requests ACRS involvement in this matter, then we should schedule a briefing and invite Mr. Specter, staff, SNL, EPRI, and NEI to present their views at that time. Even if a briefing had to be scheduled, it will not happen until mid 2007.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director obtain NRC Chairman's views with regard to the ACRS involvement in this matter.

10) ACRS Meeting Dates for CY2007

A calendar which includes proposed ACRS meeting dates for CY2007 is attached (pp. 75-86) and summarized below. The members should provide feedback on these proposed meeting dates.

--	January 2007 (No ACRS Meeting)
539	February 8-10, 2007
540	March 8-10, 2007
541	April 5-7, 2007
542	May 3-5, 2007
543	June 6-8, 2007 (Wed. - Friday)
544	July 11-13, 2007 (Wed. - Friday)
--	August (No ACRS Meeting)
545	September 6-8, 2007
546	October 4-6, 2007
547	October 31 - November 1-2, 2007 (Wed. - Friday)
548	December 6-8, 2007

RECOMMENDATION

The Subcommittee recommends that the Committee approve meeting dates for CY2007.

ANTICIPATED WORKLOAD
September 7-9, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Armijo	—	Caruso	Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements"	A	To support the staff's schedule	
Bonaca	—	Thornsbury	State-of-the-Art Consequence Analysis **[INFORMATION BRIEFING]**	—	—	—
		Thornsbury	Response to EDO on Ongoing Security-Related Activity (Closed)	B	To provide Committee's views	—
		Junge	Final Review of the License Renewal Application and the Final SER for the Monticello Nuclear Plant	A	To support the staff's accelerated schedule	
Kress	—	Caruso/Nourbakhsh	Risk-Informed Criteria for Societal Risk	B	To discuss views expressed by Dr. Kress	—
Powers	—	Fischer	Lessons Learned from the Review of ESP Applications	A	To provide Committee's views	—
		Nourbakhsh	Draft Report on the Quality Assessment of the NRC Research Projects on Containment Capacity Study at SNL and Molten Core Coolant Interaction Study at ANL	—	To be completed in October	—

ANTICIPATED WORKLOAD
September 7-9, 2006 (Cont.)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Wallis	All Members	Larkins, et. al	Preparation for Meeting with the NRC Commissioners (October 20, 2006, 2:30-4:30p.m.)	—	—	—
		Caruso	Subcommittee Report on PWR Sump Performance Issues - Subc. Mtg. 8/23-24/2006	—	—	—

ANTICIPATED WORKLOAD
October 4-6, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Proposed Revision 1 to Reg. Guide 1.200, "An Approach to Determining Technical Adequacy of PRA Results for Risk-Informed Activities"	A+	To support Agency schedule	—
		Nourbakhsh	Verification and Validation of Selected Fire Models	B	To support staff schedule	—
Kress	—	Fischer	Master Integrated Plan for New Reactor Licensing Activities	Report as needed	To support staff schedule	—
Maynard	—	Fischer	Updates to Reg. Guides and SRP Sections in Support of New Reactor Licensing **[INFORMATION BRIEFING]**	—	—	—
		Santos/Junge	**SUBCOMMITTEE REPORT**-Oyster Creek License Renewal Application - Subc. Mtg. 10/3/06	—	—	—
Powers	—	Nourbakhsh	Draft Report on the Quality Assessment of the NRC Research Projects on Containment Capacity Study at SNL and Molten Core Coolant Interaction Study at ANL	A	To support staff schedule	**Draft**
Wallis	All Members	Larkins, et. al	Preparation for Meeting with the NRC Commissioners [October 20, 2006, 2:30-4:30 p.m.]	—	—	—

ANTICIPATED WORKLOAD
November 1-3, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Potential Collaborative Research on Human Reliability Analysis Methods **[TENTATIVE]**	B	To provide Committee's views	—
		Nourbakhsh	Proposed Revisions to Reg. Guide 1.29 (DG-1156), "Seismic Design Classification"	A^{+}[1]	To support Agency schedule	—
Armijo	—	Santos	Proposed Revisions to Reg. Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"	A^{+}	To support Agency schedule	
		Santos	Proposed Revisions to 1.57 (DG-1158), "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components"	A^{+}	To support Agency schedule	
Bonaca	—	Fischer	Draft Final 10 CFR Part 26, "Fitness for Duty Programs"	A	To support the staff's schedule	—

[1]The Committee will prepare one letter commenting on all Regulatory Guides scheduled for this meeting.

ANTICIPATED WORKLOAD
November 1-3, 2006 (Cont.)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Powers	—	Nourbakhsh	Proposed Revisions to Reg. Guide 1.61 (DG-1157), "Damping Values for Seismic Design of Nuclear Power Plants"	A+	To support Agency schedule	—
		Nourbakhsh	Proposed Revisions to Reg. Guide 1.165 (DG-1146), "Seismic Sources and Safe Shutdown Earthquake Ground Motion"	A+	To support Agency schedule	—
Shack	—	Thornsbury	Proposed Revisions to Reg. Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"	A+	To support Agency schedule	—
		Santos	Proposed Revisions to Reg. Guide 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type component Supports"	A+	To support Agency schedule	—
		Thornsbury	Draft Final Rule to Risk-Inform 10 CFR 50.46, "Acceptance Criterial for ECCS for Light-Water Nuclear Power Reactors"	A	To support staff schedule	—
Sieber	—	Junge	Final Review of the License Renewal Application and the Final SER Related to Palisades Nuclear Plant	A	To support the staff's accelerated schedule	—
	—	Junge	Proposed Revisions to Reg. Guide 1.189, "Fire Protection for Operating Nuclear Power Plants"	A+	To support Agency schedule	—

ACRS Items Requiring Committee Action

1 **<u>Proposed Revision to Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident</u>** (Open)

Member: William Shack **Engineer:** Eric Thornsbury

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR/RES

The NRC staff has identified this regulatory guide (RG) and standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007. The proposed revision to the RG will be provided by September 29, 2006.

In a September 6, 2006 Memo, RES recommended that the ACRS waive review of this revised regulatory guide.

The Planning and Procedures Subcommittee recommends that Dr. Shack review the proposed revisions to this RG/SRP section and recommend a course of action on this matter.

2 **Proposed Revision to Regulatory Guide 1.196, Control Room Habitability at Light-Water Nuclear Power Reactors** (Open)

Member: ~~George Apostolakis~~ Powers **Engineer:** Eric Thornsbury

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR/RES B. Beasley, RES/DRAS/PRA

The NRC staff has identified this regulatory guide (RG) and standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007. The proposed revision to RG 1.196 will be provided by September 29, 2006.

In its August 24, 2006 Memo, RES recommended that the ACRS waive review of this revised regulatory guide.

The Planning and Procedures Subcommittee recommends that Dr. Apostolakis review the proposed revisions to this RG/SRP section and recommend a course of action on this matter.

3 **Review NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire"** (Open)

Member: George Apostolakis **Engineer:** Eric Thornsbury

Estimated Time 1.5 hours

Purpose: Determine a Course of Action

Priority: Medium

Requested by: NRR/RES E. Lois

UNCONFIRMED

The content of this report is virtually the same with DG-1136 prepared as part of the rulemaking on post-fire manual actions, which was withdrawn last January. The ACRS has seen and received a briefing on DG-1136. The staff would like to brief the ACRS on draft NUREG-1852.

The Planning and Procedures Subcommittee requests that Dr. Apostolakis review the document and recommend a course of action.

4 **PROPOSED REVISION TO SRP SECTION 6.1.1, "ENGINEERED SAFETY FEATURES MATERIALS"**

Member: Sam Armijo **Engineer:** Hossein Nourbakhsh

Estimated Time15 minutes

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR A. Keim, -1617

By Memorandum dated July 26, 2006, the staff forwarded proposed revisions to SRP Section 6.1.1, "Engineered Safety Features Materials" to the ACRS for consideration (reference ADAMS package ML061530260).

The Planning and Procedures Subcommittee recommends that Dr. Armijo recommend a course of action. Dr. Armijo recommends that the Committee not review this matter.

5 **Browns Ferry Unit 1 105% uprate**

Member: Mario Bonaca **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR Margaret Chernoff

Because of delays in resolving the steam dryer issue related to the 120% EPU request for BF-1, TVA will decide by the end of September whether it will amend its application to request a smaller 105% uprate for BF-1, to allow it to restart in February 2007 at the same power level as Units 2/3. The Staff has indicated that if TVA makes such a request, they would be able to issue a supporting SER some time afterwards, but it is not clear exactly when this would occur.

Normally, the ACRS would not review an uprate of only 5%, but on October 9, 2003, the Committee informed the staff that

"For power uprate requests of less than five percent, if the uprate request does involve important changes to the plant or potentially higher impacts, or if it presents novel issues that the staff believes might benefit from Committee participation, then the staff will inform the Committee and invite it to participate in the review." It is not clear whether the staff will invite the Committee to participate in the review of the 105% uprate for BF-1.

The P&P Subcommittee request that Dr. Bonaca consider this situation and recommend to the full Committee whether or not the Committee should review the smaller power uprate.

PRELIMINARY

August 30, 2006

ACRS REVIEW OF HIGH PRIORITY REGULATORY GUIDES

October 2006

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	GEA	EAT

November 2006

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	WJS	EAT
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	JSA	MB
1.29	Seismic Design Classification (DG-1156)	GEA	HPN
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (DG-1158)	JSA	MB
1.61	Damping Values for Seismic Design of Nuclear Power Plants (DG-1157)	DAP	HPN
~~1.76~~	~~Design Basis Tornado for Nuclear Power Plants~~	~~MC~~	
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	WJS	CXS
1.189	Fire Protection for Operating Nuclear Power Plants	JDS	MAJ
DG-1146	Seismic Sources and Safe Shutdown Earthquake Ground Motion (RG 1.165, Rev 1)	DAP	HPN

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PRELIMINARY

December 2006

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
1.13	Spent Fuel Storage Facility Design Basis	TSK	HPN
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	MVB	DCF
1.93	Availability of Electric Power Sources	JDS	MAJ
~~1.129~~	~~Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants~~	~~OLM~~	
DG-1142	Guidelines for Environmental Qualification of Safety Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	SAK	EAT
DG-1144	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Reactor Water Environment for New Reactors *	SA	CXS
DG-1145	Combined License Applications for Nuclear Power Plants (LWR Edition) *	TSK	DCF

January 2007

RG No.	Regulatory Guide Title	ACRS Mbr	F/C Mtg
~~1.13~~	~~Spent Fuel Storage Facility Design Basis~~	~~DAP~~	
~~1.26~~	~~Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants~~	~~WJS~~	
~~1.68~~	~~Initial Test Programs for Water-Cooled Nuclear Power Plants~~	~~MVB~~	
~~1.128~~	~~Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants~~	~~OLM~~	
~~1.189~~	~~Fire Protection for Operating Nuclear Power Plants~~	~~JDS~~	
~~DG-1145~~	~~Combined License Applications for Nuclear Power Plants (LWR Edition) *~~	~~TSK~~	

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PRELIMINARY

February 2007

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
~~1.23~~	~~Onsite Meteorological Programs~~	~~TSK~~	
~~1.37~~	~~Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants~~	~~OLM~~	
~~1.61~~	~~Damping Values for Seismic Design of Nuclear Power Plants (DG-1157)~~	~~DAP~~	
~~1.71~~	~~Welder Qualification for Areas of Limited Accessibility~~	~~SA~~	
~~1.93~~	~~Availability of Electric Power Sources~~	~~JDS~~	
~~1.124~~	~~Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports~~	~~WJS~~	
~~1.130~~	~~Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports~~	~~WJS~~	
~~DG-1142~~	~~Guidelines for Environmental Qualification of Safety Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants~~	~~SAK~~	
~~DG-1146~~	~~Seismic Sources and Safe Shutdown Earthquake Ground Motion Temporary Title (RG 1.165, Rev 1)~~	~~DAP~~	
~~4.15~~	~~Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment~~	~~OLM~~	

March 2007

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
~~1.200~~	~~An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities~~	~~GEA~~	~~EAT~~

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PRELIMINARY

Determine a Course of Action (November 2006)

RG No.	Regulatory Guide Title	ACRS Mbr	Eng
1.23	Onsite Meteorological Programs	TSK	DCF
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	JSA	MB
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	OLM	MAJ
1.71	Welder Qualification for Areas of Limited Accessibility	SA	MB
1.76	Design Basis Tornado for Nuclear Power Plants	MC	MAJ
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	JDS	MAJ
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	OLM	RC
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	OLM	RC
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	JSA	CXS
1.136	Materials, Construction, and Testing of Concrete Containments (DG-1159)	WJS	CXS
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	GEA	EAT
4.15	Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment	OLM	DCF

* Already on ACRS schedule

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August 24, 2005

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
Advisory Committee on Nuclear Waste

FROM: Farouk Eltawila, Director */RA/*
Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research

SUBJECT: REQUEST FOR WAIVER FROM ACRS REVIEW OF REGULATORY GUIDES NEEDED FOR NEW REACTOR LICENSING ACTIVITIES

The purpose of this memorandum is to provide advanced information to the Advisory Committee on Reactor Safeguards (ACRS) and to request a waiver from the ACRS review of select Regulatory Guides being revised in support of the update of the Standard Review Plan (SRP) (NUREG-0800). These versions are to support new, near-term reactor licensing activities by March 2007.

The staff believes ACRS does not need to review the Regulatory Guides that are listed in Enclosure 1. Enclosure 1 lists the Regulatory Guides with a short discussion supporting our request for a waiver from ACRS review.

The Regulatory Guides listed in Enclosure 2 appear to have potentially one or more significant changes which may be of interest to the ACRS. A short discussion is also provided in Enclosure 2 indicating the anticipated changes to the Regulatory Guide.

We intend to provide the ACRS with a draft version of all of the Regulatory Guides once they have received division-level approval but prior to public comment. The purpose of providing these to the ACRS at that time is for the ACRS to make a determination of interest in the Regulatory Guides individually. The ACRS could conclude for each Regulatory Guide that (1) it is not of interest to the Committee, and thus ACRS review is waived or (2) it is of interest to the Committee. In the latter case, we request that ACRS schedule subcommittees to review each Regulatory Guide. All significant comments need to be provided by the sub-committee to be incorporated into the Regulatory Guides. We suggest the full committee would then review the Regulatory Guides and provide your letters shortly thereafter such that the staff can meet its directed date of having the Regulatory Guides revised and published by March 2007.

CONTACT: Jimi Yerokun, RES
301-415-0585

Enclosures:
1. High Priority Regulatory Guides
   Recommend Waiver of ACRS Review
2. High Priority Regulatory Guides ACRS
   Review May Be Requested

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CONTACT: Jimi Yerokun, RES
301-415-0585

Enclosures:
1. High Priority Regulatory Guides
   Recommend Waiver of ACRS Review
2. High Priority Regulatory Guides ACRS
   Review May Be Requested

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High Priority Regulatory Guides
ACRS Review May Be Requested

RG #	DG#	Regulatory Guide Title	Rationale for why ACRS review is thought to be needed
1.13	DG-1162	Spent Fuel Storage Facility Design Basis	RG being revised to specify the necessary capacity of the spent fuel pool makeup system. RG will state that pool makeup rates should exceed the larger of: the pool leakage rate assuming spent fuel pool liner perforation resulting from a dropped fuel assembly, or the evaporation rate necessary to remove 0.3% of the rated reactor thermal power. Previous guidance did not give consideration for spent fuel pool boiling, only a dropped fuel assembly.
1.37	DG-1165	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	RG generally endorses ANSI standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants." Major change to this RG will be to endorse the 2004 version of this ANSI standard. RG will be consistent with the information added to the SRP due to the withdrawal of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors." The ACRS has not previously reviewed the technical basis for the revisions to RG 1.37.
1.61	DG-1157	Damping Values for Seismic Design of Nuclear Power Plants	RG was issued in 1973 and is being updated based on recommendations developed under an RES program on damping. The updated guide will also consider guidance provided in ASCE Standard 43-05 and ASME B&PV Code Sect. III, Div. 1, App. N.
1.93	DG-1153	Availability of Electric Power Sources	RG does not endorse any industry standards; rather, it compiles staff positions on the subject matter.
1.128	DG-1154	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	RG maintains the current regulatory position by (1) deleting the regulatory positions that are now incorporated into IEEE Std 484-2002, (2) adding regulatory positions to update the reference to IEEE standards applicable to nuclear power generating stations batteries that were contained in IEEE Std 484-1975 and relaxed or deleted in IEEE Std 484-2002, (3) updating the regulatory positions for preventing fires in battery rooms based on the current NRC guidance in Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants," and (4) updating and carrying forward past regulatory positions that took exception to IEEE Std 484.
1.129	DG-1155	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	RG will endorse the current standard IEEE 450-2002 on this subject with several exceptions (i.e., Regulatory Positions). In light of this, endorsement of this standard is expected to be fairly controversial within the industry.
1.200	DG-1161	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	RG being updated to address revisions to the ASME Level 1 PRA Standard and revisions to the NEI peer review and self-assessment process. Currently shown as "UNCONFIRMED" for 11/1-3/06 Full Committee Meeting on 07/06/06 ACRS Future Activities Report.
4.15	DG-4010	Quality Assurance for Radiological Monitoring Programs (Normal Operations) -- Effluent Streams and the Environment	RG was issued in 1979 and is being updated to use MARLAP as the primary reference with subsidiary references that trace back to the original technical basis. This represents a significant shift in the network of supporting documents.

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High Priority Regulatory Guides
ACRS Review May Be Requested

Enclosure 2

RG #	DG#	Regulatory Guide Title	Rationale for why ACRS review is thought to be needed
	DG-1142	Guidelines for Environmental Qualification of Safety Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	DG is a complete revision of the previous DG-1077, which received considerable pushback from the industry via public comments. In light of all the changes to the DG, and the fact that it endorses the current version of IEEE 323-2003, it is being sent back out for public comments.
	DG-1144	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Reactor Water Environment for New Reactors	DG has been issued for public comment. Reviews are planned by an ACRS Subcommittee in November '06, and by the full ACRS Committee in December '06.
	DG-1146	Identification and Characterization of Seismic Sources and Determination of a Performance-Based Safe Shutdown Earthquake Ground Motion	DG contains significant modifications to the current RG 1.165 in that it proposes a performance based method for determining the safe shutdown earth quake ground motion. It is used to support ESP and COL applications.

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High Priority Regulatory Guides
Recommend Waiver of ACRS Review

RG #	DG#	Regulatory Guide Title	Rationale for why ACRS review is not needed
1.20	DG-1163	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	RG will be revised to include steam dryers and related steam system components in BWRs, as part of the overall vibration assessment program for reactor internals, and to provide a summary of what the NRC expects the applicants to address in relation to steam dryer evaluations. In addition, other changes will also be made to address COL applications or applications that do not reference a certified reactor design. The issue of testing of steam dryers and related main steam system components has been comprehesively discussed with the ACRS in the past. As such the proposed revision to this RG need not be presented before the ACRS.
1.23	DG-1164	Onsite Meteorological Programs	Current version of RG is Revision 0 from 1972. A variety of changes are needed simply to have the RG reflect current meteorological monitoring equipment and practices. In addition, many of the regulations currently applicable to meteorological monitoring have been revised or did not exist in 1972 so discussion of the new regulations is being added (e.g. 10CFR 50 Appendix A and Appendix I, 10CFR 51). Likewise, discussion is being added for other RGs written since 1972 that are associated with meteorological monitoring (e.g. RGs 1.111, 1.145 and 1.194). Because these changes are to establish consistency with other regulatory documents and positions already reviewed by the ACRS, there is no need for ACRS review of the changes.
1.26	DG-1152	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	RG addresses quality group standards and is being updated to be consistent with 50.55(a) and ASME standards. The technical basis for this RG was not revised, and the revisions will not impact the technical or policy issues of the new reactors.
1.29	DG-1156	Seismic Design Classification	Changes are editorial, grammatical, referencing existing RG's, and one clarification of required analysis level for interface condition of seismic classifications.
1.57	DG-1158	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	RG being updated to add requirements already in effect and contained in SRP Section 3.8.2 and/or what is approved by the staff in SERs of LWRs and advanced reactors, e.g. AP1000 and CE System 80+.
1.68	DG-1166	Initial Test Programs for Water-Cooled Nuclear Power Plants	RG will be revised for editorial changes, not any technical changes. The technical information is more than adequate to accomodate testing for new reactors. RG will incorporate nomenclature references unique to the new reactor licensing process under Part 52, include a discussion of ITAAC, and provide additional information regarding testing associated with passive plant designs based on the staff's previous review and acceptance of the AP1000 application.
1.71	DG-1167	Welder Qualification for Areas of Limited Accessibility	RG addresses welder qualifications for areas with limited access. Current Inspection Manual procedures reference this RG; however, due to the design of new reactors the areas of inaccessibility are minimal. RG is being updated to be consistent with 50.55(a) and ASME standards. The technical basis for this RG was not revised, and the revisions will not impact the technical or policy issues of the new reactors.

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High Priority Regulatory Guides
Recommend Waiver of ACRS Review

RG #	DG#	Regulatory Guide Title	Rationale for why ACRS review is not needed
1.76	DG-1143 previously issued	Design Basis Tornado for Nuclear Power Plants	Revision 1 of RG was published in January 2006 as DG-1143. ACRS deferred review of DG-1143 until after public comments. Public comments have been received and changes have been made to the DG. Revision 0 of RG used two years of data and a simplified tornado model to determine tornado design requirements. RG is being changed to use tornado data from 1950 through 2003. The tornado model will now account for finite dimensions of structures as well as the variation of wind speeds along and across the tornado footprint. For finite structures, a tornado striking any point on the structure can cause damage. The original RG referenced a point model, where the power plant was assumed to be a point structure. Including the finite dimensions of structures in the revised model increases the tornado strike probability. This revision also utilizes the Enhanced Fujita scale issued by the National Weather Service in January 2006.
1.112	DG-1160	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	Changes are administrative in nature and include: changes/updates to text from old ANS 18.1 - 1975 standard to current version ANS 18.1 - 1999; changes to references from old Part 20 to current Part 20; and minor editorial updates in guide and appendix.
1.124	DG-1168	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	RG is being revised to reflect the requirements of the ASME B&PV Code, Section III, Division 1, 2001 Edition through the 2003 Addenda. ASME Code 2001 Edition and 2003 Addenda are endorsed by 10 CFR 50.55a(b)(1), published on January 1, 2006. RG is being revised to reflect changes in the ASME Code and to delete guidance supplanted by more detailed requirements found in the recent code edition and addenda. Since the changes to this RG solely reflect and are consistent with ASME code that is endorsed by 10CFR50.55a(b)(1), there is no need for ACRS to review the RG changes.
1.130	DG-1169	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	RG is being revised to reflect the requirements of the ASME B&PV Code, Section III, Division 1, 2001 Edition through the 2003 Addenda. ASME Code 2001 Edition and 2003 Addenda are endorsed by 10 CFR 50.55a(b)(1), published on January 1, 2006. RG is being revised to reflect changes in the ASME Code and to delete guidance supplanted by more detailed requirements found in the more recent code edition and addenda. Since the changes to this RG solely reflect and are consistent with ASME code that is endorsed by 10CFR50.55a(b)(1), there is no need for ACRS to review the Regulatory Guide changes.
1.136	DG-1159	Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments")	RG is being updated to add requirements already in effect and contained in SRP Section 3.8.1 and/or what is approved by the staff in SERs of LWRs and advanced reactor, e.g., ABWR, or ESBWR

High Priority Regulatory Guides
Recommend Waiver of ACRS Review

RG #	DG#	Regulatory Guide Title	Rationale for why ACRS review is not needed
1.189	DG-1170	Fire Protection for Operating Nuclear Power Plants	RG is being revised to provide additional regulatory guidance with respect to new reactor fire protection programs, as well as to incorporate the regulatory guidance included in recent generic letters and regulatory issue summaries issued by the Fire Protection Branch. There are no changes in stated staff positions. The changes expand on the guidance that was provided in revision 4 of SRP Section 9.5.1. In addition, all of the guidance in BTP 9.5-1 from the SRP is being moved into this RG. This is an administrative change and does not change any of the staff positions in either document. The Generic Letters and Regulatory Issue Summaries clarified regulatory expectations regarding operator manual actions, post-fire safe-shutdown circuit analyses, compensatory measures for fire protection program deficiencies, and cable raceway fire barriers.
1.196	DG-1171	Control Room Habitability at Light-Water Nuclear Power Reactors	Appendix B to RG was prepared as a sample technical specification for "Control Room Habitability At Light-Water Nuclear Power Reactors." The Appendix was to be removed when Technical Specification details were more carefully worked out with industry participation. The sample technical specification in Appendix has a few flaws and no utility has been granted the technical specification changes represented by Appendix B. If a utility were to request a technical specification change like Appendix B, the staff would not grant the request. Therefore, Appendix B and all references to it in Regulatory Guide 1.196 are being removed. Because the changes to RG do not represent new policy or staff position, there is no need for ACRS review of the changes.

September 5, 2006

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards (ACRS)
Advisory Committee on Nuclear Waste

FROM: David B. Matthews, Director
Division of New Reactor Licensing /RA/
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED REVISION TO NUREG-0800, STANDARD REVIEW PLAN

The purpose of this memorandum is to inform ACRS of NRR plans regarding the ongoing revision to NUREG-0800, Standard Review Plan (SRP), to identify SRP sections containing new or significantly modified staff positions (Enclosure 1), and to provide revision schedules for all SRP sections (Enclosure 2). It is intended that this plan will be used to facilitate early, focused ACRS interaction. NRR is publishing the SRP by March 2007 without providing it first as a draft for public comment. The staff is revising the SRP in this manner to provide a more timely, current SRP to support the requirement in 10 CFR 50.34(h) for a combined license applicant to evaluate its facility against the SRP in effect six months before the docket date of the application. The SRP will be available for public comment after issuance in March 2007. Staff will address any comments received after issuance in a subsequent SRP revision. Comment resolution may also be used to establish interim staff guidance prior to formal SRP revision.

Given the accelerated schedule to complete the revision to the SRP by March 2007, NRR does not plan to transmit all SRP section revisions to the ACRS for consideration. Instead, NRR plans to identify sections which contain either new staff positions or positions which have substantively changed since the 1996 draft SRP and subsequent revisions. Staff endorsement of content from the 1996 draft does not represent new staff positions. The basis for this determination is that the content contained in the 1996 draft has been used by the staff and stakeholders since issuance. For example, the 1996 draft SRP was incorporated by reference in Review Standard RS-001, Review Standard for Extended Power Uprates, was used to conduct new reactor design certification reviews; and provided the basis for Review Standard RS-002, Processing Applications for Early Site Permits.

Enclosure 1 lists the SRP sections which contain either new or substantially modified positions since the 1996 draft SRP and subsequent revisions. Enclosure 2 provides the schedule for the planned revision of each SRP section, including when the technical development is expected to be publicly available. NRR staff will work with ACRS to schedule subcommittee sessions to discuss the sections in Enclosure 1. To the extent possible, NRR and the Office of Nuclear Regulatory Research (RES) will combine relevant SRP sections with corresponding Regulatory Guide revisions to provide for a more efficient and effective ACRS review.

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The NRR staff will also support ACRS consideration of the technical content in the SRP sections not identified in Enclosure 1. If the ACRS determines that there is a need to review additional SRP sections, or if there are questions concerning this memorandum, please contact Robert Tregoning of my staff at 301 415-6657. NRR will notify ACRS staff of significant changes to either the schedule or scope of revisions for individual SRP sections.

Enclosures: 1. SRP Sections for ACRS Consideration
2. SRP Revision Schedule

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SRP Sections for ACRS Consideration

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2.3.1	Regional Climatology	10/31/2005	3/31/2007	Draft Revision issued for comment and is available on the web or in ADAMS - ML053570372. Comment period ended March 27, 2006. Revision coordinated with Standard Review Plan (SRP) Section 3.5.1.4 and Regulatory Guide (RG) 1.76 revisions.
2.3.3	Onsite Meteorological Measurements Programs	10/31/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Update references and regulatory citation. Revision coordinated with ongoing (concurrent) revision to RG 1.23.
2.4.6	Probable Maximum Tsunami Flooding	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Staff is evaluating recent tsunami data and will incorporate results of study within the SRP
2.5.2	Vibratory Ground Motion	1/31/2007	3/31/2007	Staff will revise Rev. 3, 1997, to include guidance on an performance-based approach to seismic hazards analysis; This will be based on lessons learned from experience with ESP (Clinton). Will follow revision to RG 1.165
3.2.1	Seismic Classification	11/15/2006	3/31/2007	Staff will revise the 1996 draft; the changes include some information previously not included in the SRP, but are not new staff positions. Specific changes are (1) add reference to 10 CFR Part 50, Appendix S, and state that surface deformation must be considered and that a list of SSCs necessary for continued operation during and following an operating basis earthquake (OBE) should be provided. (2) Add reference to Appendix R as it contains requirements to specifically consider seismic loading for meeting certain fire protection requirements. Update coordinated with revision to RG 1.29.
3.2.2	System Quality Group Classification	11/15/2006	3/31/2007	Staff will revise the 1996 draft; the changes include some information previously not included in the SRP, but are not new staff positions. Specific changes are (1) clarify that the provided lists of PWR and BWR fluid systems represent typical system names based on historical reviews of prior applications, are for general information purposes only, and may not be the same for passive LWR designs or non-LWR designs. (2) An SRM dated 7-21-93 for SECY 93-087 will be added as a reference. (3) The Figure A-1 illustration will be revised to more correctly show the main steam drain lines. Update coordinated with revision to RG 1.26.
3.12	ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design [new]	11/15/2006	3/31/2007	New section being developed to address piping systems and associated supports design. Will include (1) existing positions in SRP Sections 3.7.3 and 3.9 that are applicable to piping design including the current updated staff positions; (2) incorporation of Bulletin 88-08 and 88-11 criteria relative to thermal oscillations and thermal stratification; (3) incorporation of the staff position on ISLOCA; (4) additional staff positions taken in previous design certification reviews to supplement the piping design acceptance criteria (DAC); and (5) contains reference to DG-1144 issued in July 2006, which provides a new staff position to address environmental fatigue.
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	11/15/2006	3/31/2007	New section that addressed adequacy of applicant' submittal for design, material selection, fabrication, inspection and testing of threaded fasteners (resolution of Generic Safety Issue 29 and GL 91-17); changed title to delineate scope of threaded fasteners to ASMES Code Class 1, 2, and 3
4.2	Fuel System Design	12/15/2006	3/31/2007	Update will include interim acceptance criteria for reactivity-initiated accidents, that will supercede RG 1.77
6.2.5	Combustible Gas Control in Containment	9/30/2006	3/31/2007	This SRP revision was included in SECY-03-0127, "Final Rulemaking—risk-informed 10 CFR 50.44, "Combustible Gas Control in Containment;"" revision to RG 1.7 and will be administratively updated

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SRP Sections for ACRS Consideration

7.8	Diverse I&C Systems	11/24/2006	3/31/2007	Guidance on diverse instrumentation and control systems may be impacted by potential policy changes and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary confirming changes resulting from other SRP Chapter 7 updates
BTP 7-19	Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems	11/17/2006	3/31/2007	Guidance for evaluation of defense-in-depth and diversity in digital computer-based I&C systems may be impacted by potential policy changes and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary confirming changes resulting from other SRP Chapter 7 updates
9.1.1	New Fuel Storage	12/1/2006	3/31/2007	Revision will include guidance on 10 CFR 50.68, Criticality Accident Requirements
9.1.2	Spent Fuel Storage	11/15/2006	3/31/2007	1996 draft to be modified to: • Increased minimum spent fuel storage capacity to five years of spent fuel plus one full-core offload. • Added thermohydraulic considerations (i.e., no nucleate biling on fuel surface) for coolant flow through storage racks. • Specified maximum coolant inventory loss resulting from failure of a gate seal. • Organization of criticality will be located within Section 9.1.1. Coordinated with revision to RG 1.13
9.1.3	Spent Fuel Pool Cooling and Cleanup System	9/5/2006	3/31/2007	1996 draft updated as follows: removed acceptance criteria related to GDC 44, 45, and 46 as GDC 61 encompasses these criteria for this system; modified review procedures to reflect accepted practice; and administratively updated per LIC-200.
9.5.1	Fire Protection Program	10/15/2006	3/31/2007	Update coordinated with ongoing revision to RG 1.189. Update will also include references to recently issued applicable generic communications. This revision does not address NFPA 805.
11.2	Liquid Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also address the requirements of 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. The SRP will be updated administratively in accordance with LIC-200.
11.3	Gaseous Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also address the requirements of 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. The SRP will be updated administratively in accordance with LIC-200.
11.4	Solid Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also addressed the requirements 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. Also, the requirements from Chapter 16 Technical Specifications and RETS to those identified in Generic Letter 89-01, as implemented under the guidance of NUREG-1301 and NUREG-1302, will be updated. The SRP will be updated administratively in accordance with LIC-200.
12.3 - 12.4	Radiation Protection Design Features	1/6/2007	3/31/2007	Revision will reflect several revisions to 10 CFR Part 20 from the 1981 version of the SRP including 10 CFR 20.1406, update references to RGs, NUREGS, and standards, and be administratively updated in accordance with LIC-200

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SRP Sections for ACRS Consideration

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13.3	Emergency Planning	9/8/2006	3/31/2007	Revision will be issued for comment Sep 2006; ESP: Supp. 2 to NUREG-0654/FEMA-REP-1; DC: RG 1.101, NUREG-0696, NUREG-0737 (inc. Supp.1); COL RG 1.101; NUREG-0654/FEMA-REP-1, NUREG-0737 (inc. supp.1); COL Operational program SECY: Appendix E.IV.F.a: (1) full participation exercise within two years before issuance of first operating license for full power; and (2) onsite exercise within one year before issuance of operating license for full power. Appendix E.V: detailed implementing procedures submitted within 180 days prior to fuel load.
15.0	Accident Analysis - Introduction	1/5/2007	3/31/2007	
15.9 (new)	BWR Core Stability	1/5/2007	3/1/2007	Work in progress.
19.0	Probabilistic Risk Assessment	12/22/2006	3/1/2007	Revision to Chapter 19 will address staff review of COL plant specific PRA per proposed 10CFR 52.80, severe accidents per proposed 10CFR 52.79(a)(17) and 10CFR 79(a)(38) and will be based on application guidance contained in DG-1145. 19.0 will include guidance on severe guidance so there will not be a seperate 19.2. Also 19.1 will be referenced by 19.0 but will be updated persuant to RG 1.200 effort and schedule. Guidance on severe accidents is contained in Commission Policy

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SRP Revision Schedule

SRP Section	Section Title	Target for SRP Public Availability	Target Final Issuance Date	Anticipated Revision Changes
1.8	Interfaces for Standard Designs	11/15/2006	3/31/2007	This is a cross-cutting section primarily related to Design Certification (DC) reviews and COL referencing DCs; therefore there will be minimal discussion in Section C.I.1 of DG-1145, but there will be guidance on interfaces within Section in C.III.1 of the guide. This is a process section and contains no specific acceptance criteria
2.0	Site Parameter Envelope	11/15/2006	3/31/2007	This section is related to a COL referencing a DC or a DC and an Early Site Permit (ESP); therefore there will not be a corresponding section C.I.1 of DG-1145, but there will be a section in C.III.1 and C.III.2. This section is a process section and contains no specific acceptance criteria
2.1.1	Site Location and Description	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
2.1.2	Exclusion Area Authority and Control	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions. See NRC letter dated August 27, 2003 , (ML032120350) for additional information.
2.1.3	Population Distribution	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
2.2.3	Evaluation of Potential Accidents	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
2.3.1	Regional Climatology	10/31/2005	3/31/2007	Draft Revision issued for comment and is available on the web or in ADAMS - ML053570372. Comment period ended March 27, 2006. Revision coordinated with Standard Review Plan (SRP) Section 3.5.1.4 and Regulatory Guide (RG) 1.76 revisions.
2.3.2	Local Meteorology	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions) Update references and regulatory citations.
2.3.3	Onsite Meteorological Measurements Programs	10/31/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Update references and regulatory citation. Revision coordinated with ongoing (concurrent) revision to RG 1.23.
2.3.4	Short Term Dispersion Estimates for Accidental Atmospheric Releases	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Specific changes include adding text to address control room atmospheric dispersion factors, X/Q values (new to this revision, but refer to RG 1.194 (June 2003), which has been in use several years). Enhance discussion of staff check on methodology, inputs and assumptions used by applicant/licensee. The update does not contain any new staff positions.
2.3.5	Long Term Diffusion Estimates	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Specific change enhances discussion of staff check on methodology, inputs and assumptions used by applicant/licensee. The update does not contain any new staff positions.
2.4.1	Hydrologic Description	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.2	Floods	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions) with exception that reference to RG 1.59 will be supplemented with need to consider best engineering practice.
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)

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2.4.4	Potential Dam Failures	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.5	Probable Maximum Surge and Seiche Flooding	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.6	Probable Maximum Tsunami Flooding	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. Staff is evaluating recent tsunami data and will incorporate results of study within the SRP
2.4.7	Ice Effects	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.8	Cooling Water Canals and Reservoirs	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.9	Channel Diversions	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.10	Flooding Protection Requirements	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.11	Cooling Water Supply	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.12	Groundwater	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.4.14	Technical Specifications and Emergency Operation requirements	11/15/2006	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.5.1	Basic Geologic and Seismic Information	1/31/2007	3/31/2007	Staff will revise Rev. 3, 1997. The update will incorporate lessons learned from experience with ESP, and will be administratively updated per LIC-200. No new staff positions will be added
2.5.2	Vibratory Ground Motion	1/31/2007	3/31/2007	Staff will revise Rev. 3, 1997, to include guidance on an performance-based approach to seismic hazards analysis; This will be based on lessons learned from experience with ESP (Clinton). Will follow revision to RG 1.165
2.5.3	Surface Faulting	1/31/2007	3/31/2007	Staff will revise Rev. 3, 1997. The update will incorporate lessons learned from experience with ESP, and will be administratively updated per LIC-200. No new staff positions will be added
2.5.4	Stability of Subsurface Materials and Foundations	1/31/2007	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
2.5.5	Stability of Slopes	1/31/2007	3/31/2007	Staff will revise the 1996 draft. The updates will incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. No major technical changes.(i.e., stay with existing positions)
3.2.1	Seismic Classification	11/15/2006	3/31/2007	Staff will revise the 1996 draft; the changes include some information previously not included in the SRP, but are not new staff positions. Specific changes are (1) add reference to 10 CFR Part 50, Appendix S, and state that surface deformation must be considered and that a list of SSCs necessary for continued operation during and following an operating basis earthquake (OBE) should be provided. (2) Add reference to Appendix R as it contains requirements to specifically consider seismic loading for meeting certain fire protection requirements. Update coordinated with revision to RG 1.29.

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3.2.2	System Quality Group Classification	11/15/2006	3/31/2007	Staff will revise the 1996 draft; the changes include some information previously not included in the SRP, but are not new staff positions. Specific changes are (1) clarify that the provided lists of PWR and BWR fluid systems represent typical system names based on historical reviews of prior applications, are for general information purposes only, and may not be the same for passive LWR designs or non-LWR designs. (2) An SRM dated 7-21-93 for SECY 93-087 will be added as a reference. (3) The Figure A-1 illustration will be revised to more correctly show the main steam drain lines. Update coordinated with revision to RG 1.26.
3.3.1	Wind Loadings	11/15/2006	3/31/2007	No new staff positions, admin update per LIC-200
3.3.2	Tornado Loadings	3/1/2007	3/31/2007	No new staff positions, admin update per LIC-200
3.4.1	Flood Protection for Onsite Equipment Failure	3/1/2007	3/31/2007	1996 draft technically acceptable - admin update except update will: • Clarify the review areas for internal flooding to include the following: a. pipe breaks from non-seismic moderate energy lines per GDC 2, and b. pipe breaks from high energy lines determined by SRP Sections 3.6.1 and 3.6.2 per GDC 4; • Clarify the review areas to identify flow paths between interconnected rooms that might cause flooding of the rooms housing safety-related SSCs from the fluid in nonsafety-related rooms.
3.4.2	Analysis Procedures	3/1/2007	3/31/2007	No new staff positions, admin update per LIC-200
3.5.1.1	Internally Generated Missiles (Outside Containment)	3/1/2007	3/31/2007	1996 draft technically acceptable - admin update
3.5.1.2	Internally Generated Missiles (Inside Containment)	3/1/2007	3/31/2007	1996 draft technically acceptable - admin. Update
3.5.1.3	Turbine Missiles	2/1/2007	3/31/2007	Update will include staff position in NUREG-0887, Supplement 3, Safety Evaluation for the Perry Nuclear Plant, regarding probability calculations of turbine missile generation.
3.5.1.4	Missiles Generated by Natural Phenomena	10/31/2005	3/31/2007	Draft Revision issued for comment and is available on the web or in ADAMS - ML053570376. Comment period ended March 27, 2006. Currently resolving public comments. Revision coordinated with SRP Section 2.3.1 and RG 1.76 revisions.
3.5.1.5	Site Proximity Missiles (Except Aircraft)	10/31/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
3.5.1.6	Aircraft Hazards	10/31/2006	3/31/2007	Staff will revise the 1996 draft. The update will improve clarity, incorporate RS-002 and lessons learned from experience with ESP, and reference 10 CFR Part 52. The update does not contain any new staff positions
3.5.2	Structures, Systems, and Components To Be Protected From Externally Generated Missiles	1/15/2007	3/31/2007	1996 draft technically acceptable - administrative update
3.5.3	Barrier Design Procedures	1/31/2007	3/31/2007	No new staff positions, admin update per LIC-200; except update may need conforming changes resulting from revision to RG 1.76 and SRP Section 3.5.1.4
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1/15/2007	3/31/2007	1996 draft technically acceptable - admin update except update will: Draft Revision 3 to SRP Section 3.6.1 (1996 version) proposed inappropriate revisions to Appendices B and C of BTP SPLB 3-1. These are historical documents included for reference, and should not be revised. • Draft Revision 3 to SRP 3.6.1 (1996 version) proposed inappropriate deletion of most of the implementation subsection. This information is important in identifying the appropriate review criteria for current operating reactors, and should not be deleted. Clarify that moderate energy piping that is not seismically supported should be evaluated for full circumferential ruptures per GDC 2.

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SRP Revision Schedule

SRP Section	Section Title	Target for SRP Public Availability	Target Final Issuance Date	Anticipated Revision Changes
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	11/15/2006	3/31/2007	1996 draft technically acceptable - admin update performed in accordance with LIC-200 with one exception regarding currently acceptable procedures for assessing the forces induced by jets emanating from postulated piping breaks on neighboring systems, structures, and components, along with acceptable means of modeling jet expansion (which determine the spatial zones of influence of the loads within expanding jets). Several inaccuracies that may lead to nonconservative assessments of the strength, zone of influence, and space and time-varying nature of the loading effects of supersonic expanding jets on neighboring structures were raised by the ACRS [Wallis - ADAMS ML050830344, Ransom - ADAMS ML050830341] and ACRS Safety Evaluation letters to the Chairman of the NRC (ACRSR-2097 - ML042920334, and ACRSR-2110 ML043450346). Staff is currently assessing this issue in SRP Section 3.6.2 and ANSI/ANS 58.2. Until the update is complete, staff will review jet related issues on a case by case basis.
3.6.3	Leak-Before-Break Evaluation Procedures	11/15/2006	3/31/2007	The revision will not introduce new staff positions from with the previous SRP and other agency guidance. Administrative update per LIC-200
3.7.1	SEISMIC DESIGN PARAMETERS	1/15/2007	3/31/2007	Work in progress.
3.7.2	SEISMIC SYSTEM ANALYSIS	2/15/2007	3/31/2007	Work in progress.
3.7.3	SEISMIC SUBSYSTEM ANALYSIS	1/15/2007	3/31/2007	Work in progress.
3.7.4	SEISMIC INSTRUMENTATION	2/15/2007	3/31/2007	Work in progress.
3.8.1	Concrete Containment	1/15/2007	3/31/2007	Work in progress.
3.8.2	Steel Containment	1/15/2007	3/31/2007	Work in progress.
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1/15/2007	3/31/2007	Work in progress.
3.8.4	Other Seismic Category I Structures	1/15/2007	3/31/2007	Work in progress.

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SRP Revision Schedule

SRP Section	Section Title	Target for SRP Public Availability	Target Final Issuance Date	Anticipated Revision Changes
3.8.5	Foundations	1/15/2007	3/31/2007	Work in progress.
3.9.1	Special Topics for Mechanical Components	11/15/2006	3/31/2007	1996 draft technically acceptable with the addition of reference to Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," which clarifies of revises requirements for consideration of "operating basis earthquakes." Administrative update to be performed in accordance with LIC-200.
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	01/15/2007	3/31/2007	This revision will (1) add reference to Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants", which clarifies the revised requirements for consideration of "operating basis earthquakes," (2) provide an interface to SRP Section 3.10, regarding the methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment, (3) add some clarification regarding general design criteria contained in the acceptance criteria. Section will be administratively updated per LIC-200.
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	11/15/2006	3/31/2007	1996 draft technically acceptable with the addition of reference to Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," which clarifies of revises requirements for consideration of "operating basis earthquakes." Administrative update to be performed in accordance with LIC-200.
3.9.4	Control Rod Drive Systems	6/20/2006	3/31/2007	Technically Complete see: ML060470198
3.9.5	Reactor Pressure Vessel Internals	01/15/2007	3/31/2007	The section will be expanded to emphasize the guidance for review of the design of all reactor internal components (including the steam dryer of a Boiling Water Reactor (BWR)) for potential adverse flow effects (flow-induced vibrations and acoustic resonances). The details of acceptance criteria and review procedures will be specified.
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	11/15/2006	3/31/2007	The title of the SRP section has been modified from "Inservice Testing of Pumps and Valves" to reflect the revision of scope to include functional design and qualification, and inservice testing programs for pumps, valves, and dynamic restraints (snubbers).
[redacted]				
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	11/15/2006	3/31/2007	The update will: (1) transfer the review responsibility of some aspects of "Qualification for Equipment Functionality" (for pumps and valves) to SRP Section 3.9.6. (2) Add a provision to the SRP regarding review guidance if Qualification by Experience is proposed in an application, specifically the SRP will state that the details of the experience database including the procedures for ensuring the adequate qualification of equipment should be submitted for staff review and approval at the construction permit (CP) stage or design certification (DC) stage. If the DC is referenced in an application, similar information for equipment not covered in the DC should be submitted for staff review and approval at the operating license (OL) stage or combined operating license (COL) stage.
3.11	Environmental Qualification of Mechanical and Electrical Equipment	11/15/2006	3/31/2007	No new staff position. Updates consist of review requirements for implementation milestones for COL application's EQ program, consistent with SECY-05-0197 for operational programs; and incorporating current regulatory guidance and standards (10 CFR50.34 (f)(2)(ix), 10CFR50.67, RG1.183., IEEE-323, RG 1.180, RG 130, RG 189). Overall administrative update.

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3.12	ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design [new]	11/15/2006	3/31/2007	New section being developed to address piping systems and associated supports design. Will include (1) existing positions in SRP Sections 3.7.3 and 3.9 that are applicable to piping design including the current updated staff positions; (2) incorporation of Bulletin 88-08 and 88-11 criteria relative to thermal oscillations and thermal stratification; (3) incorporation of the staff position on ISLOCA; (4) additional staff positions taken in previous design certification reviews to supplement the piping design acceptance criteria (DAC); and (5) contains reference to DG-1144 issued in July 2006, which provides a new staff position to address environmental fatigue.
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	11/15/2006	3/31/2007	New section that addressed adequacy of applicant' submittal for design, material selection, fabrication, inspection and testing of threaded fasteners (resolution of Generic Safety Issue 29 and GL 91-17); changed title to delineate scope of threaded fasteners to ASMES Code Class 1, 2, and 3
4.2	Fuel System Design	12/15/2006	3/31/2007	Update will include interim acceptance criteria for reactivity-initiated accidents, that will supercede RG 1.77
4.3	Nuclear Design	12/15/2006	3/31/2007	Work in progress.
4.4	Thermal and Hydraulic Design	12/15/2006	3/31/2007	Work in progress.
4.5.1	Control Rod Drive Structural Materials	11/30/2006	3/31/2007	1996 draft technically acceptable; no new staff positions - administrative update, may include editorial changes such as updating references
4.5.2	Reactor Internal and Core Support Materials	11/15/2006	3/31/2007	Work in progress.
4.6	Functional Design of Control Rod Drive System	12/1/2006	3/31/2007	Update will incorporate lessons learned from new reactor designs. These changes will not result in new staff positions.
5.2.1.1	Compliance With the Codes and Standards Rule, 10 CFR 50.55a	2/1/2007	3/31/2007	1996 draft technically acceptable - administrative update
5.2.1.2	Applicable Code Cases	2/1/2007	3/31/2007	Changes include the consolidation of Regulatory Guide 1.84 and 1.85 into RG 1:84 for the design, fabrication, and materials code case acceptability, ASME Section III Class 1, 2 and 3 components. The review will update the section to reflect the current NRC accepted code cases in NRC Regulatory Guide 1.84, Revision 33, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (August 2005); NRC Regulatory Guide 1.147 (Revision 0-February 1981), including Revision 1 through Revision 14 (August 2005), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"; Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code" (June 2003); and Regulatory Guide 1.193, Revision 1, "ASME Code Cases Not Approved for Use." These Regulatory Guides have been approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. The update will extend the applicability to Part 52.
5.2.2	Overpressure Protection	1/15/2007	3/31/2007	Work in progress.
5.2.3	Reactor Coolant Pressure Boundary Materials	6/27/2006	3/31/2007	Technically complete see: ML053500353
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	10/30/2006	3/31/2007	Work in progress.

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5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1/15/2007	3/31/2007	1996 draft to be modified to • Incorporate new staff position in Revision 1 to RG 1.45; • Change the required leakage detection instrumentation in the plant Technical Specifications (TS) to exclude gaseous radiation monitor due to its reduced sensitivity as a result of the recent advance in fuel performance. • Add operator actions for leakage limits below TS specification for identification and localization of RCS leakage to avoid long term low level leakage.
5.3.1	Reactor Vessel Materials	6/27/2006	3/31/2007	Technically complete see: ML053500353
5.3.2	Pressure Temperature Limits and Pressurized Thermal Shock	12/5/2006	3/31/2007	Update will reflect ASME Code Cases N-588, N-640, and N-641. These Code Cases have subsequently has been included in the ASME Code. Update the document to reflect 10 CFR 50.61 requirements. The earlier SRP 5.3.2 did not address the PTS issue. Neither of these changes represent new staff positions.
5.3.3	Reactor Vessel Integrity	6/27/2006	3/31/2007	Technically complete see: ML053500353
5.4	Components and Subsystem Design	1/15/2007	3/31/2007	No technical acceptance criteria contained in 5.4. Section contains organization review responsibilities for the the subsections of 5.4
5.4.1.1	Pump Flywheel Integrity (PWR)	11/15/2006	3/31/2007	Update will combined Review Areas 1, "Material Selection," and 2, "Fracture Toughness" to add technical clarity regarding material fracture toughness requirements. Add appropriate references and discussion on RG 1.14; replace outdated requirements for ensuring adequate fracture toughness of the pump flywheel; add a new paragraph regarding fracture mechanics analysis to connect SRP fracture toughness to the driving force discussed in RG 1.14; and make other minor revisions to enhance consistency of technical guidance through out the SRP sections. In addition staff is reducing inspection frequency from 3 per 10-year ISI interval to 1 per 10-year based on approved WOG topical Report WCAP-14535 and CEOG TR SIR-94-080.
5.4.2.1	Steam Generator Materials	11/15/2006	3/31/2007	The entire SRP section 5.4.2.1, Rev.1 was revised to remove redundancy to incorporate all of the applicable Commission Regulations, to expand on acceptable approaches for satisfying the applicable regulations, and to incorporate the appropriate regulatory guidance from SRP Section 5.2.3. Specifically, added reference to (1) General Design Criteria (GDC) 4 since steam generators are important to safety and must be designed for dynamic effects; (2) GDC 30 since steam generators form part of the reactor coolant pressure boundary and must be designed and fabricated to the highest quality standards; (3) 10 CFR 50.55a since the steam generators must be constructed in accordance with the ASME Code; (4) 10 CFR 50, Appendix B since quality assurance requirements apply to the pressure boundary and can be fabricated with ferritic materials; and (5) 10 CFR 52 since licensing can occur under 10 CFR part 50 or Part 52.

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5.4.2.2	Steam Generator Tube Inservice Inspection	11/15/2006	3/31/2007	The entire SRP section was revised to remove redundancy to incorporate all of the applicable Commission Regulations, to expand on acceptable approaches for satisfying the applicable regulations, and to remove reference to RG 1.83. The pertinent recommendations from RG 1.83 were incorporated directly into the SRP. Specifically added reference to (1) 50.55a (Codes and Standards since the ASME code contains requirements that are applicable to the perfomance of steam generator tube inspections; (2) 50.36 (Technical Specifications) since the content of the steam generator technical specifications is addressed in 50.36; (3) 10 CFR 50, Appendix B since Appendix B contains requirements pertinent to the performance of non-destructive examination and requires corrective actions to be taken under specific circumstances; (4) 50.65 (Maintenance Rule) since the steam generator tubes are safely related; and (5) 10 CFR 52 since licensing can occur under 10 CFR Part 50 or Part 52.
5.4.6	Reactor Core Isolation Cooling System (BWR)	11/15/2006	3/31/2007	Work in progress.
5.4.7	Residual Heat Removal (RHR) System	12/1/2006	3/31/2007	Work in progress.
5.4.8	Reactor Water Cleanup System (BWR)	10/30/2006	3/31/2007	Work in progress.
5.4.11	Pressurizer Relief Tank	12/1/2006	3/31/2007	Work in progress.
5.4.12	Reactor Coolant System High Point Vents	1/15/2007	3/31/2007	Work in progress.
6.1.1	Engineered Safety Features Materials	7/26/2006	3/31/2007	Technically complete see: ML061370411
6.1.2	Protective Coating Systems (Paints)□Organic Materials	11/15/2006	3/31/2007	1996 draft technically acceptable with the following changes: 1) Replace ASTM D3842 with ASTM D5144. Standard D3842 was replaced with D5144 by ASTM in 1995, and subsequently updated in 2000, and 2) Add a discussion of periodic coating assessment to the technical rationale. This discussion will describe the value of routine coating assessments to ensure the coatings have not degraded. - per RG 1.54 adn the ASTM standards it endorses. These changes do not represent new staff positions.
6.2.1	Containment Functional Design	12/1/2006	3/31/2007	1996 draft technically acceptable; however will update list of containment analysis codes for all of Section 6.2.1 - administrative update
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	11/15/2006	3/31/2007	1996 draft technically acceptable - administrative update
6.2.1.1.C	Pressure-Suppression Type BWR Containments	12/1/2006	3/31/2007	1996 draft technically acceptable - administrative update

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6.2.1.2	Subcompartment Analysis	12/1/2006	3/31/2007	Update based on 1996 draft version. Update will include interface with SRP Section 3.6.3 regarding review of leak-beforebreak analyses as they may apply to containment subcompartment; Update list of containment analysis computer codes; and administrative update including applicability of Part 52.
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss of Coolant Accidents	9/1/2005	3/31/2007	Revision 2 was published in January 2006 and is available on Web or in ADAMS: ML060150002. Any update would be administrative in nature
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	12/1/2006	3/31/2007	Update based on 1996 draft version. Revision will update list of containment analysis computer codes; and administrative update including applicability of Part 52.
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	12/1/2006	3/31/2007	1996 draft technically acceptable - administrative update
6.2.2	Containment Heat Removal Systems	9/30/2006	3/31/2007	Update will include 1) RG 1.82, Rev. 3 for (i) NRC Position on NPSH determination for ECCS and containment heat removal pumps, (ii) NRC Positions on blockage of PWR sump screens and BWR ECCS suction strainers; delete reference to RG 1.1 since it contradicts RG 1.83, Rev.3; reference to NEDO 32686-A for guidance on BWR ECCS suction strainer blockage; NEI-04-07 and letter to NEI on staff position on NEI-04-07 on PWR sumps; GL 2004-02; and AP1000 passive containment cooling and the FSER on the AP1000.
6.2.3	Secondary Containment Functional Design	1/30/2007	3/31/2007	1996 draft technically acceptable - administrative update
6.2.4	Containment Isolation System	1/30/2007	3/31/2007	see 6.2.1
6.2.5	Combustible Gas Control in Containment	9/30/2006	3/31/2007	This SRP revision was included in SECY-03-0127, "Final Rulemaking—risk-informed 10 CFR 50.44, "Combustible Gas Control in Containment;"" revision to RG 1.7 and will be administratively updated
6.2.6	Containment Leakage Testing	12/1/2006	3/31/2007	This update directly related to existing reactors and is dependent on the Integrated Leak Rate Testing (ILRT) NEI task group. For new reactors, it is referred to as an Operational Program
6.2.7	Fracture Prevention of Containment Pressure Boundary	12/1/2006	3/31/2007	Update to add a new item to discuss findings pertinent to ASME Code Section III, Article NE-2300 and provided for a contingent finding based on whether materials were fracture toughness tested.
6.3	Emergency Core Cooling System	12/1/2006	3/31/2007	Update will be coordinated with SRP section 15.6.5.
6.4	Control Room Habitability System	12/31/2006	3/31/2007	Revision will combine guidance contained in SRP Section 9.4.1
6.5.1	ESF Atmosphere Cleanup Systems	12/1/2006	3/31/2007	see 6.2.1
6.5.2	Containment Spray as a Fission Product Cleanup System .	12/31/2005	3/31/2007	Revision 3 was published December 2005 and is available on Web or in ADAMS: ML060150001
6.5.3	Fission Product Control Systems and Structures	12/5/2006	3/31/2007	1996 draft technically acceptable - administrative update

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6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	12/5/2006	3/31/2007	1996 draft technically acceptable. Will update reference to RGs. Change RG 1.3 to RG1.183 and RG 1.195
6.6	Inservice Inspection of Class 2 and 3 Components	10/30/2006	3/31/2007	Work in progress.
[illegible]				
7.0	I&C Overview of Review Process	11/3/2006	3/31/2007	Overview section contains no technical acceptance criteria will be administratively updated per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
7.0-A	Review Process for Digital I&C Systems	11/9/2006	3/31/2007	Guidance on the review process for digital I&C will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.1	Instrumentation and Controls – Introduction	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
7.1-A	Acceptance Criteria & Guidelines for I&C Systems Important to Safety	11/27/2006	3/31/2007	General acceptance criteria for I&C will only be administratively updated per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
7.1-B	Guidance for Evaluation of Conformance to IEEE Std - 279	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
7.1-C	Guidance for Evaluation of Conformance to IEEE Std - 603	11/3/2006	3/31/2007	Guidance on the criteria for safety systems will have information on digital I&C moved to Appendix 7.1-D and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
-D (new)	Guidance for Evaluation of Conformance to IEEE Std - 7-4.3.2	11/3/2006	3/31/2007	New appendix providing guidance on the use of computers in safety systems which is being moved from Appendix 7.1-C and will include information based on Regulatory Guide 1.152, Rev. 2 and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.1-T	Table 7-1 Acceptance Criteria	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
7.2	Reactor Trip System	11/24/2006	3/31/2007	Guidance on reactor trip systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.3	Engineered Safety Features Systems	11/13/2006	3/31/2007	Guidance on engineered safety features systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.4	Safe Shutdown Systems	10/20/2006	3/31/2007	Guidance on safe shutdown systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.5	Information Systems Important to Safety	11/27/2006	3/31/2007	Guidance on information systems important to safety will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.6	Interlock Systems Important to Safety	11/9/2006	3/31/2007	Guidance on interlock systems important to safety will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.7	Control Systems	11/9/2006	3/31/2007	Guidance on control systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
7.8	Diverse I&C Systems	11/24/2006	3/31/2007	Guidance on diverse instrumentation and control systems may be impacted by potential policy changes and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary confirming changes resulting from other SRP Chapter 7 updates
7.9	Data Communications Systems	10/20/2006	3/31/2007	Guidance on data communication systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
App 7-A	Branch Technical Positions - (21)	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
App 7-B	General Agenda, Station Site visits	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates

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App 7-C	Acronyms, Abbreviations, Glossary, and Index	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-1	Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-2	Guidance on Requirements of Motor Operated Valves in the Emergency Core Cooling System Accumulator Lines	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-3	Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps out of Service	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-4	Guidance on Design Criteria for Auxiliary Feedwater Systems	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-5	Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-6	Guidance on I&C Provided to Accomplish Changeover from Injection to Recirculation Mode	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-8	Guidance for Application of Regulatory Guide 1.22	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-9	Guidance on Requirements for Reactor Protection System Anticipatory Trips	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-10	Guidance on Application of Regulatory Guide 1.97	12/18/2006	3/31/2007	Guidance on application of Regulatory Guide 1.97 which will include information based Regulatory Guide 1.97, Rev. 4 and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-11	Guidance on Application and Qualification of Isolation Devices	11/3/2006	3/31/2007	Guidance on application and qualification of isolation devices will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-12	Guidance on Establishing and Maintaining Instrument Setpoints	11/20/2006	3/31/2007	Guidance on establishing and maintaining instrument setpoints may be impacted by potential policy changes on setpoint methodology including Regulatory Information Summary 2006-17 and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary confirming changes resulting from other SRP Chapter 7 updates
BTP 7-13	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	2/15/2007	3/31/2007	Administrative update per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-14	Guidance on Software Reviews for Digital Computer-Based I&C Systems	11/3/2006	3/31/2007	Guidance on software reviews for digital computer-based I&C will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-16	Guidance on Level of Effort Required for Design Certification Applications Under 10 CFR Part 52	12/24/2006	3/31/2007	Guidance on the level of detail required for design certification applications under 10 CFR Part 52 may be deleted as most if not all of the information from this BTP is being transferred to DG-1145 and will be updated per LIC-200 and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-17	Guidance on Self-Test and Surveillance Test Provisions	10/20/2006	3/31/2007	Guidance on self-test and surveillance test provisions will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-18	Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	11/13/2006	3/31/2007	Guidance on the use of programmable logic controllers in digital computer-based I&C systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates
BTP 7-19	Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems	11/17/2006	3/31/2007	Guidance for evaluation of defense-in-depth and diversity in digital computer-based I&C systems may be impacted by potential policy changes and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary confirming changes resulting from other SRP Chapter 7 updates
BTP 7-21	Guidance on Digital Computer Real-Time Performance	11/9/2006	3/31/2007	Guidance on digital computer real-time performance, may add information on digital sampling and digital operating system time if not in separate BTPs and will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from other SRP Chapter 7 updates

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BTP 7-22	Guidance on Digital Sampling	11/20/2006	3/31/2007	New proposed BTP on digital sampling in computers will be written in accordance with LIC-200, updates to referenced regulatory guides and standards, and necessary conforming information resulting from other SRP Chapter 7 updates. However, this information may instead be included in existing BTP-21
BTP 7-23	Guidance on Digital Operating System Timing	12/1/2006	3/31/2007	New proposed BTP on digital operating system time in computer systems will be written in accordance with LIC-200, updates to referenced regulatory guides and standards, and necessary conforming information resulting from other SRP Chapter 7 updates. However, this information may instead be included in existing BTP-21.
8.1	Electric Power / Introduction	11/1/2006	3/31/2007	Work in progress.
8.2	Offsite Power System	11/1/2006	3/31/2007	No new staff position. Updates consist of incorporating current regulatory guidance and standards (BTP ICSB 11 , GL 2006-02, RG 1.204, 10 CFR 50.63, BTP PSB-1, NUREG-1793, DG-1145 , IN 2002-12, RG 1.155). Note 8.2 Appendix B is subsumed into new SRP Section 8.4. Administrative update per LIC-200
8.3.1	A C Power Systems (Onsite)	11/1/2006	3/31/2007	Work in progress.
8.3.2	D C Power Systems (Onsite)	11/1/2006	3/31/2007	Work in progress.
8.4 NEW	Station Blackout	11/1/2006	3/31/2007	New SRP section that provides guidance related to the review of an applicant or licensee's overall conformance with the requirements of 10 CFR 50.63 "Loss of All Alternating Current Power" and describes approaches that the staff has found acceptable for meeting the requirements of the rule.
8-A	Branch Technical Positions (PSB)	11/1/2006	3/31/2007	Work in progress.
8-B	General Agenda, Station Site Visits	11/1/2006	3/31/2007	No new staff position. Updates consist of adding aditional agenda items based on current regulatory guidance and standards (IN 2002-12, RG 1.204, RG 1.180, IEEE 1050-1996, IEEE 603-1998, SECY 05-0219 Attachment #2, GL 2006-02, IN 97-05, IN 98-07, IEEE C37.013-1997, and NUREG 1793). Overall administrative update.
9.1.1	New Fuel Storage	12/1/2006	3/31/2007	Revision will include guidance on 10 CFR 50.68, Criticality Accident Requirements
9.1.2	Spent Fuel Storage	11/15/2006	3/31/2007	1996 draft to be modified to: • Increased minimum spent fuel storage capacity to five years of spent fuel plus one full-core offload. • Added thermohydraulic considerations (i.e., no nucleate biling on fuel surface) for coolant flow through storage racks. • Specified maximum coolant inventory loss resulting from failure of a gate seal. • Organization of criticality will be located within Section 9.1.1. Coordinated with revision to RG 1.13
9.1.3	Spent Fuel Pool Cooling and Cleanup System	9/5/2006	3/31/2007	1996 draft updated as follows: removed acceptance criteria related to GDC 44, 45, and 46 as GDC 61 encompasses these criteria for this system; modified review procedures to reflect accepted practice; and administratively updated per LIC-200.
9.1.4	Light Load Handling System (Related to Refueling)	12/15/2006	3/31/2007	1996 draft technically acceptable - administrative update
9.1.5	Overhead Heavy Load Handling Systems	12/15/2006	3/31/2007	1996 draft to be modified to: • Endorse ASME NOG-1 2004 criteria for Type 1 Cranes as acceptable for use in a single failure proof heavy load handling system. • Revise guidance regarding slings for use in single failure proof handling systems to specify wire rope or chain slings. • Update CMAA-70 and ASME B30.2 and B30.9 to the current versions. • Clarify implementation of NUREG-0612 guidance.

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9.2.1	Station Service Water System	1/15/2007	3/31/2007	1996 draft to be modified to: • Add GL 96-06 as a reference and provide additional waterhammer and two-phase flow review guidance. • Eliminate the review guidance related to implementation of TMI Action Plan Item II.K.1.22 and IEB 79-08 for BWRs added in the 1996 Draft Revision as this does not apply to sevice water(applies to RCIC); • Eliminate review guidance that is redundant to and/or more suitably addressed by other SRP sections (such as seismic design criteria).
9.2.2	Reactor Auxiliary Cooling Water Systems	1/15/2007	3/31/2007	1996 draft to be modified to: • Add GL 96-06 as a reference and provide additional waterhammer and two-phase flow review guidance. Eliminate specific reference to 10 CFR 50.34(f)(1)(iii) since it was applicable only to certain specific applications that were pending as of February 16, 1982. • Eliminate review guidance that is redundant to and/or more suitably addressed by other SRP sections (such as seismic design criteria).
9.2.3	Demineralized Water Makeup System	1/15/2007	3/31/2007	1996 draft technically acceptable - administrative update
9.2.4	Potable and Sanitary Water Systems	1/15/2007	3/31/2007	1996 draft technically acceptable - administrative update
9.2.5	Ultimate Heat Sink	1/15/2007	3/31/2007	1996 draft to be modified to: • Replace Branch Technical Position ASB 9-2 with reference to appropriate industry standard for determining decay heat (e.g., ANSI/ANS 5.1 or ORIGEN). • Eliminate review guidance that is redundant to and/or more suitably addressed by other SRP sections (such as seismic design criteria and criteria for determining cooling capability of reservoirs and ponds).
9.2.6	Condensate Storage Facilities	1/15/2007	3/31/2007	1996 draft to be modified to: • Specify that coatings and floating tank covers whose failure could result in blockage of the AFW suction pipe should not be used in the condensate storage tank.
9.3.1	Compressed Air System	1/15/2007	3/31/2007	1996 draft technically acceptable - admin. Update
9.3.2	Process and Post Accident Sampling Systems	1/15/2007	3/31/2007	1996 draft to be revised to provide an alternative to the post-accident sampling system (PASS), to replace reference to RG 1.56 "Maintenance of Water Purity in Boiling Water Reactors," with EPRI water chemistry guidelines, update references (WCAP-14986-P,Rev1, CE NPSD-1157 Rev 1, NUREG -1793), and administratively update per LIC-200
9.3.3	Equipment and Floor Drainage System	1/15/2007	3/31/2007	important only w/rt flood protection; 1996 draft technically acceptable - admin. Update
9.3.4	Chemical and Volume Control System (PWR) Including Boron Recovery System)	1/15/2007	3/31/2007	1996 draft technically acceptable - admin. Update
9.3.5	Standby Liquid Control System (BWR)	1/15/2007	3/31/2007	Work in progress.
9.4.1	Control Room Area Ventilation System	1/15/2007	3/31/2007	Guidance will be combined with SRP Section 6.4.
9.4.2	Spent Fuel Pool Area Ventilation System	1/15/2007	3/31/2007	With alternative source terms, systems more defense-in-depth.
9.4.3	Auxiliary and Radwaste Area Ventilation System	1/15/2007	3/31/2007	With alternative source terms, systems more defense-in-depth
9.4.4	Turbine Area Ventilation System	1/15/2007	3/31/2007	With alternative source terms, systems more defense-in-depth
9.4.5	Engineered Safety Feature Ventilation System	12/1/2006	3/31/2007	ESBWR and AP1000 reviews to inform update.
9.5.1	Fire Protection Program	10/15/2006	3/31/2007	Update coordinated with ongoing revision to RG 1.189. Update will also include references to recently issued applicable generic communications. This revision does not address NFPA 805.

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9.5.2	Communications Systems	12/4/2006	3/31/2007	Guidance on communications systems will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from SRP Chapter 7 updates
9.5.3	Lighting Systems	11/1/2006	3/31/2007	No new staff position. Technical updates consist of deleting reference to onsite power sources, revising review procedures to include an engineering assessment, and adding reference NUREG-0700, "Human-System Interface Design review Guidelines," Rev. 2, May, 2002. Overall administrative update.
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	1/19/2007	3/31/2007	RTNSS for AP1000; 1996 draft technically acceptable - admin. Update - except: • This SRP section references ANSI Standard N195, "Fuel Oil Systems for Standby Diesel Generators," (1976) as an acceptance criterion. RG 1.137, endorses ANSI N195 and supplements it with Positions C.1 and C.2 of the RG. However, ANSI N195 has been now been approved as ANSI/ANS 59.51-1997, "Fuel Oil Systems for Safety-Related Emergency Diesel Generators." This 1997 standard seems to address all the positions identified in RG 1.137. Any SRP references to ANSI N195 should be changed to ANSI/ANS 59.51 and RG 1.137 should be withdrawn or revised as necessary.
9.5.5	Emergency Diesel Engine Cooling Water System	1/19/2007	3/31/2007	RTNSS for AP1000; 1996 draft technically acceptable - admin. Update
9.5.6	Emergency Diesel Engine Starting System	1/19/2007	3/31/2007	RTNSS for AP1000; 1996 draft technically acceptable - admin. Update
9.5.7	Emergency Diesel Engine Lubrication System	1/19/2007	3/31/2007	RTNSS for AP1000; 1996 draft technically acceptable - admin. Update
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	1/19/2007	3/31/2007	RTNSS for AP1000; 1996 draft technically acceptable - admin. Update
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NEW	ESBWR Isolation Condenser	12/1/2006	3/31/2007	Work in progress.
10.2	Turbine Generator	1/19/2007	3/31/2007	1996 draft technically acceptable - admin. Update
10.2.3	Turbine Rotor Integrity	10/15/2006	3/31/2007	Update will Clarify the examinations for the welded surfaces in Revision 1.
10.3	Main Steam Supply System	1/19/2007	3/31/2007	GDC 34/44 DID; 1996 draft technically acceptable - admin. Update
10.3.6	Steam and Feedwater System Materials	10/15/2006	3/31/2007	Work in progress.
10.4.1	Main Condensers	1/19/2007	3/31/2007	1996 draft technically acceptable - admin. Update
10.4.2	Main Condenser Evacuation System	1/19/2007	3/31/2007	1996 draft technically acceptable - admin. Update
10.4.3	Turbine Gland Sealing System	1/19/2007	3/31/2007	1996 draft technically acceptable - admin. Update
10.4.4	Turbine Bypass System	1/19/2007	3/31/2007	GDC 34/44 DID; 1996 draft technically acceptable - admin. Update
10.4.5	Circulating Water System	1/19/2007	3/31/2007	1996 draft technically acceptable - admin. Update
10.4.6	Condensate Cleanup System	1/6/2007	3/31/2007	administrative update based on 1981 version; with exception that update will replace reference to RG 1.56 with reference to EPRI BWRVIP-79, TR-103515, "BWR Water Chemistry Guidelines."
10.4.7	Condensate and Feedwater System	1/6/2007	3/31/2007	GDC 34/44 DID; 1996 draft technically acceptable - admin. Update
10.4.8	Steam Generator Blowdown System (PWR)	1/6/2007	3/31/2007	Administrative update based on 1981 version - no new staff positions

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10.4.9	Auxiliary Feedwater System (PWR)	1/6/2007	3/31/2007	RTNSS for AP1000; 1996 draft to be modified to: • Specify the reliability of the minimum recirculation flow path to address operating experience; • Address design interface with safety-related water source (i.e., AFW system design to accommodate water of lower quality); • Address timing and reliability of connection to backup safety-related water source.
11.1	Source Terms	11/9/2006	3/31/2007	1996 draft technically acceptable. Adminstrative update with minor changes which will not result in new staff positions.
11.2	Liquid Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also address the requirements of 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. The SRP will be updated administratively in accordance with LIC-200.
11.3	Gaseous Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also address the requirements of 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. The SRP will be updated administratively in accordance with LIC-200.
11.4	Solid Waste Management Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. The update will address the use of mobile waste treatment systems connected to permanent plant systems. The guidance will also be revised to clarify the performance criteria for ion exchange and charcoal adsorbent media. The revision will also addressed the requirements 10 CFR 20.1406. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. Also, the requirements from Chapter 16 Technical Specifications and RETS to those identified in Generic Letter 89-01, as implemented under the guidance of NUREG-1301 and NUREG-1302, will be updated. The SRP will be updated administratively in accordance with LIC-200.
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	1/6/2007	3/31/2007	Staff will revise the 1996 Draft. References to current Regulatory Guides (RG) and industry standards will be revised, as is applicable, as well as necessary conforming changes to the SRP. Also, the requirements from Chapter 16 Technical Specifications and RETS to those identified in Generic Letter 89-01, as implemented under the guidance of NUREG-1301 and NUREG-1302, will be updated. The SRP will be updated administratively in accordance with LIC-200.
12.1	Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	1/6/2007	3/31/2007	Revision will reflect several revisions to 10 CFR Part 20 from the 1981 version of the SRP, update references to RGs, NUREGS, and standards, and be administratively updated in accordance with LIC-200
12.2	Radiation Sources	1/6/2007	3/31/2007	Revision will reflect several revisions to 10 CFR Part 20 from the 1981 version of the SRP, update references to RGs, NUREGS, and standards, and be administratively updated in accordance with LIC-200
12.3 - 12.4	Radiation Protection Design Features	1/6/2007	3/31/2007	Revision will reflect several revisions to 10 CFR Part 20 from the 1981 version of the SRP including 10 CFR 20.1406, update references to RGs, NUREGS, and standards, and be administratively updated in accordance with LIC-200
12.5	Operational Radiation Protection Program	12/30/2005	3/31/2007	Draft Revision 3 was published December 2005 for comment and is available on the Web or in ADAMS: ML060170759
13.1.1	Management and Technical Support Organization	12/8/2006	3/31/2007	Administrative update
13.1.2 - 13.1.3	Operating Organization	8/31/2005	3/31/2007	Published 8/05; correct Pt 52 terminology late-stage/early stage"
3.2.1	Reactor Operator Training	11/30/2005	11/30/2005	Revision 2 was published November 2005 and is available on Web or in ADAMS: ML060030205; Previously issued for public comment 12/2002.
13.2.2	Training for Non Licensed Plant Staff	11/30/2005	11/30/2005	Revision 2 was published November 2005 and is available on Web or in ADAMS: ML060030199; Previously issued for public comment 12/2002.

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SRP Revision Schedule

[redacted]				
13.3	Emergency Planning	9/8/2006	3/31/2007	Revision will be issued for comment Sep 2006; ESP: Supp. 2 to NUREG-0654/FEMA-REP-1; DC: RG 1.101, NUREG-0696, NUREG-0737 (inc. Supp.1); COL RG 1.101; NUREG-0654/FEMA-REP-1, NUREG-0737 (inc. supp.1); COL Operational program SECY: Appendix E.IV.F.a: (1) full participation exercise within two years before issuance of first operating license for full power; and (2) onsite exercise within one year before issuance of operating license for full power. Appendix E.V: detailed implementing procedures submitted within 180 days prior to fuel load.
[redacted]				
13.5.1.1	Administrative Procedures - General	12/8/2006	3/31/2007	Administrative update
[redacted]				
13.5.2.1	Operating and Emergency Operating Procedures	11/30/2005	11/30/2005	Revision 1 was published November 2005 and is available on Web or in ADAMS: ML060030233; Previously issued for public comment 12/2002.
[redacted]				
13.6	Physical Security	12/8/2006	3/1/2007	13.6 is being revised in total to be aligned to format and content of NRC endorsed NEI 03-12, "Template For The Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program]," (Revision 1 — March 2004) see ML033640038, as well as incorporate an updated version of the acceptance criteria as previously issued in NUREG-0908, Acceptance Criteria for the Evaluation of the Nuclear Power Reactor Security Plans, dated August 1982. However, in light of anticipated/proposed security rulemakings, certain acceptance criteria will be revised accordingly consistent with the respective rulemaking schedules.
14.2	Initial Plant Test Program	11/15/2006	3/1/2007	The prioritization/schedule of this update is consistent with the COL applicants' needs for developing their initial test program. The update of RG 1.68 will include test requirements of passive systems.
[redacted]				
14.3	Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification	12/22/2006	3/1/2007	Need techncial update by 4/2006 and template for balance of 14.3 sections prior to individual updates
14.3.1	Site Parameters (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3
14.3.2	Structural and Systems Engineering (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3
14.3.3	Piping Systems and Components (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3.
14.3.4	Reactor Systems (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3.
14.3.5	Instrumentation and Controls (Tier 1)	12/22/2006	3/1/2007	Guidance on ITAAC for I&C will be updated per LIC-200, updates to referenced regulatory guides and standards, and necessary conforming changes resulting from SRP Chapter 7 updates

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SRP Revision Schedule

[redacted]				
14.3.6	Electrical Systems (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3
14.3.7	Plant Systems (Tier 1)	12/22/2006	3/1/2007	1996 draft technically acceptable - admin. Update
14.3.8	Radiation Protection and Emergency Preparedness (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3.
14.3.9	Human Factors Engineering (Tier 1)	12/22/2006	3/1/2007	will be coordinated 14.3.
14.3.10	Initial Test Program and D-RAP (Tier 1)	12/22/2006	3/1/2007	Will be developed on same schedule as Section 14.2.
14.3.11	Containment Systems and Severe Accidents (Tier 1)	12/22/2006	3/1/2007	1996 draft technically acceptable - admin. Update
15.0	Accident Analysis - Introduction	1/5/2007	3/31/2007	
[redacted]				
15.0.2	Review of Transient and Accident Analysis Methods	12/1/2005	12/1/2005	Issued December 2005 with Regulatory Guide 1.203, "Transient and Accident Analysis Methods"
15.0.3(new)	Radiological Consequences of Design Basis Accidents - for ESP, DC, and COL applications	12/1/2006	3/1/2007	This is a new section that will address Part 52 licensing, it will incorporate by reference RG 1.183, it will subsume Att 2, Section 15 of RS-0002, and be informed by the ESBWR/AP1000 Design Certification reviews. The schedule for updating RG 1.183 is independent of development of this section
15.1.1 - 15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1/5/2007	3/1/2007	Work in progress.
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	1/5/2007	3/1/2007	Work in progress.
[redacted]				
15.2.1 - 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	1/5/2007	3/1/2007	Work in progress.
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	1/5/2007	3/1/2007	Work in progress.
15.2.7	Loss of Normal Feedwater Flow	1/5/2007	3/1/2007	Work in progress.
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	1/5/2007	3/1/2007	Work in progress.
15.3.1 - 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	1/5/2007	3/1/2007	Work in progress.

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SRP Revision Schedule

[redacted]				
15.3.3 - 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	1/5/2007	3/1/2007	Work in progress.
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	1/5/2007	3/1/2007	Work in progress.
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	1/5/2007	3/1/2007	Work in progress.
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	1/5/2007	3/1/2007	Work in progress.
15.4.4 - 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1/5/2007	3/1/2007	Work in progress.
15.4.6	Chemical and Volume Control System Malfunction that Results in Decrease in Boron Concentration in the Reactor Coolant (PWR)	1/5/2007	3/1/2007	Work in progress.
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1/5/2007	3/1/2007	Work in progress.
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	1/5/2007	3/1/2007	Work in progress.
[redacted]				
15.4.9	Spectrum of Rod Drop Accidents (BWR)	1/5/2007	3/1/2007	Work in progress.
[redacted]				
15.5.1 - 15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	1/5/2007	3/1/2007	Work in progress.
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	1/5/2007	3/1/2007	Work in progress.
[redacted]				
15.6.5	Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	1/5/2007	3/1/2007	Update to be coordinated with section 6.3
[redacted]				

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SRP Revision Schedule

[redacted]				
15.8	Anticipated Transients Without Scram	1/5/2007	3/1/2007	Work in progress.
15.9 (new)	BWR Core Stability	1/5/2007	3/1/2007	Work in progress.
[redacted]				
16.0	Technical Specifications	12/8/2006	3/1/2007	SRP revision will be administrative in nature
16.1	Risk-Informed Decision Making: Technical Specifications	12/8/2006	3/1/2007	SRP revision will be administrative in nature
[redacted]				
17.4	Reliability Assurance Program	10/27/2006	3/1/2007	Update of initial 1996 draft based on Secy-95-0132
17.5	Quality Assurance new section	9/30/2006	3/1/2007	Issued as draft and is available on the web or in ADAMS - ML060180622, public comment period end date of April 11, 2006

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17.6	Maintenance Rule	12/22/2006	3/1/2007	Work in progress.
18.0	Human Factors Engineering Introduction	2/15/2007	3/1/2007	Revision 1 issued 2/2004, changes will be administrative per LIC-200
19.0	Probabilistic Risk Assessment	12/22/2006	3/1/2007	Revision to Chapter 19 will address staff review of COL plant specific PRA per proposed 10CFR 52.80, severe accidents per proposed 10CFR 52.79(a)(17) and 10CFR 79(a)(38) and will be based on application guidance contained in DG-1145. 19.0 will include guidance on severe guidance so there will not be a seperate 19.2. Also 19.1 will be referenced by 19.0 but will be updated persuant to RG 1.200 effort and schedule. Guidance on severe accidents is contained in Commission Policy
[redacted]				

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September 1, 2006

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards (ACRS)
Advisory Committee on Nuclear Waste

FROM: David B. Matthews, Director
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: TRANSMITTAL OF DRAFT REGULATORY GUIDE DG-1145 "COMBINED LICENSE APPLICATIONS FOR NUCLEAR POWER PLANTS (LWR EDITION)"

The purpose of this memorandum is to transmit draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition) [Enclosure 1]," to the ACRS in support of its upcoming review. This draft Regulatory Guide was made publicly available on September 1, 2006 on the NRC website and the 45 day public comment period will officially begin on September 6, 2006, upon posting in the Federal Register.

A presentation of DG-1145 to the ACRS has been previously scheduled for the December 2006 meeting so that DG-1145 technical content, public comments, and public comment resolution can be summarized and discussed. The current transmittal is provided to allow initial ACRS review in order to identify technical topics that could be discussed in more detail prior to the December meeting. Staff is available either during or after the public comment period to support in-depth technical discussions contained in DG-1145 on selected topics.

The purpose of DG-1145 is to provide guidance regarding the information to be submitted in a combined license (COL) application for a nuclear power plant. As such, this guide is intended to address many, albeit not all, of the application options allowed by Title 10, Part 52, of the Code of Federal Regulations (10 CFR Part 52). Although a COL applicant is not required to conform to this guidance, its use will facilitate both the applicant's preparation of a COL application and timely review of the application by the staff of the U.S. Nuclear Regulatory Commission (NRC).

Please contact Robert Tregoning (301-415-6657) with any questions concerning DG-1145 and to schedule additional ACRS meetings on DG-1145.

Enclosures: 1. Draft Regulatory Guide DG-1145 "Combined License Applications for Nuclear Power Plants (LWR Edition)"

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ACRS REVIEW OF COL REGULATORY GUIDE (DG-1145)

Chap.	Title	ACRS Member
1	Introduction and General Description of Plant	T. Kress
2	Sites Characteristics	D. Powers
3	Design of Structures, Components, Equipment, and Systems	W. Shack
4	Reactor	S. Amijo
5	Reactor Coolant System and Connected Systems	J. Sieber
6	Engineered Safety Features	M. Corradini
7	Instrumentation and Controls	S. Abdel-Khalik
8	Electric Power	J. Sieber
9	Auxiliary Systems	O. Maynard
10	Steam and Power Conversion System	S. Abdel-Khalik
11	Radioactive Waste Management	D. Powers
12	Radiation Protection	D. Powers
13	Conduct of Operations	O. Maynard
14	Initial Test Program and ITAAC-Design Certification	T. Kress
15	Accident Analysis	S. Bannerjee
16	Technical Specifications	O. Maynard
17	Quality Assurance	O. Maynard
18	Human Factors Engineering	M. Bonaca
19.1	PRA	G. Apostolakis
19.2	Severe Accidents	M. Corradini
19A	Seismic Margins Analysis	D. Powers
20	Generic Issues	T. Kress
21	Testing and Computer Code Validation	S. Bannerjee
22	Regulatory Treatment of Non-Safety Related Equipment	G. Apostolakis

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

DRAFT

MEMORANDUM TO: ACRS Members/ACRS Staff

FROM: Sam Duraiswamy, Technical Assistant

SUBJECT: PROPOSED REVISION TO ACRS SUBCOMMITTEE STRUCTURE

A proposed revision to the ACRS Subcommittee Structure is provided below. This involves combining certain existing Subcommittees, creation of new Subcommittees to deal with COL applications, and member assignments. The revised subcommittee structure was sent to the members and ACRS staff for review and comment. Comments received were incorporated as appropriate. After approval by the Committee at the September meeting, the revised subcommittee structure will become effective on September 15, 2006.

Combined Existing Subcommittees

- The Reactor Fuels Subcommittee has been combined with the Materials and Metallurgy Subcommittee and is renamed as Materials, Metallurgy, and Reactor Fuels. Dr. Armijo will chair this Subcommittee and the current Chairman Dr. Shack will remain as a member. Dr. Powers, the current Chairman of the Reactor Fuels Subcommittee, will remain as a member and will handle specialized assignments (e.g., operating license application for the MOX Fuel Fabrication Facility)

- The Fire Protection Subcommittee has been combined with the Plant Operations Subcommittee and is renamed as Plant Operations and Fire Protection. Selected Tasks (e.g., Fire PRA Models and Verification/validation of selected fire models) will be assigned to the Reliability and PRA Subcommittee. Mr. Sieber, the current Chairman of the Plant Operations Subcommittee, will chair this combined Subcommittee.

- The Human Factors Subcommittee has been combined with the Reliability and Probabilistic Risk Assessment Subcommittee. Dr. Apostolakis will be the Chairman of this combined Subcommittee. Dr. Bonaca, current Chairman of the Human Factors Subcommittee, will remain as a member.

Chairmanship Assignments

- Plant License Renewal Subcommittee will be chaired by Dr. Bonaca, Mr. Sieber, or Mr. Maynard, as assigned, to review specific license renewal applications.

- Dr. Bonaca will become the Chairman of the Power Uprates Subcommittee. Dr. Powers, Dr. Banerjee, and Dr. Abdel-Khalik will assist in reviewing specific power uprate applications, as needed.

- Dr. Banerjee will become the Chairman of the Thermal-Hydraulic Phenomena Subcommittee. The current Chairman, Dr. Wallis, will remain as a member.

New Subcommittees

- AP1000 Combined License Application will be chaired by Dr. Bonaca
- ESBWR Combined License Application will be chaired by Mr. Maynard
- EPR Combined License Application will be chaired by Dr. Powers
- ABWR Combined License Application will be chaired by Dr. Abdel-Khalik

Design Certification Applications

Future Plant Designs Subcommittee is responsible for reviewing the applications for certification of the ESBWR and EPR designs as well as the Framework document and other generic matters associated with future plant designs. Dr. Kress, current Chairman of this Subcommittee, is responsible for reviewing the Framework document and other generic matters. Dr. Corradini is responsible for reviewing the design certification applications.

ACRS SUBCOMMITTEE STRUCTURE
MEMBER ASSIGNMENTS

Subcommittees	GEA	JSA	SAK	MVB	SB	MC	TSK	OLM	DAP	JDS	WJS	GBW
AP1000 COL Application		x		**X**	x	x	x					
ABWR COL Application			**X**		x			x		x	x	
Digital I&C Systems	**X**		x	x					x	x		
Early Site Permits							x	x	**X**		x	
EPR COL Application	x	x			x	x			**X**	x		
Future Plant Designs	x	x	x		x	x	**X**				x	x
ESBWR COL Application		x		x	x	x		**X**	x			
Materials, Metallurgy, & Reactor Fuels		**X**		x					x	x	x	
Planning & Procedures										x	x	**X**
Plant License Renewal				**X**				x		x	x	x
Plant Operations and Fire Protection	x			x	x			x		**X**		
Power Uprates		x	x	**X**	x		x	x	x			x
Reg Policies & Practices		x	x			x	x				**X**	
Reliability and PRA	**X**		x	x		x	x	x			x	
Safety Research Program	x	x	x	x	x	x	x				x	x
Safeguards & Security	x			**X**			x	x	x	x		
T-H Phenomena			x		**X**	x			x	x		x
Joint ACRS/ACNW Subc.							x		x			
Total	7	8	8	10	9	8	9	8	9	9	9	6

Chairman - [bold] **X**
Member - x

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Cognizant ACRS Staff:

RC - Ralph Caruso
SD - Sam Duraiswamy
DCF - David C. Fischer
JHF - John H. Flack
JTL - John T. Larkins
HPN - Hossein P. Nourbakhsh
CS - Cayetano (Tanny) Santos
MAJ - Michael A. Junge
MRS - Michael R. Snodderly
EAT - Eric A. Thornsbury

ACRS Members:

SAK - Said Abdel-Khalik
GEA - George E. Apostolakis
JSA - Joseph Sam Armijo
SB - Sanjoy Banerjee
MVB - Mario V. Bonaca
MC - Michael Corradini
TSK - Thomas S. Kress
OLM - Otto L. Maynard
DAP - Dana A. Powers
MTR - Michael T. Ryan, ACNW
WJS - William J. Shack
JDS - John D. Sieber
GBW - Graham B. Wallis

TOPICAL SUBCOMMITTEES

AP1000 COMBINED LICENSE APPLICATION (DCF) .... **BONACA**, Armijo, Banerjee, Corradini, Kress

- Review combined license applications associated with the AP1000 design.
- Review , as needed, Westinghouse topical reports referenced in the COL application.
- Review associated Design Acceptance Criteria (DAC) and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).
- Review plant-specific PRAs and/or updates in coordination with the Reliability and PRA Subcommittee.
- Review resolution of site specific issues identified in the early site permit.
- Review materials, metallurgical, and reactor fuel issues in coordination with the Materials, Metallurgy, and Reactor Fuels Subcommittee.

ABWR COMBINED LICENSE APPLICATION (TBD) .... **ABDEL-KHALIK**, Banerjee, Corradini Maynard, Shack, Sieber

- Review combined license applications associated with the ABWR design.
- Review, as needed, GE topical reports referenced in the COL application.
- Review, operating experience associated with the ABWR plant.
- Review associated Design Acceptance Criteria (DAC) and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).
- Review plant-specific PRAs and/or updates in coordination with the Reliability and PRA Subcommittee.
- Review resolution of site specific issues identified in the early site permit.
- Review materials, metallurgical, and reactor fuel issues in coordination with the Materials, Metallurgy, and Reactor Fuels Subcommittee.

DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS (EAT) .... **APOSTOLAKIS**, Abdel-Khalik, Bonaca, Powers, Sieber

- Review digital instrumentation and control systems research activities and identify any new research needs.
- Review NRC staff and industry activities associated with digital instrumentation and control systems for operating and future plants.

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- Review regulatory guidance associated with digital instrumentation and control systems
- Review the use of formal methods to improve requirements for digital system requirements and quality.
- Review methods for evaluating digital systems reliability as part of PRA.

EARLY SITE PERMITS (DCF) .... **POWERS**, Kress, Maynard, Shack

- Review early site permit applications
- Review seismic requirements associated with early site permit applications
- Monitor the effectiveness of the early site permit review standard and propose changes, as needed, based on the lessons learned from reviewing early site permit applications.
- Prepare a lessons learned report subsequent to completing the review of the initial applications.

EPR COMBINED LICENSE APPLICATION (HPN) .... **POWERS**, Banerjee, Corradini, Sieber, Shack

- Review combined license applications associated with the EPR design.
- Review, as needed, Framatome, AMP, INC. topical reports referenced in the COL application.
- Review associated Design Acceptance Criteria (DAC) and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).
- Review plant-specific PRAs and/or updates in coordination with the Reliability and PRA Subcommittee.
- Review resolution of site specific issues identified in the early site permit.
- Review materials, metallurgical, and reactor fuel issues in coordination with the Materials, Metallurgy, and Reactor Fuels Subcommittee.

ESBWR COMBINED LICENSE APPLICATION (TBD) .... **MAYNARD**, Armijo, Banerjee, Bonaca, Corradini, Powers

- Review combined license applications associated with the ESBWR design.
- Review, as needed, GE topical reports referenced in the COL application.

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- Review associated Design Acceptance Criteria (DAC) and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).
- Review plant-specific PRAs and/or updates in coordination with the Reliability and PRA Subcommittee.
- Review resolution of site specific issues identified in the early site permit.
- Review materials, metallurgical, and reactor fuel issues in coordination with the Materials, Metallurgy, and Reactor Fuels Subcommittee.

<u>FUTURE PLANT DESIGNS</u> (DCF) .... **<u>KRESS</u>**, Apostolakis, Abdel-Khalik, Armijo, Banerjee, Corradini, Powers, Shack, Wallis

- Review the technology - neutral framework for future plant licensing.
- Review regulatory challenges associated with advanced reactor designs.
- Review draft Regulatory Guide, DG-1145, COL Application Guidance.
- Identify the research needs for developing an infrastructure for review of future plant designs.
- Perform pre-application and design certification reviews of future plant designs (ESBWR, US EPR, US APWR, etc.).
- Review draft final 10 CFR Part 52 construction inspection program framework document in coordination with the Subcommittee on Plant Operations and Fire Protection.

<u>MATERIALS, METALLURGY, AND REACTOR FUELS</u> (CS) .... **<u>ARMIJO</u>**, Bonaca, Powers, Shack, Sieber

- Review proactive materials degradation assessment program.
- Review NRC program to evaluate plant aging of metal components (e.g., pressure vessel embrittlement, steam generator tube degradation, and thermal aging of cast stainless steel piping and components).
- Review the adequacy of nondestructive examination techniques in detecting and sizing flaws in metal components, piping systems, and steam generator tubes.

- Review regulatory approach associated with the steam generator tube integrity, and the staff's safety evaluation on industry proposed technical specifications for addressing steam generator tube integrity.
- Review flow-accelerated corrosion issues.
- Review industry and NRC activities associated with primary water stress corrosion cracking issue.
- Review NRC staff's resolution of Steam Generator Action Plan items, including those issues raised by the ACRS in NUREG-1740 associated with the Differing Professional Opinion (DPO) on steam generator tube integrity.
- Review proposed revisions to 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS), in coordination with the Subcommittees on Thermal-Hydraulic Phenomena and on Reliability and Probabilistic Risk Assessment.
- Review proposed amendment to 10 CFR 50.55a regarding volumetric examination of the small- bore piping of the high pressure safety injection system.
- Review issues associated with license renewal for Independent Spent Fuel Storage Installations and the structural integrity of fuel shipping casks and adequacy of spent fuel encapsulators, as requested by the Advisory Committee on Nuclear Waste (ACNW).
- Assume lead responsibility for activities related to reactor fuel and review fuel-related issues.
- Review NRC staff activities related to revising fuel design acceptance criteria for high-burnup fuel.
- Review programs of industry, Fuel Vendors, and Owners Groups to address concerns associated with use of high-burnup fuel.
- Consider fuel performance during normal and abnormal conditions, including fuel failure propagation.
- Review NRC and Industry fuel performance codes.
- Review reactor neutronics analytical methods.
- Consider the nature and characteristics of core-coolant interactions (e.g., steam explosions) and core-concrete interactions.
- Review DOE Tritium Production Program using commercial nuclear power plants, including the license amendment for batch loading of assemblies in Catawba and McGuire.

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- Review the licensing of uranium enrichment facilities.
- Review research activities associated with reactor fuels.
- Evaluate the design of spent fuel storage pools, including pool storage capability and provisions to preclude criticality and to cool the fuel under normal and abnormal conditions and following external events such as earthquakes.
- Review spent fuel pool accident risk for decommissioning and operating plants.
- Review safety issues associated with new and modified fuel designs.
- Review the operating license application for the MOX Fuel Fabrication Facility and the associated NRC staff's SER.

PLANNING AND PROCEDURES (SD) .... **WALLIS**, Shack, Sieber

(NOTE: This Subcommittee includes the ACRS Chairman, Vice Chairman, and member-at-large elected each year at the same time as the Chairman and Vice Chairman, and is chaired by the ACRS Chairman.)

- Prioritize items proposed for each ACRS Full Committee meeting.
- Organize ACRS retreats and identify proposed topics for discussion.
- Follow-up on the resolution and implementation of the commitments made at the ACRS retreats.
- Develop proposals for changes in ACRS policies, practices, and bylaws for consideration by the Full Committee. Consider especially changes mandated by revisions to the Federal Advisory Committee Act.
- Implement policies of ACRS in planning Full Committee activities, articulating priorities, and scheduling and monitoring activities of the ACRS Subcommittees.
- Perform annual review of the Subcommittee structure, tasks, and workload of members and recommend changes, as needed, for Full Committee consideration.
- Assume lead responsibility for the coordination of ACRS meeting with international organizations.
- Monitor the adequacy of implementation of the memorandum of understanding (MOU) between the ACRS and the EDO.

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PLANT LICENSE RENEWAL (CS/MAJ) .... **BONACA**, Maynard, Shack, Sieber, Wallis

- Review license renewal applications and associated NRC staff's Safety Evaluation Reports.

- Consider the NRC program to evaluate plant aging except for those aspects being considered by the Materials and Metallurgy Subcommittee (e.g., pressure vessel embrittlement, stream generator tube degradation, and thermal aging of cast stainless steel piping and components).

- Review selected industry topical reports associated with license renewal.

- Review interim staff guidance dealing with license renewal issues.

- Review results of RES study to support NRR decisionmaking on the need for establishing limits for phosphate ion concentration in groundwater at the site of plants applying for license renewal (Dr. Powers has the lead).

- Consider potential improvements to the license renewal process.

- Review updates to the Standard Review Plan, Generic Aging Lessons Learned (GALL) Report, and Regulatory Guidance associated with license renewal to reflect lessons learned from the review of license renewal applications.

PLANT OPERATIONS AND FIRE PROTECTION (MAJ) ... **SIEBER**, Apostolakis, Banerjee, Bonaca, Maynard

- Assume lead responsibility for reviewing activities related to operating plants.

- Review significant operating events and provide periodic briefing to the Full Committee. Identify those events that should be discussed with the staff at the Full Committee meetings.

- Review restart of plants that have been shut down for an extended period (more than one year) and make recommendation to the Full Committee.

- Review, in coordination with the Reliability and PRA Subcommittee, initiatives related to risk-informed Technical Specifications.

- Review issues associated with the operation and maintenance of fuel cycle facilities and the adequacy of associated regulatory requirements.

- Take lead responsibility in coordinating annual meeting with different NRC Regional Offices and ACRS members' visit to a plant to obtain information on Regional activities and industry issues.

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- Review the Mitigating Systems Performance Index (MSPI) Program in coordination with the Reliability and PRA Subcommittee.
- Review NRC staff and industry activities associated with grid reliability.
- Review enhancement to the Significance Determination Process in coordination with the Reliability and PRA Subcommittee.
- Consider generic safety implications of the performance of systems not assigned to other Subcommittees (e.g., air powered systems, cleanup systems, and chilled water systems).
- Consider biological effects of ionizing radiation, standards for protection against radiation (10 CFR Part 20), and associated regulatory guidance.
- Review mechanical component operability assurance and reliability, including the functioning of valves under accident loading conditions.
- Review systems interaction issues and criteria, including consideration of functional interactions for existing and future plants.
- Consider the effects of harsh and adverse environment on the plant safety systems.
- Review reliability of AC/DC power systems in nuclear facilities, including the potential for disruption of offsite power sources and backup power systems.
- Review lightning protection provisions for future plants.
- Provide oversight and coordination of the prioritization and resolution of generic safety issues, handling those items it is competent to deal with and assigning others to appropriate Subcommittees for review.
- Review adequacy of fire protection requirements for operating plants.
- Review the Regulatory Guide to endorse NEI implementing guidance document for the revised 10 CFR 50.48.
- Review the Boiling Water Reactor Owners Group (BWROG)/Nuclear Energy Institute (NEI) post-fire safe shutdown circuit analysis and associated NRC staff's evaluation.
- Review the Significance Determination Process for findings of inspections dealing with fire protection
- Review fire protection aspects of the advanced reactor designs in coordination with the Future Plant Designs Subcommittee.
- Review the fire protection research program.

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POWER UPRATES (RC) .... **Bonaca**, Abdel-Khalik, Armijo, Banerjee, Kress, Powers, Sieber, Wallis

- Review extended power uprate applications.
- Review staff guidance documents, technical assessments, and topical reports associated with power uprate applications.
- Review potential synergistic effects and margin reduction associated with the concurrent regulatory activities (e.g., power uprates, license renewal, and risk-informed regulation)
- Review issues associated with core reload analysis for plants seeking power uprates.

REGULATORY POLICIES AND PRACTICES (EAT) .... **SHACK,** Abdel-Khalik, Armijo, Corradini, Kress, Maynard

- Review the rulemaking package for risk-informing 10 CFR 50.46.
- Review regulatory guidance in support of risk-informed revision to 10 CFR 50.46.
- Examine the coherence and specific aspects of the NRC regulatory process, as appropriate, and consider changes in emphasis needed in safety-related NRC rules and regulatory practices.
- Identify important safety issues needing increased (or less) attention and/or resolution in the NRC regulatory process.
- Review proposed NRC safety-related rules not assigned to specific ACRS Subcommittees.
- Review the NRC staff's reevaluation of the effectiveness of those existing regulations which were not assigned to other Subcommittees.
- Consider activities associated with the NRC oversight of DOE facilities.
- Consider the use of defense-in-depth concept in the regulatory process.
- Review NRC research and information needs in the seismic area in coordination with the Reliability and PRA Subcommittee.
- Review NRC/industry seismic design margins evaluation program in coordination with the Reliability and PRA Subcommittee.

(8)

RELIABILITY AND PROBABILISTIC RISK ASSESSMENT (EAT) .... **APOSTOLAKIS**, Abdel-Khalik, Bonaca, Corradini, Kress, Maynard, Shack

- Review the risk-informed and performance-based regulatory approaches, including the NUREG document on estimating the LOCA Frequencies through the elicitation process.

- Review the staff's plan for achieving coherence among risk-informed regulatory activities within the reactor safety arena.

- Consider application of risk insights in the regulatory process.

- Review updates to risk-informed regulation implementation plan.

- Consider the consistent and extended use of PRAs in the regulatory process and the associated NRC programs.

- Gather information for developing recommendations to the Commission on the significance of low-power and shutdown operations risk and review the adequacy of the staff's analytical tools for independently assessing the risk significance of plant configurations during low-power and shutdown operations, especially during plant transitions.

- Review Appendix C to Regulatory Guide 1.200, endorsing ANS External Events Standard.

- Review Appendix D to Regulatory Guide 1.200, endorsing ANS Standard on Low-Power and Shutdown Operating PRA.

- Review Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components for Nuclear Power Reactors."

- Review draft final Regulatory Guide 1.200 to incorporate lessons learned from trial use period.

- Review risk-based performance indicators and the Significance Determination Process in coordination with the Plant Operations Subcommittee.

- Review the initiatives related to risk-informed Technical Specifications in coordination with the Plant Operations Subcommittee.

- Review guidance on performance of sensitivity and uncertainty analyses of PRA results for risk-informed activities.

- Review risk-based analysis of reactor operating experience.

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- Review the bases for the assumptions used in Probabilistic Seismic Hazard Analysis and its use in nuclear plant regulation.

- Review impact of common-made failures on the performance of plant safety systems.

- Review the ASP program and the development of SPAR models.

- Review verification and validation of selected fire models.

- Consider man-machine interaction, including design and arrangement of the control room and operator response under stress.

- Review control room habitability issues, associated regulatory guidance.

- Review Human Performance and Human Reliability Research activities.

- Review regulatory requirements for dealing with human factors issues.

- Review human/organizational factors issues associated with significant operating events in coordination with the Plant Operations Subcommittee.

- Review qualifications and training of personnel at nuclear facilities in coordination with the Plant Operations Subcommittee.

- Review the effects of power uprates on time available for manual operations during plant upset conditions.

- Monitor the NRC staff and industry activities in dealing with the Safety culture issue and gather information for use by the ACRS in reviewing this matter, as needed.

<u>SAFETY RESEARCH PROGRAM</u> (SD/HPN) .... **<u>POWERS</u>**, Apostolakis, Armijo, Banerjee, Bonaca, Corradini, Kress, Shack, Wallis

- Take lead responsibility in preparing biennial report to the Commission on the overall NRC Safety Research Program.

- Identify new areas of research that are essential for regulatory decisionmaking and research projects that are no longer cost effective and can be eliminated.

- Review the adequacy of the user office needs for research.

- Evaluate whether NRC research places proper emphasis on resolving important regulatory issues.

- Consider what research should be done by the NRC and the industry and cooperative research arrangements between NRC and other organizations.

- Take lead responsibility for establishing schedule and assigning members for assessing the quality of selected NRC research projects and preparing a proposed report documenting the results of the assessment.

SAFEGUARDS AND SECURITY (EAT) .... **BONACA**, Apostolakis, Maynard, Powers, Sieber

- Keep Informed of the NRC post-911 activities in the area of safeguards and security and be prepared to advise the Commission and the staff as requested.
- Review RES-sponsored work related to evaluations of nuclear facilities.
- Review technical issues associated with the Phase 1 and Phase 2 pilot plant studies.
- Review technical and risk-management issues associated with the evaluation of nuclear facilities, including those related spent fuel pools and decommissioning plants.
- Review technical issues associated with the reevaluation of emergency planning.
- Review proposed design features to mitigate the effects of sabotage, and plant arrangements to enhance security.
- Review NRC staff's rulemaking activities associated with safeguards and security

THERMAL-HYDRAULIC PHENOMENA (RC) .... **BANERJEE**, Abdel-Khalki, Corradini, Kress, Sieber, Wallis

- Consider evaluation of emerging safety issues associated with thermal-hydraulic phenomena.
- Review thermal-hydraulic issues associated with the development of revised PTS screening criterion.
- Consider NRC staff activities associated with the TRACE Code.
- Review issues related to water hammer and corrective measures.
- Review RES thermal-hydraulic research program, including experimental programs and the program to upgrade the NRC thermal-hydraulic codes.
- Review best estimate thermal-hydraulic codes submitted by licensees.
- Review issues associated with the use of industry-developed thermal-hydraulic codes.
- Review the thermal-hydraulic aspects of the future plant designs.
- Review proposed resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," and related NRC and industry guidance.

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ACRS/ACNW JOINT SUBCOMMITTEE (MRS) .... **KRESS/RYAN**, Powers

- Review proposed framework for risk-informing NMSS regulations.
- Review the application of the defense-in-depth concept in a risk-informed regulatory system.
- Review PRA for dry cask storage.
- Review risk-informed case studies being developed by NMSS.
- Other tasks to be determined jointly by the ACRS and ACNW.
- Review reports by ICRP and NCRP on radiation effects.

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From: "The Specters" <mhspecter@ns.sympatico.ca>
To: "Kress, Dr. Thomas S." <tskress@aol.com>
Date: 08/20/2006 11:04:01 AM
Subject: Emergency Planning at Indian Point

Dear Tom,

There has been a large effort to modernize the emergency plan at Indian Point, the nation's most populated, but not most popular, nuclear site. Indian Point has always been a "hot bed " of controversy, especially in the area of emergency planning. There are 305,000 permanent residents in the Emergency Planning Zone and the road system is congested. American Airlines Flight 11 flew down the Hudson Valley near Indian Point on its way to crashing into the World Trade Center. Documents seized from Al Qaeda have shown that nuclear plants are on the list of targets they are looking at. After the tragedy of the hurricane Katrina response, things have gotten into an uproar again. Several Congressmen and Senator Clinton have demanded an explanation as to why FEMA and the NRC have approved the emergency plan at Indian Point. To my knowledge no response has been provided to Congress yet.

For about two years now, as a consultant to Entergy the owner of the Indian Point plants, I have headed up the technical effort to modernize the Emergency Plan at Indian Point. This phase of the effort is nearing completion and Entergy and its supporting team would like to present our analyses to the ACRS sometime after Thanksgiving this fall.

There has been a great deal of progress. We have assumed a successful terrorist attack as our starting point, even though this is highly unlikely. We have assumed that a huge hole in the reinforced concrete containment was made by the terrorists who then went on to cause a reactor meltdown. We have calculated source terms for such scenarios and then used this information to calculate offsite consequences for a wide variety of health effects such as early fatalities, early injuries (of numerous kinds) and long term latent cancer effects. Detailed traffic analyses were made to determine the speeds and locations of people as they would evacuate, using the actual road network in the Indian Point area. The results of these traffic studies were then used as input to consequence analyses using the MACCS2 computer code. This consequence code, the same one the national labs and the NRC staff uses, was run in a sophisticated way to account for the time and location dependent movement of people away from the site. We also accounted for changing wind directions, a frequent occurrence at Indian Point.

Many important observations have come out of this study. We would suggest an improved keyhole shape and different protective actions within this new keyhole, we have calculated the importance of ad hoc measures such as breathing through a wet handkerchief...or using a face mask... that effectively eliminate respiratory damage, we have used the traffic analysis as a search engine and located specific roads in the area that might be made just one way during an emergency. We then went on to calculate the improved traffic flow and then determined how such traffic improvements would reduce offsite health effects. We are in the process of reviewing the benefits of sounding the General Emergency alarm sooner and the protocols in place to get emergency warnings out to the public, to see if they can be improved.

By combining advanced traffic analyses and advanced ways to calculate consequences we have created a new analytical tool. We can now quantify, in terms of health effects, the significance of any emergency action, such as the value of improving specific road traffic controls, an improved keyhole, earlier warnings, etc.

All of our results are being pulled together and we expect to compare them to a number of broader regulatory issues, such as safety goals, backfit analyses and possibly the new Reg. Guide 1.2. Most important, we are showing that even though this is the nation's most populated site and even though we have assumed that the terrorists were successful, the offsite health consequences are very small. This is important to Indian Point and perhaps even more important to all nuclear plants. It appears that the major effect of large releases of radioactive material is not health effects, but economic consequences.

I ask that you guide us in arranging such a presentation to the ACRS. I believe that Sandia and the NRC

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staff are also active in modernizing emergency planning and that they may be ready to present their results in a similar time frame. Other industry groups like EPRI and NEI may be able to add to the general review of emergency planning too, but I am not aware of their schedules.

I thank you for your help and look forward to your reply.

Best,
Herschel

CC: "Mario Fontana" <mhfontana1@comcast.net>

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JANUARY 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1 Holiday	2	3	4	5	6
7 PP3	8	9	10	11	12	13
14	15 Holiday	16	17	18	19	20
21 PP4	22	23	24	25 ACRS Retreat (tentative)	26 ACRS Retreat (tentative)	27
28	29	30	31			

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FEBRUARY 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4 PP5	5	6	7	8 ACRS 539 meeting	9 ACRS 539 meeting	10 ACRS 539 meeting
11	12	13 ACNW 175th meeting	14 ACNW 175th meeting	15 ACNW 175th meeting	16	17
18 PP6	19	20	21	22	23	24
25	26	27	28			

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MARCH 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4 PP7	5	6	7	8 ACRS 540th Meeting	9 ACRS 540th Meeting	10 ACRS 540th mtg
11	12	13	14	15	16	17
18 PP8	19	20 ACNW 176th Meeting	21 ACNW 176th Meeting	22 ACNW 176th Meeting	23	24
25	26	27	28	29	30	31

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APRIL 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1 PP9	2	3 Passover begins	4	5 ACRS 541st Meeting	6 ACRS 541st Meeting	7 ACRS 541st mtg
8	9	10 ACNW 177 meeting Passover Ends	11 ACNW 177 meeting	12 ACNW 177 meeting	13	14
15 PP10	16	17	18	19	20	21
22	23	24	25	26	27	28
29 PP11	30					

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MAY 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1	2	3 ACRS 542nd Meeting	4 ACRS 542nd Meeting	5 ACRS 542nd Mtg
6	7	8	9	10	11	12 ACNW, Paris, FR
13 PP12 ACNW, Paris, FR	14 ACNW, Paris, FR	15 ACNW, Paris, FR	16 ACNW, Paris, FR	17 ACNW, Paris, FR	18 ACNW, Paris, FR	19 ACNW, Paris, FR
20	21	22	23	24	25	26
27 PP13	28 Holiday	29	30	31		

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JUNE 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
					1	2
3	4	5	6 ACRS 543rd Meeting	7 ACRS 543rd Meeting	8 ACRS 543rd Meeting	9
10 PP14	11	12	13	14	15	16
17	18	19 ACNW 179th Meeting	20 ACNW 179th Meeting	21 ACNW 179th Meeting	22	23
24 PP15	25 ANS Annual Meeting, Boston, MA	26 ANS Annual Meeting, Boston, MA	27 ANS Annual Meeting, Boston, MA	28 ANS Annual Meeting, Boston, MA	29	30

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JULY 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1	2	3	4 Holiday	5	6	7
8 PP16	9 52nd Annual HP Meeting, Portland, OR	10 52nd Annual HP Meeting Portland, OR	11 ACRS 544th Meeting 52nd Annual HP Meeting Portland, OR	12 ACRS 544th Meeting 52nd Annual HP Meeting Portland, OR	13 ACRS 544th Meeting	14
15	16	17 ACNW 179th Meeting	18 ACNW 179th Meeting	19 ACNW 179th Meeting	20	21
22 PP17	23	24	25	26	27	28
29	30	31				

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AUGUST 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			1	2	3	4
5 PP18	6	7	8	9	10	11
12	13	14	15	16	17	18
19 PP19	20	21	22	23	24	25
26	27	28	29	30	31	

SEPTEMBER 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2 PP20	3 Holiday	4	5	6 ACRS 545th Meeting	7 ACRS 545th Meeting	8 ACRS 545th Mtg
9	10	11	12	13 Rosh Hashanah	14 Rosh Hashanah	15
16 PP21	17	18 ACNW 180st Meeting	19 ACNW 180st Meeting	20 ACNW 180st Meeting	21 Yom Kippur	22 Yom Kippur
23	24	25	26	27 Sukkot begins	28	29
30 PP22						

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OCTOBER 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1	2	3 Sukkot ends	4 ACRS 546th Meeting	5 ACRS 546th Meeting	6 ACRS 546th Mtg
7	8 Holiday	9	10	11	12	13
14 PP23	15	16 ACNW 181st Meeting	17 ACNW 181st Meeting	18 ACNW 181st Meeting	19	20
21	22	23	24	25	26	27
28 PP24	29	30	31			

(PC)

NOVEMBER 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			(10/31) ACRS 547th Meeting	1 ACRS 547th Meeting	2 ACRS 547th Meeting	3
4	5	6	7	8	9	10
11 PP25	12 Holiday	13 ACNW 182nd Meeting	14 ACNW 182nd Meeting	15 ACNW 182nd Meeting	16	17
18	19	20	21	22 Holiday	23	24
25 PP26	26	27	28	29	30	

DECEMBER 2007

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2	3	4	5 Chanukah begins	6 ACRS 548th Meeting	7 ACRS 548th Meeting	8 ACRS 548th Mtg.
9 PP1	10	11	12 Chanukah ends	13	14	15
16	17 ACNW 186th Meeting	18 ACNW 186th Meeting	19 ACNW 186th Meeting	20	21	22
23 PP2	24	25 Holiday	26	27	28	29
30	[illegible]					

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
535th	13	1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

List of Documents Attached

See attached list

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Instructions to Preparer	Lead Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Draft Final Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations (MAJ/JDS)	09/06/06 (p. 1)	07/14/06 (p. 2)	06/16/06 (pp. 3-5)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

September 6, 2006

MEMORANDUM TO: Jack Sieber, Chairman
Plant Operations Subcommittee

FROM: Michael A. Junge, Senior Staff Engineer
Advisory Committee on Reactor Safeguards Staff

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED JULY 14, 2006, CONCERNING THE DRAFT FINAL GENERIC LETTER 2006-XX, "POST-FIRE SAFE-SHUTDOWN CIRCUITS ANALYSIS SPURIOUS ACTUATIONS."

Attachment 1 contains a copy of the Executive Director for Operations (EDO) July 14, 2006 response to the Advisory Committee on Reactor Safeguards (ACRS) June 16, 2006, letter regarding draft final Generic Letter (GL) 2006-XX, "Post-Fire Safe-Shutdowns Circuits Analysis Spurious Actuations." Attachment 2 contains a copy of the Committee letter.

Recommendation

The staff should issue the proposed GL after clarifying the scope of requested information and adjusting submittal dates to be more realistic.

EDO Response

The staff has revised the GL to provide additional time for licensees to perform the expected analyses. Furthermore, the staff clearly defined the scope of the information requested in the revised letter.

Analysis

The EDO agrees with the ACRS recommendation.

RECOMMENDATION

The EDO response is satisfactory for the Recommendation.

cc: ACRS Members
J. Larkins
M. Snodderly
S. Duraiswamy

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 14, 2006

Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: DRAFT FINAL GENERIC LETTER 2006-XX, "POST-FIRE SAFE-SHUTDOWN CIRCUITS ANALYSIS SPURIOUS ACTUATIONS"

Dear Dr. Wallis:

I am responding to your June 16, 2006, letter on the draft final generic letter (GL) titled, "Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations." The Advisory Committee on Reactor Safeguards (ACRS or the Committee) recommended that the staff issue the proposed GL after clarifying the scope of requested information and adjusting the submittal dates to be more realistic. Specifically, the ACRS noted that it is unreasonable to expect the licensees to perform the requested analyses of multiple spurious actuations within 90 days, as it would be necessary to assess the functionality of systems, structures, and components and to identify appropriate compensatory measures.

The staff has revised the GL to provide additional time for licensees to perform the expected analyses. Furthermore, the staff clearly defined the scope of information requested in the revised letter by requesting licensees to (a) submit their conclusion regarding compliance within 90 days, and (b) submit their corrective action plans, if applicable, within 6 months.

The staff believes that the revised timeframe is reasonable because the results of the Electric Power Research Institute and Nuclear Energy Institute cable fire tests became available to the licensees in 2001 and the NRC has communicated the staff's expectations through a series of public meetings since 2003.

We appreciate the time and effort the Committee devoted to this subject and will schedule a briefing with the Committee after we have reviewed the licensee's responses. We will continue to work closely with the ACRS on future fire protection issues.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 16, 2006

Mr. Luis A. Reyes
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL GENERIC LETTER 2006-XX: POST-FIRE SAFE-SHUTDOWN CIRCUIT ANALYSIS SPURIOUS ACTUATIONS

Dear Mr. Reyes:

During the 533rd meeting of the Advisory Committee on Reactor Safeguards, May 31-June 1, 2006, we reviewed the Draft Final Generic Letter (GL) 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations. During our review, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), Duke Energy, and Progress Energy. We also had the benefit of the documents referenced.

RECOMMENDATION

The Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations should be issued after the scope of requested information is clarified and the submittal dates are made more realistic.

BACKGROUND

One of the consequences of the Browns Ferry fire in 1975 was a number of spurious actuations of equipment. The proper treatment of spurious actuations that could affect the ability of a nuclear power plant to safely shut down during a fire has been a long-standing source of differing opinion between the NRC staff and the nuclear industry. For many years, the industry contended that it was extremely unlikely that a cable fire would lead to multiple spurious actuations. They argued that it should only be necessary to consider one spurious actuation for a particular cable fire or that, if multiple actuations occurred, they would be spaced sufficiently in time to permit each actuation to be mitigated separately.

In 2001, cable fire tests performed by Electric Power Research Institute (EPRI)/NEI indicated not only that multiple spurious actuations are likely to occur but also that the time between actuations may be insufficient to allow the mitigation of each actuation separately.

If a licensee has not accounted for multiple spurious actuations in its circuits analysis, it may not be in compliance with 10 CFR 50.48 and 10 CFR Part 50, Appendix A, General Design Criterion 3, which require that a licensee provide and maintain free of fire damage one train of systems necessary to achieve and maintain safe shutdown. The intent of the GL is to obtain the information needed to ensure that licensees have adequately addressed the potential for spurious actuations that compromise the capability for safe shut down.

The GL requests that each licensee:

- Within 90 days, submit a description of the plant's licensing basis with respect to the regulatory requirement for protecting redundant safe shutdown trains from multiple simultaneous spurious actuations and maintaining one train free of fire damage and submit a conclusion regarding the compliance of the plant.

  a. If not in compliance, submit a functionality assessment of systems, structures, and components (SSCs) that affect ability to achieve and maintain safe shutdown.

  b. If not in compliance, submit a description of compensatory measures put in place.

- Within 6 months, submit a plan to return all affected SSCs to compliance with regulatory requirements.

Within 30 days of issuance of the GL, the licensee can submit a request for additional time.

DISCUSSION

There are three likely approaches that the licensee will take to bring its plant into compliance:

- Make the modifications necessary to ensure safe shutdown regardless of fire location and with multiple simultaneous spurious actuations.

- Use a risk-informed approach based on Regulatory Guide 1.174 to justify exemptions or license amendments in accordance with 10 CFR 50.12 or 10 CFR 50.90.

- Adopt a performance-based fire protection program in accordance with 10 CFR 50.48, National Fire Protection Association Standard (NFPA) 805.

Among the principal comments by the industry regarding the draft GL are that it: establishes a new regulatory position; does not allow risk-informed methods (as in NEI 00-01) to be used by licensees that are not adopting NFPA 805; and imposes an unreasonable schedule for providing information.

With regard to the question whether the GL establishes a new regulatory position, the NRC's Committee to Review Generic Requirements reviewed this issue and stated that it had no objection to issuing this GL. Consequently, we did not pursue this issue further.

The request for information within 90 days regarding the extent of compliance from licensees with the regulatory intent described in the GL is reasonable. However, it is unreasonable to expect the licensees to perform the requested analyses of multiple spurious actuations within that time period, as would be necessary to assess the functionality of SSCs and to identify appropriate compensatory measures. We agree with the staff's objective to bring the licensees into compliance with regulatory requirements expeditiously. However, we recognize the magnitude of the effort required and the potential benefit of additional experiments that will be performed over the next six months. The staff has agreed to more clearly define the scope of the information that is to be provided at each deadline and to extend the time by which affected SSCs are identified and compensatory measures are reported.

Many licensees will address multiple spurious actuations by adopting a performance-based fire protection program (NFPA 805). For licensees that do not adopt the performance-based approach, a large number of exemption requests and license modifications may be required. Some combinations of spurious actuations, although conceivable, would have an extremely low frequency of occurrence. In their response to public comments, the staff indicated that the industry should develop screening tools to eliminate low-frequency combinations. In NEI 00-01, Rev. 1, "Guidance for Post-Fire Protection for Existing Light-Water Nuclear Power Plants," NEI proposes such an approach. Regulatory Issue Summary 2004-003 was developed to provide a risk-informed approach to inspections to focus on risk-significant configurations. Similar guidance could be developed as an aid to the exemption or amendment process.

The staff has agreed to clarify the scope of information to be provided at each milestone in the schedule and to provide additional time for the functionality assessment of affected SSCs. The GL should be issued after making these changes.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum dated May 10, 2006, from James E. Lyons, Office of Nuclear Reactor Regulation to John T. Larkins, Advisory Committee on Reactor Safeguards, transmitting for final ACRS review of Draft NRC Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations, and the Staff's Resolution of public comments.
2. NRC Regulatory Issue Summary 2004-03: Risk-informed Approach for Post-fire Safe-Shutdown Associated Circuit Inspections.
3. NRC Regulatory Issue Summary 2005-30: Clarification of Post-fire Safe-shutdown Circuit Regulatory Requirements.
4. Title 10 Code of Federal Regulations, 50.48 "Fire Protection".
5. U.S. Nuclear Regulatory Commission Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," July 1998.
6. Title 10 Code of Federal Regulations, 50.12 "Specific Exemptions."
7. Title 10 Code of Federal Regulations, 50.90 "Application for Amendment of License or Construction Permit."
8. NFPA 805 "Performance-Based Standard for Fire Protection for Light-Water Reactor Generating Plants."
9. NEI 00-01 "Guidance for Post-Fire Protection for Existing Light-Water Nuclear Power Plants."

that the indirect transfer of control of the license as held by FPL Energy Seabrook, is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission pursuant thereto.

The findings set forth above are supported by a safety evaluation dated August 3, 2006.

III

Accordingly, pursuant to Sections 161b, 161i and 184 of the Atomic Energy Act of 1954, as amended (the Act), 42 U.S.C. 2201(b), 2201(i) and 2234; and 10 CFR 50.80, it is hereby ordered that the application regarding the proposed merger and indirect license transfer is approved, subject to the following condition:

Should the proposed merger not be completed within one year from the date of issuance, this Order shall become null and void, provided, however, upon written application and good cause shown, such date may in writing be extended.

This Order is effective upon issuance.

For further details with respect to this Order, see the application dated January 20, 2006, and the safety evaluation dated August 3, 2006, which are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01 F21, 11555 Rockville Pike (first floor), Rockville, Maryland and accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by E-mail to *pdr@nrc.gov*.

Dated at Rockville, Maryland this 3rd day of August 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E6–13131 Filed 8–10–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on September 7–9, 2006, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Tuesday, November 22, 2005 (70 FR 70638).

Thursday, September 7, 2006, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open): The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10 a.m.: Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant (Open): The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Nuclear Management Company, LLC regarding the license renewal application for the Monticello Nuclear Generating Plant and the associated NRC staff's final Safety Evaluation Report.

10:15 a.m.–11:45 a.m.: Lessons Learned from the Review of the Early Site Permit Applications (Open): The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the lessons learned from the review of the early site permit applications for the Grand Gulf, North Anna, and Clinton sites.

12:45 p.m.–2:45 p.m.: Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements" (Open): The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revision to 10 CFR 50.68, "Criticality Accident Requirements".

3 p.m.–4 p.m.: State-of-the Art Consequence Analysis (Open): The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's plans to perform a state-of-the art consequence analysis for each site and compare the results with those in NUREG/CR–2239, "Technical Guidance for Siting Criteria Development".

4 p.m.–4:30 p.m.: EDO Response to the ACRS Report on the Review of Ongoing Security-Related Activities (Closed): The Committee will hold discussions with representatives of the NRC staff regarding the June 29, 2006 response from the NRC Executive Director for Operations (EDO) to the comments and recommendations included in the April 24, 2006 ACRS report on Review of Ongoing Security-Related Activities.

Note: This session will be closed to protect information classified as National Security information as well as safeguards information pursuant to 5 U.S.C. 552b(c) (1) and (3)].

4:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed): The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, September 8, 2006, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open): The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:30 a.m.–10:30 a.m.: Risk-Informed Criteria for Societal Risk (Open): The Committee will hear a report by and hold discussions with the cognizant ACRS member regarding risk-informed criteria for societal risk.

10:45 a.m.–11:45 a.m.: Draft Report on the Quality Assessment of Selected NRC Research Projects (Open): The Committee will discuss a draft ACRS report on the quality assessment of the NRC research projects on Containment Capacity Study at Sandia National Laboratories and on Molten Core Coolant Interaction Study at the Argonne National Laboratory.

11:45 a.m.–12 Noon: Subcommittee Report (Open): Report by and discussions with the Chairman of the ACRS Subcommittee on Thermal-Hydraulic Phenomena regarding industry perspectives on PWR sump performance issues that were discussed at the August 23–24, 2006 Subcommittee meeting.

1 p.m.–2 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open): The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

2 p.m.–2:15 p.m.: Reconciliation of ACRS Comments and Recommendations (Open): The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:30 p.m.–4 p.m.: Preparation for Meeting With the NRC Commissioners

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 13, 2006

MEMORANDUM TO: Sherry A. Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 535TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), SEPTEMBER 7-9, 2006

I certify that based on my review of the minutes from the 535th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

N/A
Comments

APPENDIX II

August 3, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
535th ACRS MEETING
SEPTEMBER 7-9, 2006

THURSDAY, SEPTEMBER 7, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   1.1) Opening statement
   1.2) Items of current interest

2) 8:35 - ~~10:00~~ A.M. **9:22 AM** — Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant (Open) (MVB/MAJ)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Management Company, LLC regarding the license renewal application for the Monticello Nuclear Generating Plant and the associated NRC staff's final Safety Evaluation Report.

10:00 - 10:15 A.M. — *****BREAK*****

3) 10:15 - ~~11:45~~ A.M. **11:14 AM** — Lessons Learned from the Review of the Early Site Permit Applications (Open) (DAP/DCF)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the lessons learned from the review of the early site permit applications for the Grand Gulf, North Anna, and Clinton sites.

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

11:45 - 12:45 P.M. — *****LUNCH*****

4) 12:45 - 2:45 P.M. — Draft Final Revision to 10 CFR 50.68,"Criticality Accident Requirements" (Open) (JSA/RC)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision to 10 CFR 50.68, "Criticality Accident Requirements."

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

(Open): The Committee will discuss topics of mutual interest for ACRS meeting with the NRC Commissioners that is scheduled for Friday, October 20, 2006.

4:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed): The Committee will discuss proposed ACRS reports.

Saturday, September 9, 2006, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

8:30 a.m.–12:30 p.m.: Preparation of ACRS Reports (Open): The Committee will continue discussion of proposed ACRS reports.

12:30 p.m.–1 p.m.: Miscellaneous (Open): The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 29, 2005 (70 FR 56936). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

In accordance with subsection 10(d) Public Law 92–463, I have determined that it will be necessary to close a portion of this meeting noted above to discuss and protect information classified as National Security information as well as safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., ET. ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., ET, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: August 7, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–13123 Filed 8–10–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on September 6, 2006, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b (c) (2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, September 6, 2006, 11 a.m.–12 Noon

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4:15 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: August 7, 2006.

Antonio F. Dias,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–13129 Filed 8–10–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Early Site Permits; Notice of Meeting

The ACRS Subcommittee on Early Site Permits will hold a meeting on September 6, 2006, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, September 6, 2006—1 p.m. Until the Conclusion of Business

The Subcommittee will review and develop "Lessons-Learned" items as a result of the three (North Anna, Grand Gulf, and Clinton) early site permits reviews. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Dominion Nuclear North Anna, LLC (Dominion), System Energy Resources, Inc. (SERI), Exelon Generation Company, LLC (Exelon), Southern Nuclear Operating Company, Inc. (Southern), and other interested persons regarding this matter. The

8. Thermal-Hydraulic Phenomena Subcommittee Report

The Chairman of the Thermal-Hydraulic Phenomena Subcommittee provided a report to the Committee summarizing the results of the August 23-24, 2006 meeting with representatives of the NRC staff, Nuclear Energy Institute, PWR Owners Group, and various PWR sump screen vendors concerning their activities related to the resolution of Generic Safety Issue (GSI)-191. The Subcommittee reviewed the tests that have been performed and encouraged both the NRC staff and the industry to continue their research and modeling efforts.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of July 14, 2006 to comments and recommendations included in the June 16, 2006 ACRS letter on the draft final Generic Letter 2006-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations." The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from July 14, 2006 through September 6, 2006, the following Subcommittee meetings were held:

- Plant Operations – July 26, 2006

The Subcommittee reviewed inspection, enforcement, and operational activities in Region I.

- Thermal-Hydraulic Phenomena – August 23-24, 2006

The Subcommittee reviewed issues associated with GSI-191 such as sump screen designs and testing, chemical effects, and downstream effects.

- Early Site Permits – September 6, 2006

The Subcommittee reviewed and developed "Lessons-Learned" items as a result of the review of three (North Anna, Grand Gulf, and Clinton) early site permit applications.

- Planning and Procedures – September 6, 2006

The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like to be kept informed of the staff's progress on a Commission paper regarding options for performing additional studies related to the third recommendation in the April 24, 2006 ACRS report on ongoing security-related activities.

- The Committee plans to review the staff's progress on technical issues associated with performing state-of-the-art reactor consequence analyses during a future meeting.
- The Committee would like to be informed of any significant changes made to the proposed revisions to Regulatory Guide 1.23 (DG-1164), "Meteorological Monitoring Programs for Nuclear Power Plants," prior to its final publication.
- The Committee plans to review the draft final version of NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," during a future meeting.
- The Committee suggested that the NRC staff consider revising the guidance associated with 10 CFR Parts 71 and 72 to allow for burnup credit, as is now permitted in the guidance for 10 CFR Part 50.

PROPOSED SCHEDULE FOR THE 536th ACRS MEETING

The Committee agreed to consider the following topics during the 536th ACRS meeting, to be held on October 4-6, 2006:

- Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment"
- Proposed Updates to Regulatory Guides and Standard Review Plan (SRP) Sections in Support of New Reactor Licensing
- Master Integrated Plan for New Reactor Licensing Activities
- Draft Report on the Quality Assessment of Selected NRC Research Projects
- Proposed Revision 1 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
- Verification and Validation of Selected Fire Models

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 11/03/2006
Date Certified: 11/13/2006

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MINUTES OF THE 535th ACRS MEETING

SEPTEMBER 7-9, 2006

REPORT:

Report to Dale E. Klein, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the Monticello Nuclear Generating Plant, dated September 19, 2006

LETTERS:

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

- Proposed Direct Final Rule to Amend 10 CFR 50.68, "Criticality Accident Requirements," dated September 21, 2006

- Lessons Learned from the Review of Early Site Permits, dated September 22, 2006

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," dated September 13, 2006

- Proposed Revision to Standard Review Plan, NUREG-0800, Section 6.1.1, "Engineering Safety Features Materials," dated September 13, 2006

- Proposed Revision to Regulatory Guide 1.23 (DG-1164), "Meteorological Monitoring Programs for Nuclear Power Plants," September 13, 2006

- Draft Final Revision to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," dated September 13, 2006

- Questions Raised by Members of the Public During the ACRS subcommittee Meeting on Palisades Nuclear Plant License Renewal Application, dated September 13, 2006

APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

CERTIFIED

MINUTES OF THE 535th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SEPTEMBER 7-9, 2006 ROCKVILLE, MARYLAND

The 535th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on September 7-9, 2006. Notice of this meeting was published in the *Federal Register* on August 11, 2006 (65 FR 46248) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. Said Abdel-Khalik, Dr. George E. Apostolakis, Dr. J. Sam Armijo, Dr. Sanjoy Banerjee, Dr. Mario V. Bonaca, Dr. Michael L. Corradini, Dr. Thomas S. Kress, Mr. Otto L. Maynard, and Dr. Dana A. Powers. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 a.m. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Wallis also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. He discussed the items of current interest and administrative details for consideration by the full Committee. He announced that the Committee had appointed two new ACRS Members, Dr. Said Abdel-Khalik and Dr. Michael L. Corradini.

II. Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant (Open)

[Note: Mr. Michael A. Junge was the Designated Federal Official for this portion of the meeting.]

The Chairman of the Plant License Renewal Subcommittee provided an introduction to the NRC staff. The Committee had the benefit of presentations and discussions with representatives of the NRC staff and the licensee regarding the license renewal application for the Monticello Nuclear Generating Plant (MNGP), and the Safety Evaluation Report (SER) prepared by the NRC staff. The operating license will expire on September 8, 2010. The applicant has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates.

Committee Action

The Committee issued a letter dated September 19, 2006, recommending that NMC's application for renewal of the operating license for MNGP be approved.

III. Lessons Learned from the Review of the Early Site Permit Applications (Open)

[Note: Mr. David C. Fischer was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and applicants to discuss any lessons that may have been learned from the preparation, evaluation, and review of the North Anna, Grand Gulf, and Clinton ESP applications. The staff and applicants agreed that there should be better communications and guidance related to the information contained in applications. Specific areas that would benefit from clearer guidance include: guidance for the electronic submission of applications, guidance on the treatment of the high frequency component of seismic ground motion, guidance for computing the probable maximum flood at proposed sites, and guidance for assuring the integrity of data posted on the Internet. Some issues that consumed a lot of time during the preparation and review of the first three ESP applications, such as the development of the "plant parameter envelope" and the review of specific major features of an emergency plan, are unlikely to require the same level of attention in the future.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated September 22, 2006, summarizing the lessons learned from the review of early site permit applications.

IV. Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements" (Open)

[Note: Mr. Ralph Caruso was the Designated Federal Official for this portion of the meeting.]

The Committee reviewed the proposed direct final rule to amend 10 CFR 50.68, "Criticality Accident Requirements."

Committee Action

The Committee issued a letter to the EDO dated September 21, 2006, recommending that the proposed direct final rule to amend 10 CFR 50.68 should be issued for public comment. The Committee also recommended that the NRC staff should complete the research to quantify the reactivity effects of fission products in the fuel. The results of this research may enable additional burnup credit to be allowed in the guidance for 10 CFR Part 71 and 72.

V. State-of-the-Art Consequence (Open)

[Note: Mr. Eric A. Thornsbury was the Designated Federal Official for this portion of the meeting.]

Dr. Bonaca, the cognizant Committee Member for this issue, introduced the topic. Dr. Bonaca provided an overview of the topic and background information on the project. He briefly discussed the September 1982 Sandia Siting Study, which used several known conservative assumptions and bounding analyses to demonstrate results that met overall risk goals. Dr. Bonaca noted that the Commission has directed the staff to work with the ACRS on technical issues such as identification of accident scenarios for evaluation, evaluation of source terms, credit for operator actions and plant mitigation systems, modeling of emergency preparedness, modeling of offsite consequences, and definition and characterization of analysis uncertainty.

NRC Staff Presentation

Ms. Michele Laur, Office of Nuclear Regulatory Research (RES), outlined the presentation, noting that the discussion would cover the objectives, approach, potential uses of the analysis, and technical details regarding the improvements in the state-of-the-art that the staff will be using in the analysis – specifically for scenario selection, accident progression, and consequence analysis. Ms. Laur discussed the objectives of the project – to perform a realistic evaluation of severe accident progression, radiological releases, and offsite consequences with a focus on scenarios most likely to contribute to a radioactive release and offsite consequences, using a risk-informed approach. She noted that a Commission paper and related SRM had been developed which described these objectives. Ms. Laur continued by describing the overall approach for the project, which included realistic modeling of plant systems, radionuclide transport, radionuclide deposition, likely release pathways, emergency preparedness, and plant improvements. The project also developed a faster-than-real-time tool to assist incident response coordinators in making decisions during actual events.

Ms. Laur then discussed the potential uses for the results of the project. These included improved safety-related decisionmaking, insights for new reactor sites, emergency response improvements, and regulatory analysis guidelines. The most important use is to provide a more accurate assessment of the potential offsite consequences for the current state of nuclear power plants as a means to improve communication with the public. Following several questions from the Members, she continued discussing the technical improvements involved in the project, some of which also served as motivation for the work. In the Level 1 portion of the analysis, the staff took advantage of improved Level 1 PRA modeling, improved plant performance, and added design features, such as station blackout improvements. For the Level 2 and 3 portions, recent phenomenological experiments provided a better understanding of source terms, the MELCOR code which provided an integrated severe accident analysis tool, and the overall increases in computing speed provided to analyze a greater range of scenarios. Ms. Laur stated that this is a three-year project and described the types of plants which would be examined each year and the types of staff that would be involved.

Mr. Chris Hunter, RES, continued the presentation by discussing a flowchart of the scenario selection process. The process begins with a screening of internal event sequences from the SPAR model, then evaluates the dominant cutsets for grouping into scenarios. The staff evaluated these scenarios for mitigation capabilities. The staff considered the effects of containment bypass scenarios and external events on the selection of scenarios. The final step results in the selection of scenarios that are likely to lead to a significant radiological release. Mr. Hunter concluded by discussing the technical issues affecting scenario selection, including the treatment of external event scenarios, the evaluation of human reliability for mitigation, and the calculation of scenario release frequencies.

Mr. Jason Schaperow, RES, discussed technical details of the accident progression analysis, including issues related to reactor coolant pump seal leakage, safety relief valve operation without power, and containment failure modes. He also discussed the issues related to consequence analysis, including representative source term definitions for plant groups and site specific factors such as emergency response, population distribution, weather data, and shielding factors.

Ms. Laur continued by describing the many types of communications occurring during the project, including steering committee meetings, ACRS meetings, EDO meetings, Commission staff briefings, and public meetings. The staff has selected six pilot sites, selected preliminary scenarios for two plant types, and held a MELCOR/MACCS expert meeting. The staff will prepare input for MELCOR and MACCS calculations for the six pilot sites and will continue examination of the SPAR models to identify accident scenarios for the remaining sites.

During the above discussions, the ACRS Members and other participants made the following points:

- Dr. Kress asked if the term 'risk-informed' meant that they would exclude some scenarios. Ms. Laur answered the question later in the presentation by showing how the staff will use the scenario screening process to select/eliminate scenarios.

- Dr. Wallis asked if the analyses would all be plant-specific. Ms. Laur answered affirmatively, adding that the MELCOR analyses would be done in plant groups, but they would perform the MACCS offsite consequence analyses for every plant.

- Dr. Apostolakis asked if they would make the results a part of the SPAR models. Ms. Laur pointed out that the project is using the SPAR models to select the scenarios. Dr. Farouk Eltawila, Director of the Division of Risk Assessment and Special Projects in RES, stated that the staff will decide later whether to incorporate any of the results into the SPAR LERF models.

- Dr. Kress asked how and where the staff plans to truncate the offsite effects. Mr. Schaperow stated that the staff will use both a dose limit and distance limit, and they will perform sensitivity studies on those variables. He pointed out that this issue was a topic at a recent expert meeting, and they are still discussing it.

- Dr. Apostolakis asked if industry was playing a part in the project. Ms. Laur stated that the staff was holding a public meeting the next day to engage them. The staff hopes the industry will help by providing important information such as meteorological data and post-accident procedures. Dr. Corradini asked if industry has a Level 3 PRA to which they could compare the work. Ms. Laur was not aware of any, but Drs. Kress, Apostolakis, and Bonaca listed a few possibilities.

- Dr. Kress asked how they would handle multiple reactors on a site. Mr. Schaperow indicated that they would treat them separately. Dr. Kress suggested the risk should be added. Mr. Schaperow replied that the analyses are not examining risk, just consequence estimates. Dr. Apostolakis then followed up by asking if uncertainty was being considered. Dr. Tinkler responded that the conditional core damage probability curves focus on low probability outcomes, while this project will focus on a best estimate of more likely scenarios. He noted that they are planning to look at the uncertainty in the predictions of consequences, through an integrated examination of uncertainty that will capture uncertainties in both the source term and consequence calculations.

- Dr. Wallis asked if they may screen out high LERF scenarios. Mr. Hunter noted that the screening values were set low enough to capture these scenarios. Dr. Apostolakis questioned if LERF would be a better screening variable. Dr. Kress pointed out that by screening with a low CDF, they would also screen on LERF. Dr. Wallis suggested the staff explicitly show the connection between the screening and LERF.

- Dr. Wallis suggested that the staff needs to be more realistic in their assessment of fire scenarios.

- Dr. Apostolakis asked if the staff is using NUREG-1150 in the analysis. Mr. Hunter replied that they would use it for a comparison check on selected scenarios.

- Dr. Bonaca asked how the new study will be comparable to the old study if the staff uses different scenarios. Dr. Eltawila replied that the staff would like the Committee's help regarding how to communicate the results.

- Mr. Sieber asked if they would include shutdown scenarios. Mr. Hunter answered that they would not since the shutdown SPAR models were not mature.

- Dr. Corradini asked if any new evidence existed on containment failure. Dr. Eltawila answered that new data and analyses now exist on containment failure, such that they can eliminate issues such as alpha-mode failure.

- Dr. Kress suggested that a good way to answer many of the questions would be to perform a full Level 3 PRA for comparison purposes.

- Dr. Corradini commented that the staff could be open to criticism for selecting only some scenarios.

- Dr. Nourbakhsh, ACRS staff engineer, noted the availability of relevant research in NUREG/CR-6295.

- Ken Canavan, EPRI, stated that much of the information the staff will need already exists at sites including scenario screening and containment failure characteristics.

- Dr. Wallis asked if the staff will capture how well the evacuation will actually work, as this is a big public concern. Mr. Schaperow noted that one member of the team is an emergency preparedness expert, and that the staff would be able to answer this question in the future. Dr. Kress commented that the results will be very sensitive to the EP assumptions. Ms. Laur confirmed that the staff recognizes its importance.

- Dr. Shack asked what kinds of consequences the staff plans to compute. Mr. Schaperow stated that early fatalities and latent cancers would be, but was not sure about land contamination. Dr. Eltawila stated that they would not compute land contamination. Dr. Kress argued that land contamination is the dominant consequence. Dr. Shack noted that we normally examine such costs as part of a regulatory analysis, and asked why it would not be done here as well. Dr. Kress pointed out that because the staff is using MACCS for the consequence analysis, the extra effort to report the land contamination consequences is minuscule. Dr. Bonaca noted that the old siting study did not include an equivalent calculation, and suggested that including one would focus the results of the study in a different direction from what the Commission intended.

- Dr. Apostolakis suggested that the staff meet with the ACRS subcommittee(s) early in the process.

<u>Committee Action:</u>

This was an information briefing. No Committee action was necessary. The Committee plans to review the staff's progress on technical issues periodically throughout the project.

VI. <u>EDO Response to the ACRS Report on the Review of Ongoing Security-Related Activities</u> (Open)

[Note: Mr. Eric A. Thornsbury was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to discuss the EDO's response of June 29, 2006, to comments and recommendations included in the ACRS' April 24, 2006 report on Ongoing Security-Related Activities. Specifically, the staff responded to a request from the Committee to clarify their response to the third recommendation. The staff discussed the preparation of an options paper for the Commission on whether to perform the types of studies discussed in the Committee's recommendation.

Committee Action:

The Committee conditionally accepted the staff's clarification of the EDO response, with particular emphasis on the options paper for further studies. The Committee would like to be kept informed of the staff's progress on the Commission paper.

VII. Risk-Informed Criteria for Societal Risk (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

Dr. Kress presented a proposal to establish risk-informed criteria for considering the acceptability of the risk associated with nuclear power plants. He noted that the current risk metrics, the core damage frequency (CDF) and the large early release frequency (LERF) do not address all types of releases, their frequencies, their frequencies, or other effects such as land contamination. He proposed that a new method be developed, using frequency- consequence curves and actual risk-benefit analyses. If the risk is very small, on the order of 0.1% of the risk associated with other activities in society, then the risk could be judged to be acceptable.

Committee Action

After a lively discussion, the Committee decided that it could not endorse Dr. Kress's proposal, but it encouraged him to continue to work on the concept, and possibly prepare a paper that might be useful in the future.

VIII. Draft Report on the Quality Assessment of Selected NRC Research Projects (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

The Committee discussed the status of the quality assessment of the research projects selected for FY 2006. The Committee discussed the results of panel review and the numerical rating scores for the projects on containment capacity studies at Sandia National Laboratory and melt coolability and concrete interaction program at Argonne National Laboratory.

Committee Action

The Committee plans to discuss the draft ACRS report on quality assessment of the selected projects during October 4-6, 2006 ACRS meeting.

IX. Thermal-Hydraulic Phenomena Subcommittee Meeting (Open)

[Note: Mr. Ralph Caruso was the Designated Federal Official for this portion of the meeting.]

The Chairman of the Thermal-Hydraulic Phenomena Subcommittee provided a report to the Committee summarizing the results of the August 23-24, 2006 meeting with the nuclear Energy Institute, the PWR Owners Group, and various PWR sump screen vendors concerning their activities related to Generic Safety Issue 191. The subcommittee members were impressed by the level of testing that was being performed, and they encouraged both the NRC staff and the industry to continue its research and modeling efforts.

Committee Action

The Committee took no specific action as a result of this presentation, but will continue to follow staff and industry efforts to resolve the GSI.

X. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

The Committee discussed the response from the NRC Executive Director for Operations (EDO) to ACRS comments and recommendations included in recent ACRS reports:

- The Committee considered the EDO's response of July 14, 2006 to comments and recommendations included in the ACRS' June 16, 2006 letter regarding draft final Generic Letter (GL) 2006-XX, "Post-Fire Safe-Shutdowns Circuits Analysis Spurious Actuations." The Committee decided that it was satisfied with the EDO's response.

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on September 6, 2006. The following items were discussed:

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the September ACRS meeting

Member assignments and priorities for ACRS reports and letters for the September ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through November 2006 were discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action.

Regulatory Guides and Standard Review Plan Updates in Support of New Reactor Licensing

The staff is in the process of developing and/or updating several Regulatory Guides and Standard Review Plan (SRP) Sections in support of new reactor licensing. This is being done to comply with the requirement in 10 CFR part 52 that all Regulatory Guides and SRP Sections that are applicable to new reactors should be completed six months prior to receiving the first COL application. Also, the Commission directed the staff to complete this task by March 2007.

The staff has identified 28 Regulatory Guides to be completed by March 2007 to comply with the 10 CFR Part 52 requirement and the Commission direction. The staff has identified several Regulatory Guides that do not need ACRS review because they either deal with process issues or the changes are minor. In addition, the staff requested that the ACRS hold a special meeting in January 2007 to review about 12 Regulatory Guides. The staff has been informed by the ACRS staff and the ACRS Executive Director that the Committee will not hold a special meeting in January 2007.

An alternate proposal by the ACRS staff was discussed and summarized below. Assuming that the staff will provide the documents by the end of September:

- In October, the ACRS will review one Regulatory Guide, and decide whether to review certain Regulatory Guides.
- In November, the ACRS is tentatively scheduled to review eight Regulatory Guides.
- In December, six Regulatory Guides are tentatively scheduled for review.

Assignments have been made for the members and ACRS staff for reviewing and/or making recommendations on whether to review these Guides. The staff is also revising the SRP Sections applicable to the future plant licensing. Upon receiving information on this matter, they will be scheduled for ACRS review.

RES and NRR staff are scheduled to meet with the ACRS during the October 2006 full committee meeting to provide the staff's views on which Regulatory Guides and SRP sections require ACRS review. Based on cognizant member's review and recommendations, the Committee will decide on a course of action. To complete the review of these Guides to accommodate the Agency schedule, the Committee may have to hold 4-day meetings in November and December.

Another option for consideration would be the establishment of an Ad Hoc Subcommittee to review these Guides and SRP Sections in October and November and refer to the full Committee only those Guides and SRP Sections that need to be reviewed by the full Committee. Following the Ad Hoc Subcommittee meeting, the Subcommittee Chairman will prepare one proposed letter commenting on all Regulatory Guides and SRP Sections and submit to the full Committee for consideration. Even with this approach, the Committee may need to hold 4-day meetings in November and December.

Quadripartite Meeting Status

In response to the invitation letters sent to NRC Commissioners, the EDO, and selected Program Office Directors, Chairman Klein has agreed to be a keynote speaker for the opening session. Dr. Paul Epstein, M.D., from Harvard University will be the keynote speaker for evening session 1. Commissioner Jaczko has agreed to be a keynote speaker for the opening session of day two. Mr. Dennis Spurgeon, Assistant Secretary of Nuclear Energy, DOE, has agreed to be a keynote speaker for evening session 2. The EDO has agreed to attend the meeting.

Arrangements have been made to visit TMI-1 Nuclear Plant on October 17, 2006. Several meeting attendees from Japan, Germany, and France as well as ACRS members Armijo, Maynard, Sieber, and Wallis will participate in this plant visit.

ACRS Meeting with the NRC Commissioners

The ACRS meeting with the NRC Commissioners is scheduled for Friday, October 20, 2006, between 2:30 and 4:30 p.m. . The following topics have been approved by the Commission:

I. Overview (GBW)
   - Accomplishments
   - License Renewal
   - Power Uprate
   - Risk-Informing 10 CFR 50.46
   - Ongoing/Future Activities

II. PWR Sump Performance (GBW)

III. Safety Research Program Report (MVB)

IV. Lessons Learned from the Review of Early Site Permit Applications (WJS)

V. Future Plant Design Activities and coordination with the NRC staff on the Master Integrated schedule. [Including 10 CFR Part 52] (TSK)

During the September ACRS meeting, the Committee needs to discuss and provide comments on the presentation slides. Following approval by the Committee at the October meeting, the final slides will be sent to the Commission.

Proposed Revision to the ACRS Subcommittee Structure

A proposed revision to the ACRS Subcommittee structure was discussed. This revision involves combining certain existing Subcommittees, the creation of new Subcommittees to work with COL applications and member assignments.

Annual Retreat, visit to a Nuclear Plant, and Meeting with the Regional Administrator

Each year, the members visit a nuclear plant and meet with the Regional Administrator to discuss items of mutual interest. In 2006, the members visited the Limerick Nuclear Plant and met with the Region I Administrator.

In 2007, the Committee will visit a plant in Region IV and meet with the Region IV Administrator. During the discussion of Risk Management Technical Specification Initiative 4b, "Use of Configuration Management for Determining Technical Specification Completion Times Related to the use of PRA and Risk-Monitoring Tools," at the April 28, 2006 Reliability and PRA Subcommittee meeting, Dr. Apostolakis suggested that in 2007 the members visit a plant with Risk Monitor. The plants in Region IV that use Risk Monitors are San Onofre, South Texas, and Fort Calhoun. It was also suggested that the 2007 plant visit and meeting with the Regional Administrator be held in January 2007.

Meeting with the Nuclear Installations Inspectorate (NII) United Kingdom

During a conversation with Mr. Paul Harvey, Principal Inspector, NII, at the July 26, 2006 meeting with the Region I Administrator, Dr. Wallis expressed some interest in a meeting between NII and members of the ACRS to discuss items of mutual interest. Subsequently, Mr. Harvey sent an e-mail to the NRC Office of International Programs (OIP), stating that NII would like to find out whether Dr. Wallis wants to pursue his interest in meeting with NII and if so when. Dr. Larkins has discussed this matter with the OIP Desk Officer for the U.K. and noted that the Committee has had bilateral exchanges with the U.K. in the past and would get back to OIP shortly.

It should be noted that the Committee met with Mr. Lawrence Williams, Her Majesty's Chief Inspector, NII during the December 5-7, 2002 ACRS meeting to discuss several items of mutual interest, including pre-decisional plans to expand the nuclear program in U.K.

Request by Mr. Herschel Specter to brief ACRS on Indian Point Emergency Planning

In an e-mail to Dr. Kress, dated August 20, 2006, Mr. Herschel Specter stated the following:

- There has been a large effort to modernize the emergency plan at the Indian Point Nuclear Plant.
- For about two years as a consultant to Entergy (the Indian Point plant owner) Mr. Specter led the technical effort to modernize the emergency plan at Indian Point. This phase of effort is nearing completion and Entergy and its supporting team would like to present their analyses to the ACRS sometime after Thanksgiving this fall.
- The NRC staff and SNL are also active in modernizing the emergency plan and they may be ready to present their results in a similar timeframe.

ACRS does not normally get involved in reviewing plant-specific emergency plans. We need to discuss with the NRC Chairman whether ACRS should get involved in this matter. In addition, since staff and SNL are involved in modernizing the emergency plan, we should wait until they complete their work. If the Commission, EDO, or the staff requests ACRS involvement in this matter, then a briefing will be scheduled and Mr. Specter, staff, SNL, EPRI, and NEI will be invited to present their views.

<u>ACRS Meeting Dates for CY2007</u>

A calendar which includes proposed ACRS meeting dates for CY2007 was discussed. The proposed meeting dates for calendar year 2007 are listed below:

- -	January 2007 (No ACRS Meeting)
539	February 8-10, 2007
540	March 8-10, 2007
541	April 5-7, 2007
542	May 3-5, 2007
543	June 6-8, 2007 (Wed. - Friday)
544	July 11-13, 2007 (Wed. - Friday)
- -	August (No ACRS Meeting)
545	September 6-8, 2007
546	October 4-6, 2007
547	October 31 - November 1-2, 2007 (Wed. - Friday)
548	December 6-8, 2007

C. <u>Future Meeting Agenda</u>

Appendix IV summarizes the proposed items endorsed by the Committee for the 536th ACRS Meeting, October 4-6, 2006.

The 535th ACRS meeting was adjourned at 7:00 p.m. on Friday, September 8, 2006.

APPENDIX III

MEETING ATTENDEES
535TH ACRS MEETING
SEPTEMBER 7-9, 2006

<u>NRC STAFF</u> (9/7/2006)
J. Storch, OIG
M. Morgan, NRR
R. Hernandez, NRR
R. Mathew, NRR
A. Pal, NRR
M. Mitchell, NRR
R. Subbaratram, NRR
D. Heng, NRR
E. Oesterle, NRR
C. Arguas, NRR
P. Prescott, NRR
J. Stravetos, NRR
M. Hart, NRR
M. Blumberg, NRR
D. Barss, NSIR
N. Gilles, NRR
S. Klementowicz, NRR
K. Campe, NRR
M. Concephow, NRR
B. Musica, NRR
J. Mitchell, RES
F. Eltawila, RES
M. Dusaniwskyj, NRR
J. Wood, RES
D. Dube, RES
S. Coffin, NRR
R. Weisman, OGC
E. McKenna, NRR
J. Lee, NRR
N. Chokshi, NRR
C. Hinson, NRR
D. Harrison, NRR
T. Affard, NRR
P. Clifford, NRR
A. Bart, NMSS
G. Tartal, NRR
C. Withe, NMSS
G. Bjorkman, NMSS
J. Eangle, NRR
J. Wilson, NRR
A. Obodoako, NRR
P. Heher, NRR
B. Ruland, NMSS
G. Mizuno, OGC
J. Lamb, OEDO
D. Merzke, NRR
C. Ader, RES
T. McCane, NRR
J. Vail, NRR
M. Lauer, RES
R. DeLaGarca, NRR
D. Ashley, NRR
K. Howard, NRR
J. Medoff, NRR
H. Asher, NRR
J. Zimmerman, NRR
J. Fair, NRR
P. Lougheed, RIII
N. Dudley, NRR
E. Gettys, NRR
K. Chang, NRR
L. Tran, NRR
B. Palla, NRR
J. Davis, NRR
R. Auluck, NRR
P. Buckberg, NRR
J. Ma, NRR
J. Ayala, NRR
A. Szabo, RES
J. Yerokuan, RES
J. Monninger, RES
C. Munson, NRR
C. Hunter, RES\

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>
J. Rootes, NMC
M. Aleksey, NMC
J. Grubb, NMC
P. Burkey, NMC
G. Eckholt, NMC
J. Pairitz, NMC
D. Potter, NMC
R. Dennis, NMC
C. Kerr, Exelon
G. Zinke, Entergy
R. Kuyler, Morgan Lewis
A. Michaiels, EPRI
S. Kraft, NEI
B. Gutherman, ACI Nuclear
B. Bradley, NEI
S. Dolby, Inside NRC
S. Leblang, NMC
T. Brookmire, Dominion
S. Nesbit, Duke Energy
R. Beall, Constellation Energy
J. Weil, McGraw-Hill

<u>NRC STAFF</u> (9/8/2006)
J. Mitchell. RES

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>
L. DeBesse, MIT

4:00 - 4:15 P.M. *****BREAK*****

15) 4:15 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of proposed ACRS reports on:
15.1) Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant (MVB/MAJ)
15.2) Lessons Learned from the Review of the Early Site Permit Applications (DAP/DCF)
15.3) Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements" (JSA/RC)
15.4) Response to the EDO on Security-Related Activities (Closed) (MVB/EAT)
15.5) Risk-Informed Criteria for Societal Risk (TSK/RC/HPN)

~~**SATURDAY, SEPTEMBER 9, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**~~

~~16) 8:30 - 12:30 P.M. Preparation of ACRS Reports (Open/Closed)~~
~~**(10:15-10:30 A.M. BREAK)** Continue discussion of proposed ACRS reports listed under Item 15.~~

~~17) 12:30 - 1:00 P.M. Miscellaneous (Open) (GBW/JTL)~~
~~Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.~~

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

9) 8:35 - 10:30 A.M. Risk-Informed Criteria for Societal Risk (Open) (TSK/RC/HPN)
Discussion with the Cognizant ACRS member regarding risk-informed criteria for societal risk.

10:30 - 10:45 A.M. ***BREAK***

10) 10:45 - 11:45 A.M. Draft Report on the Quality Assessment of Selected NRC Research Projects (Open) (DAP/WJS/HPN)
Discussion of a draft ACRS report on the quality assessment of the NRC research projects on Containment Capacity Study at Sandia National Laboratories and on Molten Core Coolant Interaction Study at the Argonne National Laboratory.

11) 11:45 - 12:00 Noon Subcommittee Report (Open) (GBW/RC)
Report by and discussions with the Chairman of the ACRS Subcommittee on Thermal-Hydraulic Phenomena regarding Industry perspectives on PWR sump performance issues that were discussed at the August 23-24, 2006 Subcommittee meeting.

12:00 - 1:00 P.M. ***LUNCH***

12) 1:00 - ~~2:00~~ P.M. **3:30 PM** Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

12.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

12.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

13) 2:00 - 2:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:15 - 2:30 P.M. ***BREAK***

14) 2:30 - 4:00 P.M. Preparation for Meeting With the NRC Commissioners (Open) (GBW, et al./JTL, et al.)
Discussion of topics of mutual interest for ACRS meeting with the NRC Commissioners that is scheduled for Friday, October 20, 2006.

2:45 - 3:00 P.M. ***BREAK***

5) 3:00 - ~~4:00~~ P.M. **4:15 PM** — <u>State-of-the Art Consequence Analysis</u> (Open) (MVB/EAT)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's plans to perform a state-of-the art consequence analysis for each site and compare the results with those in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

6) ~~4:00 - 4:30~~ P.M. **4:15-5:10 PM** CLOSED — <u>EDO Response to the ACRS Report on the Review of Ongoing Security-Related Activities</u> (Closed) (MVB/EAT)

6.1) Remarks by the Subcommittee Chairman

6.2) Discussion with representatives of the NRC staff regarding the June 29, 2006 response from the NRC Executive Director for Operations (EDO) to the comments and recommendations included in the April 24, 2006 ACRS report on Review of Ongoing Security-Related Activities.

[Note: This session will be closed to protect information classified as National Security information as well as safeguards information pursuant to 5 U.S.C. 552b (c) (1) and (3)].

4:30 - 4:45 P.M. ***BREAK***

7) 4:45 - ~~7:00~~ P.M. **6:15 PM** — <u>Preparation of ACRS Reports</u> (Open/Closed)

Discussion of proposed ACRS reports on:

7.1) Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant (MVB/MAJ)

7.2) Lessons Learned from the Review of the Early Site Permit Applications (DAP/DCF)

7.3) Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements" (JSA/RC)

7.4) Response to the EDO on Security-Related Activities (Closed) (MVB/EAT)

<u>FRIDAY, SEPTEMBER 8, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

8) 8:30 - 8:35 A.M. — <u>Opening Remarks by the ACRS Chairman</u> (Open) (GBW/JTL/SD)

September 19, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
536th ACRS MEETING
OCTOBER 4-6, 2006

WEDNESDAY, OCTOBER 4, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
- 1.1) Opening statement
- 1.2) Items of current interest

2) 8:35 - 9:30 A.M. Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment" (Open) (WJS/EAT)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff regarding draft final revision 3 to Regulatory Guide 1.7, which provides guidance for implementing the risk-informed 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors."

3) 9:30 - 11:45 A.M. **(10:30-10:45 A.M. BREAK)** Proposed Updates to Regulatory Guides and Standard Review Plan (SRP) Sections in Support of New Reactor Licensing (Open) (OLM/DCF)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding proposed updates to Regulatory Guides and SRP Sections that are being made in support of new reactor licensing, criteria used by the staff in selecting Regulatory Guides and SRP Sections applicable to new reactor licensing, and staff's recommendations that the ACRS not review certain Regulatory Guides and SRP Sections along with the reasons therefor.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

11:45 - 12:45 P.M. *****LUNCH*****

4) 12:45 - 2:15 P.M. Master Integrated Plan for New Reactor Licensing Activities (Open) (TSK/DCF)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the development of the Master Integrated Plan for new reactor licensing activities.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

	2:15 - 2:30 P.M.	*****BREAK*****
5)	2:30 - 4:00 P.M.	<u>Draft Report on the Quality Assessment of Selected NRC Research Projects</u> (DAP/HPN) Discussion of the draft ACRS report on the quality assessment of the NRC research projects on Containment Capacity Study at the Sandia National Laboratories and on Melt Coolability and Concrete Interaction Study at the Argonne National Laboratory.
	4:00 - 4:15 P.M.	*****BREAK*****
6)	4:15 - 4:30 P.M.	<u>Subcommittee Report</u> (Open) (OLM/MAJ/CS) Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the Oyster Creek license renewal application that was discussed at the October 3, 2006 Subcommittee meeting.
7)	4:30 - 6:30 P.M.	<u>Preparation of ACRS Reports</u> (Open/Closed) Discussion of proposed ACRS reports on: 7.1) Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment" (WJS/EAT) 7.2) Proposed Updates to Regulatory Guides and SRP Sections in Support of New Reactor Licensing (OLM/DCF)

<u>THURSDAY, OCTOBER 5, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

8)	8:30 - 8:35 A.M.	<u>Opening Remarks by the ACRS Chairman</u> (Open) (GBW/JTL/SD)
9)	8:35 - 10:15 A.M.	<u>Proposed Revision 1 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"</u> (Open) (GEA/EAT) 9.1) Remarks by the Subcommittee Chairman 9.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revision 1 to Regulatory Guide 1.200, which incorporates the lessons learned from the trial use of this Guide. Representatives of the nuclear industry and members of the public may provide their views, as appropriate.
	10:15 - 10:30 A.M.	*****BREAK*****

10)	10:30 - 12:00 Noon	Verification and Validation of Selected Fire Models (Open) (GEA/HPN) 10.1) Remarks by the Subcommittee Chairman 10.2) Briefing by and discussions with representatives of the NRC staff, Electric Power Research Institute, and National Institute of Standards and Technology regarding the draft final NUREG document, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." Representatives of the nuclear industry and members of the public may provide their views, as appropriate.
	12:00 - 1:00 P.M.	*****LUNCH*****
11)	1:00 - 2:00 P.M.	Preparation for Meeting With the NRC Commissioners (Open) (GBW, et al./JTL, et al.) Discussion of the following topics scheduled for discussion during the ACRS meeting with the NRC Commissioners between 2:30 and 4:30 p.m. on Friday, October 20, 2006: • PWR Sump Performance (GBW/RC) • Safety Research Program Report (MVB/HPN) • Lessons Learned from the Review of Early Site Permit Applications (WJS/DCF) • Future Plant Design Activities and Coordination with the NRC staff on the Master Integrated Schedule [including 10 CFR Part 52 Rulemaking] (TSK/DCF)
12)	2:00 - 2:45 P.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD) 12.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. 12.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.
13)	2:45 - 3:00 P.M.	Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
	3:00 - 3:15 P.M.	*****BREAK*****

14) 3:15 - 7:00 P.M. <u>Preparation of ACRS Reports</u> (Open)
Discussion of proposed ACRS reports on:

14.1) Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment" (WJS/EAT)

14.2) Proposed Revision 1 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (GEA/EAT)

14.3) Verification and Validation of Selected Fire Models (GEA/HPN)

14.4) Proposed Updates to Regulatory Guides and SRP Sections in Support of New Reactor Licensing (OLM/DCF)

FRIDAY, OCTOBER 6, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 12:30 P.M. **(10:15-10:30 A.M. BREAK)** <u>Preparation of ACRS Reports</u> (Open)
Continue discussion of proposed ACRS reports listed under Item 14.

16) 12:30 - 1:00 P.M. <u>Miscellaneous</u> (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

APPENDIX V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
535TH ACRS MEETING
SEPTEMBER 7-9, 2006

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman 1. Items of Interest dated
2	Final Review of the License Renewal Application for the Monticello Nuclear Generating Plant 2. Monticello Nuclear Generating Plant presentation by Xcel Energy [Viewgraphs] 3. Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report presentation by NRR [Viewgraphs]
4	Draft Final Revision to 10 CFR 50.68, "Criticality Accident Requirements" 4. Criticality Accident Requirements 10 CFR 50.68 Rulemaking presentation by NRR and NMSS
5	State-of-the-Art Consequence Analysis 5. State-of-the-Art Reactor Consequence Analyses (SOAR CA) presentation by NRR, RES, NSIR [Viewgraphs]
12	Future ACRS Activities/Report of the Planning and Procedures Subcommittee 6. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - September 6, 2006 [Handout #12.1]
13	Reconciliation of ACRS Comments and Recommendations 7. Reconciliation of ACRS Comments and Recommendations [Handout #**XX**]

Appendix V
5XXth ACRS Meeting

MEETING NOTEBOOK CONTENTS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
534th FULL COMMITTEE MEETING

<u>September 7-9, 2006</u>

<u>TODAY'S DATE: September 7, 2006</u>

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Jaclyn Storch	NRC-OIG
2	Michael Morgan	NRC - RLRC
3	Raul Hernandez	NRR \ DSS \ SBPB
4	Roy MATHEW	NRR/RLRC
5	Amar Pal	NRR / DE / EEIB
6	Matthew A. Mitchell	NRR/DCI/CVIB
7	Ram Subbaratnam	NRR | DLR | RLRA
8	Dan Hoang	NRR/DLR RLRC
9	ERIC OESTERLE	NRR/DNRL/NGDB
10	Christian Araguas	NRR/DNRL/NAPB
11	Paul Prescott	NRR/DE/EQVA
12	Joelle Starefos	NRC
13	Michelle Hart	NRR/DRA/AADB
14	Mark Blumberg	" " "
15	DAN BARSS	NSIR/DPR
16	Nanette Gilles	NRR/DNRL
17	Stephen Klementowicz	NRR/DIRS
18	KAZ CAMPE	NRR/DRA
19	MILTON CONCEPCION	NRR/DE/EQVA
	Bruce Musico	NRC/NSIR/DPR
	Jocelyn Mitchell	NRC/RES/
	~~[illegible]~~	~~[illegible]~~

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
534th FULL COMMITTEE MEETING

September 7-9, 2006

TODAY'S DATE: September 7, 2006

NRC STAFF - PLEASE SIGN BELOW

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	NAME	NRC ORGANIZATION
1	Stephane Coffin	NRR/DNRL/NARB
2	Robert M. Weisman	OGC/NRP
3	Eileen McKenna	NRR/DPR
4	Jay Lee	NRR/DRA/AADB
5	Nilesh Chokshi	NRR/DE
6	Charles Hinson	NRR/DIRS
7	Donnie Harrison	NRR/DRA/APLA
8	Tony Attard	NRR/DRA/SPNB
9	~~Robert~~ ~~Be~~all	
10	PAUL CLIFFORD	NRR/DSS/SNPB
11	Andrew Barto	NMSS/SFPO/TRD
12	George Tartal	NRR/DPR/PRAB
13	Carl Withee	NMSS/SFPO
14	GORDON BJORKMAN	NMSS/SFPO
15	Jason Zorn	NRR/DSS/SBPB
16	Josh Wilson	NRR/DSS/SBPB
17	Aloysius Obodoako	NRR/DCI/CSGB
18	Patrick Heher	NRR/DSS/SPWB
19	Bill Ruland	NRC/NMSS/SFPO
	Geary S. Mizuno	NRC/OGC/R&FC
	JOHN G. LAMB	NRC OEDO

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
534th FULL COMMITTEE MEETING

September 7-9, 2006

TODAY'S DATE: September 7, 2006

NRC STAFF - PLEASE SIGN BELOW

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	NAME	NRC ORGANIZATION
1	DANIEL MERZKE	NRR/DLR
2	Rodrigo De La Garza	NRR/DLR
3	Donnie Ashley	NRR/DLR
4	Kent Howard	NRR/DLR
5	JAMES MEDOFF	NRR/DLR
6	Hans Asher	NRR/DE
7	Jake Zimmerman	NRR/DLR
8	John Fair	NRR/DE
9	Patricia Lougheed	RIII
10	NOEL DUDLEY	NRR/DLR
11	Evelyn Gettys	NRR/DLR
12	Ken Chang	NRR/DLR
13	Linh Tran	NRR/DLR
14	BOB PALLA	NRR/DRA
15	Jennifer A. Davis	NRR/DLR
16	Raj Auluck	NRR/DLR
17	P T Kuo	NRR/DLR
18	Perry Buckberg	NRR/DLR
19	John S. Ma	NRR/DE
20	JUAN AYALA	NRR/DLR
	Autumn Szabo	RES/DRASP

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
535th FULL COMMITTEE MEETING
September 7-9, 2006

September 7, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	JAMES L ROOTES	NUCLEAR MANAGEMENT COMPANY
2	Michael F. Aleksey	Nuclear Management Co.
3	John Grubb	Nuclear Management Co.
4	PAT BURKE	Nuclear Management Co
5	Gene Eckholt	Nuclear Management Co
6	Joseph Pairitz	NMC
7	David J. Potter	Nuclear Management Company
	Ray Dennis	NMC – Monticello
9	Chris Kerr	Exelon
10	GEORGE ZINKE	ENTERGY / NUSTART
11	Ray Kuyler	Morgan Lewis
12	~~Christian Araguas~~	
13	[illegible]	
14	Jenny Weil	McGraw-Hill
15	Cliff Munson	NRC
16	Robert Beall	Constellation Energy
17	Steve Nesbit	Duke Energy
18	Tom Brookmire	Dominion Generation
	Suzanne Leblang	NMC
20	Albert MACHIELS	EPRI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
535th FULL COMMITTEE MEETING
September 7-9, 2006

September 7, 2006
Today's Date

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	NAME	AFFILIATION
1	STEVEN P. KRAFT	NEI
2	BRIAN GUTHERMAN	ACI Nuclear Energy Solutions
3	KENT WOOD	NRR/DSS/SPWB
4	Don Carlson	NRC/RES
5	James Metcalf	Polestar
6	SHAWN BURNS	SANDIA NATIONAL LABS
7	DON DUBE	NRC/RES
	Michele Laur	NRC/RES
9	MICHAEL DUSANIWSKYJ	NRR/DPR/PRAB
10	[illegible]	
11	Biff Bradley	NEI
12	KEN CANAVAN	EPRI
13	Steven Dolley	Fnds NRC / Watch
14	Tim McCune	NRC/NRR
15	CS HUNTER	NRC/RES
16	Jeff Wood	NRC/RES
17	JAMES VAIL	NRC/NRR
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
534th FULL COMMITTEE MEETING

September 7-9, 2006

TODAY'S DATE: September 8, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Jocelyn Mitchell	NRC/RES
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
535th FULL COMMITTEE MEETING
September 7-9, 2006

September 8, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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	NAME	AFFILIATION
1	Laurene DEBESSE	Massachusetts Institute of Technology.
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ITEMS OF INTEREST

535th ACRS MEETING

September 7-9, 2006

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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NRC ANNOUNCEMENT

SPEECHES

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CONGRESSIONAL LETTER

STATEMENT SUBMITTED TO U.S. SENATE

FINAL SIGNFICANCE DETERMATION

GENERIC COMMUNICATION

INSIDE NRC

④ NRC NEWS
U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs Telephone: 301/415-8200
Washington, D.C. 20555-0001 E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov

No. S-06-022

THE FUTURE OF NUCLEAR ENERGY IN THE U.S.

Prepared Remarks by
Chairman Dale Klein

at the
Goizueta Leadership Center
Atlanta, GA

August 29, 2006

Good afternoon. I was introduced as the new Chairman of the Nuclear Regulatory Commission, and I am new – just completing my second month. However, I have spent just about my entire career in the nuclear field, beginning with my academic training as a nuclear engineer at the University of Missouri and then at The University of Texas at Austin.

My academic training and work experience have prepared me well for my present position, but I think my Missouri roots are also a valuable qualification.

More than a century ago, an educator and politician named Willard Duncan Vandiver coined the saying that has defined my home state of Missouri for all time, a saying that I often like to quote.

Speaking to an audience in blue-blooded Philadelphia, he said, "I came from a state that raises corn and cotton and cockleburs and Democrats, and frothy eloquence neither convinces nor satisfies me. I am from Missouri. You have got to show me."

We've grown a bit in Missouri since then – we have some Republicans, and we even have a nuclear plant. But concepts don't change.

Today we hear predictions that nuclear power can make a pivotal contribution to the world in the 21st century. But when I hear it said we're going to build 50 nuclear plants in the next 20 years, I say, show me – show me the designs, and then show me the hardware and the construction, and then show me you have the people and procedures in place to run those plants in a way that protects public health and safety. And as importantly, show me that you are maintaining the capability of running the current fleet of plants at the same high level.

I think that a questioning, "show me" attitude is an absolute necessity for a regulator in this time of rebirth for the nuclear industry. Both the NRC and the industry have enormous and complex challenges facing us for the foreseeable future. Vision is a fine thing, but it will take a lot of hard work to realize the vision. The U.S. nuclear sector must recreate a nuclear design and construction industry that essentially has been dormant for the past 20 years.

I have a vision for the NRC, as well – true to my roots. First and foremost, NRC needs to be a strong regulator. We will hold our licensees accountable, will articulate our requirements clearly, we will be demanding and we will be responsive to their legitimate needs and concerns. In other words, the industry needs to show the NRC the attention to detail and the focus on quality necessary to protect the public health and safety. And in turn, the NRC needs to show the industry, the financial community – and above all, the public – regulatory stability as we all play our roles in this massive new venture.

The nuclear industry itself has more than 40 years of operating experience that are serving it well in its current operations. All of the measures of productivity and safety in nuclear plants reached impressively high levels starting in the mid-1990s, and have been maintained there since then. Operation and Maintenance electricity production costs from nuclear plants are now less expensive than from coal plants, and far less expensive than natural gas. The improved economics over an extended period, coupled with the fact that nuclear plants emit no greenhouse gases, have led to a new and intensifying interest in building new plants. Promising new technologies and a streamlined NRC licensing process are contributing to the new economic viability of nuclear energy.

You have heard something from Commissioner Merrifield about the NRC and our plans for handling this enormous influx of expected work. So I'm not going to dwell on the coming organizational and procedural changes at NRC. I am instead going to speak a bit about what the NRC will expect from the nuclear industry over the next few years . . . what they must show me and my colleagues to translate the great promise of nuclear power, and the vision of the industry's leaders, into reality. And in return, the NRC should show the industry predictability and stability.

Nuclear plants are tremendously complex industrial facilities. Their construction must be robust enough not only to contain radiation, but to control steam temperatures in excess of 500 degrees and to channel the high-voltage electricity on its way to consumers. The vast majority of the technology to accomplish those difficult tasks was developed in the United States after World War 2.

The planning, design and construction of the first generation of nuclear facilities, was an effort that occupied industrial giants such as Westinghouse and GE for decades, at a total cost well up in the hundreds of billions in today's dollars.

In the three decades since the last nuclear plant order, and the two decades since the bulk of the nuclear plant construction was completed in the U.S., the nuclear design, manufacturing, and construction industry in the U.S. has withered on the vine.

The leading U.S. firms have either ceased operation, consolidated or become subsidiaries of non-U.S. parent companies. The companies that remain have survived on retrofits and maintenance of existing U.S. plants, and plant construction outside the U.S., where new nuclear construction has continued to flourish.

If the U.S. is going to build new nuclear plants, the architect-engineers, construction and component supply industries must re-establish themselves. NRC's primary charge as a regulator is to protect public health and safety, and those planning to build these new plants must come to us with quality designs and hardware, and workable construction and operational plans to meet our rigorous regulatory standards.

It will not be feasible to manufacture all of the major plant components, such as the massive reactor vessels themselves, in the U.S. But in terms of the logistics of quality control and safety inspections, it would be desirable to have as much of the contents originate in the U.S. as possible.

Restoring the U.S. supplier network needed to provide components – from the steam generators and vessel heads to the thousands of valves, pumps, heat exchangers and other parts used in a nuclear plant – would have advantages. There are now 442 nuclear plants in operation worldwide, and 27 more under construction. The most ambitious construction projects are in China, India and Russia – all of whom have announced plans for further expansions in their nuclear power production capabilities. There will be competition for materials, and a home-grown manufacturing industry should benefit those building U.S. plants.

Whatever this country does, it is clear that nuclear power is growing elsewhere in the world. The nation would be well served if our own energy needs serve as a springboard to rebuild U.S. technology and manufacturing capabilities to something approaching the leadership the nation once enjoyed, contributing to foreign markets as well as supporting our own.

Not only does the U.S. industry need additional infrastructures to supply the components for future nuclear plants, it also needs to ensure the skilled workforce needed to manufacture them. The lack of a skilled workforce is a problem that goes far beyond the manufacturing and construction segments. The nuclear industry must answer a fundamental question regarding new plants: who will run them? What are their educational qualifications? What is their training? As a regulator, the NRC has the responsibility of asking these questions, and of determining the adequacy of the answers.

To some degree, the knowledge amassed by the industry in 40 years of operation is institutional, and is transferable to future operations. But to a large extent, the knowledge is in the minds of older workers. A nuclear industry survey shows that nearly half of current nuclear industry workers are more than 47 years old, and that nuclear energy companies could lose as many as 23,000 workers over the next five years – about 40 percent of the total jobs in the sector. That is a tremendous brain drain. How do we transfer the knowledge to their replacements – who may form the cadre of workers as the next generation of plants starts up?

At the same time, the key suppliers to the industry – the architect/engineering firms, fuel suppliers and reactor manufacturers, anticipate that 32 percent of their workers will be eligible to retire within the next three years. They clearly must be replaced and their numbers augmented if the nation is to restore its manufacturing capability.

I might add that the government also will be competing for the same nuclear-related skills. The NRC alone will increase staff by a net of 200 professionals per year through 2008 to handle the increased workload of new plant applications and other business. The U.S. Department of Energy, national laboratories, NNSA and other government agencies also have personnel needs.

The Nuclear Energy Institute estimates that 90,000 entry-level workers will be needed to support existing industry operations through 2011. The nuclear industry is working on many fronts to address this critical need – it has launched major programs to provide scholarships, training programs and recruitment drives, and so on. But I have the sense that it's just nibbling around the edges of an enormous challenge.

My background is in academia, running a university nuclear engineering program, and during my time in the University of Texas program I fought constantly against budget erosion and declining interest both by students and school administration.

Many of my nuclear colleagues at other universities fought the same fight – and some lost. The number of four-year nuclear engineering programs now stands at about 25, nationwide – down from 38 in the 1970s. That is a matter of extreme concern at a time when we need to increase the numbers of academic training grounds to meet sharply increasing needs. And the potential for increased student interest has not influenced all remaining schools. Recently the University of Cincinnati announced that it would close its nuclear engineering study. Many concerned industry and government officials, myself included, are hoping that they remain open.

The potential student interest is clearly there. A Department of Energy survey shows that undergraduate enrollment at 23 reporting institutions in nuclear engineering, health physics, radiological and related fields nationwide has increased from 668 in 2001 to 1,520 last year. Graduate enrollment has risen above 1,000.

The Navy nuclear program is not as large as it was in the past and will not supply the workforce in the same percentage.

I would suggest to you that a major industry effort is necessary, and that it must address every level of education in this country, starting with a commitment to fostering the interest in science and engineering of elementary and middle school children. We also must concentrate our efforts on women and minority students, who now represent the majority of potential candidates, but less than a quarter of the students currently enrolled in nuclear-related undergraduate programs. When I arrived at the NRC, I was pleased to note the diversity of the professional workforce. That is a trend I intend to continue and encourage.

Scholarships, training centers and recruitment efforts are commendable ways to steer the technically-inclined toward careers in the nuclear industry. So are beefed-up internship programs with meaningful work. And once they're on board, mentoring programs will help to augment training as we engage in generational knowledge transfer.

Every segment of the nuclear industry needs to work to increase the talent pool, though, so that we are not competing for a small number of candidates. If we all spend the next 20 years waving money and benefits at the same people, there will be winners and losers. And if the industry wins and the NRC loses, or the industry wins and the manufacturers lose, we all lose. This is an issue that should be addressed, urgently, at the CEO level at every company with any involvement in the nuclear industry.

I hope that I don't sound unduly alarmist or negative. Our glass is half full and not half empty. As I said, I have spent my career in the nuclear field, and I am personally excited by the possibilities ahead of us. I think the Nuclear Regulatory Commission has a very important and very positive role to play. We are gearing up for a vastly increased workload, and I am convinced that the NRC can discharge our obligation to provide rigorous regulatory scrutiny of the new reactor applications and associated duties without unnecessary delays. In fact, I believe that we will be able to reduce the lead times for regulatory approvals from their current duration while ensuring public health and safety.

I assure you that the NRC will do the hard work of creating the needed framework of regulatory stability. We, in turn, must be assured that the manufacturers, builders, owners and operators of the coming plants are prepared to meet their obligations to the public. You should show us good applications and we should show you a timely response.

- First, in my brief time at the NRC, I have been very impressed by both the competence and the dedication of the staff. I have been pleased with the quality of the work I have seen. They come early, stay late and focus on the job to be done.

- That said, the NRC itself places too much emphasis on process. I would like to see us concentrate more on progress, with no compromise on safety.

- We need to develop more milestones and deliverables, and articulate them clearly to those we regulate.

- I also would like to see the NRC focus more on real risk and less on risk that is simply perceptual. The tritium issue is an example of the latter.

Thank you. I will be pleased to answer your questions.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs Telephone: 301/415-8200
Washington, D.C. 20555-0001 E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov

No. S-06-021

THE ROLE OF RADIOLOGICAL PROTECTION RECOMMENDATIONS IN STRONG REGULATORY PROGRAMS

Peter B. Lyons
Commissioner
U. S. Nuclear Regulatory Commission

NEA Forum on the Evolution of the System of Radiation Protection
Washington, D.C.
August 28-29, 2006

Good morning, I want to extend my welcome and appreciation for your involvement in this Forum, the second of three regional Nuclear Energy Agency (NEA) Forums to discuss a new generation of draft International Commission on Radiological Protection (ICRP) recommendations. These Forums are a unique opportunity to discuss the content and possible implications of these draft recommendations that have been made available for public comment.

I want to offer a special welcome to our international attendees. I would like to particularly recognize Dr. Lars-Erik Holm, the Chairman of the ICRP, and Dr. Luis Echávarri, the Director General of the Nuclear Energy Agency. And since this is the North American workshop, I am pleased and honored to welcome the representatives from the Canadian Nuclear Safety Commission, Health Canada, and the Mexican National Commission on Nuclear Safety and Safeguards, as well as industry and professional society representatives from Canada and Mexico.

I am also pleased to welcome representatives from the United States government, including the Department of Energy, the Environmental Protection Agency, the Food and Drug Administration, and the Occupational Safety and Health Administration. In addition, I welcome U.S. State regulatory organizations, including the Organization of Agreement States and the Conference of Radiation Control Program Directors, industry representatives, and representatives from the Sierra Club and the Nuclear Information and Resource Service. All of you bring viewpoints that will contribute to the success of this Forum.

I understand that the first regional Forum, held in Tokyo in early July, was a great success, with significant feedback. In particular, I understand that during that meeting there was a growing consensus on the meaning and use of constraints, a topic that has generated much discussion in the last few years. Following this Forum, the U. S. Nuclear Regulatory Commission (NRC) staff will be hosting a separate ad hoc NEA expert group meeting on Wednesday, and, if needed, Thursday, following this workshop to collect more specific comments.

The ICRP has, for some time, embarked on an effort to expand, revise and consolidate the current set of radiological protection recommendations. I commend them on the open process that is being used to gather feedback from the many interested groups, in particular this opportunity for stakeholders from North America to discuss how the ICRP draft recommendations can best meet the health and safety needs of their national radiological protection programs. The subject of this Forum is one of fundamental importance to the NRC, as an independent regulatory agency, in our responsibilities to establish and enforce safety and security standards for civilian applications of nuclear technologies while ensuring the right balance of public health and safety requirements and impact on the industry we regulate.

The development of radiation standards is also of great personal interest to me, particularly the application and implementation of the linear-no-threshold hypothesis, despite the lack of scientific data underpinning its validity at doses below 100 mSv. I understand the draft report's view that: " The Commission [emphasizes] that whilst the linear-no-threshold hypothesis remains a plausible element in its practical system of radiological protection, biological information that would unambiguously verify the hypothesis is unlikely to be forthcoming." Nevertheless, in my view, one goal of researchers in this field should be to provide that missing biological information.

In a time when scientific information is significantly increasing, it is critical that we carefully and continually evaluate the scientific basis for radiological protection recommendations. However, it is also critical that we are clear, constructive, consistent and predictable in dealing with both licensees and the public. Thus, it is important that we take an opportunity such as this to evaluate how best to move forward without unnecessarily changing processes that are working effectively.

The NRC appreciates the long-standing contributions of ICRP to improve the understanding and regulatory framework for low-dose radiation exposures. The ICRP has, for many years, provided recommendations that supported radiation protection practice and regulation, starting in 1928 with X-rays, and moving to increasingly sophisticated approaches to calculating doses to individuals. For example, the radiation protection regulations promulgated in 1956 were based, in part, on recommendations of genetics groups that observed a linear dose-response relationship between radiation exposure and mutations in Drosophila (fruit fly).

At that time, the ICRP also suspected that there was an increased incidence of leukemia amongst the early radiologists. But they didn't have any dose information for this group of occupational workers, so a 15 rem annual limit for individual organs was recommended, based in part on the genetic fruit fly work. ICRP recommendations have continued to evolve over time as better information and knowledge on exposures has been developed. During the middle of the 1970s, the ICRP recognized that information on risk was becoming available. For the first time, principles and recommended dose limits were based on a scientific approach to risk estimation. Thus, separate recommendations were made to prevent nonstochastic effects such as skin erythema, and new recommendations were made to minimize the risk of stochastic effects like cancer and hereditary disease. Today, our radiation protection standards limit occupational and public doses to levels well below those where any of these effects can be observed, even in large populations.

This morning I would like to help set the stage for this Forum by discussing what I believe is an ongoing challenge to NRC and other regulators and the industry: the need for our regulatory programs to properly reflect the scientific evidence in an effective and efficient way. I believe that we face several challenges in this regard. First, do we have a solid, up-to-date, peer-reviewed basis for the recommendations? Second, do we have a set of recommendations that, while reflecting the science, is sufficiently pragmatic and practical to be efficient and effective in regulation and risk communications? And third, do these new ICRP draft recommendations suggest that changes are needed in our regulations, guidance, or licensees' radiation protection programs?

Let me start with the seemingly age-old question of the relationship of dose to risk. I agree with the ICRP that the so-called linear-no-threshold hypothesis is currently the most appropriate and conservative regulatory approach for managing risk from radiation exposure. Other recent reports are also evaluating this issue. This past year, the U.S. National Academies published their most recent report on Biological Effects of Ionizing Radiation (BEIR VII). Internationally, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) also is examining it. These reports have reaffirmed, for the present, that the linear-no-threshold hypothesis is an appropriate approach for radiation protection. But, by contrast, the French Academies published a report that argues in support of a practical threshold for radiation cancer risk. It is thus obvious that a great deal of work is being done in the area, but more work is needed to clarify the fundamental science.

In addition, even if we use this linear-no-threshold hypothesis, the issue of how and where to use this hypothesis deserves considerable discussion. I agree with ICRP that this hypothesis, if extended to calculate collective dose on large groups where population characteristics are poorly defined, is an inappropriate use of collective dose and is not a valid prediction of health effects from very small doses. I support ICRP's view that "Collective dose is mainly an instrument for optimization, for comparing radiological technologies and protection procedures. Collective dose is not intended as a tool for epidemiologic risk assessment and it is therefore inappropriate to use it in risk projections based on epidemiological studies." Other studies of this issue have reached similar conclusions. For example, the conference on Bridging Radiation Policy and Science concluded that "The concept of collective dose is often misapplied, e.g., to estimate health impacts of very low average radiation doses in large populations . . . Collective dose can be a useful comparative tool, for instance in the evaluation of protection options." In addition, the National Council on Radiation Protection and Measures (NCRP) Report No. 121, "Principles and Application of Collective Dose in Radiation Protection," covers many of the challenges of using collective dose.

Wildly varying estimates of risk can be derived by inappropriate use of collective dose. For example, the scientifically respected IAEA Chernobyl Forum estimated that there will be approximately 4,000 deaths associated with individuals who received the greatest radiation exposure from Chernobyl. This group of approximately 600,000 individuals includes the emergency workers, those individuals evacuated from their homes near Chernobyl and individuals living in very highly contaminated areas in Belarus, Ukraine and the Russian Federation. By contrast, some epidemiologists, in cooperation with the International Agency for Cancer Research, recently predicted that more than 40,000 cases of leukemia and solid cancer (including thyroid cancer) are expected among Europeans between 1986 and 2065 due to fallout from the 1986 accident. Finally, Greenpeace notes that "recently published figures indicate that in Belarus, Russia and the Ukraine alone the accident resulted in an estimated 200,000 additional (cancer) deaths between 1990 and 2004.

In my view, such inappropriate uses of collective dose only serve to confuse and frighten the public. After all, the average radiation exposure from Chernobyl to the 570 million residents of Europe will be approximately 0.5 mSv during that time, or less than 10 microSievert per year. Such a small dose is four orders of magnitude below the lowest level of statistical sensitivity for epidemiological studies and is well below dose variations experienced by average citizens with slightly different daily experiences. While I certainly agree with the ICRP statement that such calculations are, as they stated, "inappropriate," I encourage the ICRP to provide stronger statements to further discourage misuse of this concept and to provide recommendations on applications where collective dose may be appropriate and more important, when it is not appropriate to use collective dose.

Another issue of concern to me has been the lack of sensitivity of scientific tools for examining low dose radiation effects. For example, epidemiological studies are insensitive below doses of about 100 mSv. But, much progress has been made examining radiation effects in cellular and molecular systems. Today, assay systems are able to detect radiation-induced changes following several centiGray exposures. This represents at least an order of magnitude improvement in the state of technology, but, the regulatory community is concerned about managing public exposures several orders of magnitude below these levels. As such, I challenge the scientific community to push the boundaries of our scientific knowledge of low dose radiation effects even further. Toward this goal, the U.S. Department of Energy (DOE) is managing a Low Dose Radiation Research Program, funding research projects at a number of laboratories to help establish risk assessment standards based on a strong scientific foundation. I'm personally proud that I had the opportunity during my years on Senate staff to assist in the creation of this program.

The DOE work is focused on understanding:

- how radiation damages DNA and how the cell responds by repairing this damage;
- how radiation-induced DNA damage differs from oxidative damage induced by cellular metabolism;
- how cells respond or adapt when repeatedly exposed to radiation;
- how irradiation of a single cell impacts those cells surrounding it (that is to say, bystander effects); and
- determining if there is a genetic basis for individual differences in sensitivity to radiation exposure.

To date, this program has demonstrated new techniques and instrumentation for measuring the biological and genetic changes induced by exposure to low doses of radiation, and I applaud the efforts of the principal investigators participating in this program.

Projects funded by DOE include activities where cells can be irradiated with a single alpha particle and the response of the irradiated cell and its neighbors can be monitored. Thus far, the results on topics such as bystander effects, repair mechanisms, and individual cellular responses to radiation exposure have not led to a single clear mechanism or model for radiation damage and repair. Not surprising, what is clear is that humans are very complex organisms, and that there is a great need for continuing research to more clearly understand how we react to various hazards. Congressional testimony describing this research has stated, with confidence, that the linear-no-threshold hypothesis model is not an accurate prediction of risk at low doses, a conclusion also reached independently, as I noted previously, by the French Academies. It is my earnest hope that future work will quantify this qualitative assessment.

Nevertheless, as I mentioned earlier, the linear-no-threshold hypothesis is seen by both the BEIR VII report and the draft ICRP recommendations as providing a prudent basis for radiological protection. As a regulatory basis, it provides a consistent and predictable basis for establishing standards, and the implications and costs are fairly well known. Unfortunately, as pointed out in a report by the U. S. General Accounting Office in 2000, even with the same sets of data and the same underlying model, regulatory agencies can come to somewhat different conclusions on acceptable levels of protection, with very different public impacts.

For example, very large incremental public costs are entailed by selecting different low levels of residual dose for decommissioning projects. In a conference earlier this year held in Carlsbad, New Mexico, it was noted that cost estimates for remediation of sites such as Rocky Flats or the Brookhaven National Laboratory roughly double in going from a 25-millirem dose criterion to a 15-millirem dose criterion. With many billions of dollars of public funds being expended for such cleanups, and many workers and members of the public potentially exposed at some decommissioning sites, better understanding and consensus on such radiation dose levels is an issue of significant public impact.

The ICRP's draft recommendations also contain a number of other areas where it is critically important that we have a sound technical basis for our radiation protection standards. Changes are proposed in both the radiation weighting factors and tissue weighting factors, two key components in calculating the dose to an individual. As we discuss the recommendations over these next two days, I would encourage all of you to consider if the scientific basis has been adequately represented and justified. I would also suggest that one way to consider this issue is to ask if the report provides a sufficient and acceptable basis, within each of our legal and administrative systems, to decide if changes need to be made to our regulations and guidance.

When the ICRP embarked upon its current efforts to simplify, consolidate and update their recommendations, they had several key objectives. These objectives included: 1) to consider new biological and physical information and trends in the setting of radiation protection standards, 2) to improve and streamline the presentation of the recommendations, and 3) to maintain as much stability in the recommendations as is consistent with the new scientific information. I have already touched on the first point, that of accounting for new biological and physical information. Let me now briefly address the other two points.

Regulatory programs must provide for the protection of public health and safety. Adequate protection of public health and safety is my Agency's mandate under the law, applying to both workers and members of the public. We also have the obligation to develop a set of regulations that are predictable and stable so that the users of radioactive material know what to expect and how to function in their day-to-day activities. In the United States, licensees, such as the operators of power reactor facilities, have developed and maintained a systematic and structured approach to assure adequate protection. Their activities include a radiation protection program, administrative limits and levels, and the continuous application of the As Low As Reasonably Achievable concept, which internationally is known by the term "optimization." It is becoming increasingly apparent that the ICRP description on constraints as a boundary of optimization is a description of what our licensees are doing each and every day.

As we consider these draft recommendations, I encourage you to consider the material in the ICRP draft recommendations from the standpoint of the extent to which the text of the draft does, or perhaps does not, contribute to continuing a sound regulatory program that is up to date scientifically and builds upon the current best practices of radiation protection without unnecessarily adding new burdens, impediments, or recommendations. The desired outcome for the NRC would be that we would be able to continue a performance-based approach to regulation which clearly articulates the basic requirements and provides each licensee with sufficient flexibility to best achieve protection.

I appreciate the significance of ICRP enabling each of us to contribute to the development of recommendations and encourage each of you to actively participate in open and frank discussion during this Forum. Such exchanges strengthen the development of the ICRP radiological protection recommendations, which in turn contribute to public health and safety and the consistency and effectiveness of our respective regulatory programs.

Thank you for giving me this opportunity today, and I look forward to excellent discussions and information exchanges during the course of this Forum.

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NRC NEWS

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No. S-06-020

**REMARKS AS PREPARED FOR DELIVERY
CHAIRMAN DALE KLEIN
NUCLEAR REGULATORY COMMISSION
TO THE
NUCLEAR ENERGY INSTITUTE NSIAC DINNER**

AUGUST 16, 2006

Good evening everyone. I've been in my new position at NRC for a little over six weeks, and I'm nearly past the stage where I can get away with saying that I'm new and don't know the answers yet. So I've been giving a lot of thought to my vision for the NRC.

Most of the metaphors related to vision have to do with the vastness of the skies, and limitless horizons. Mine has more to do with my roots. More than a century ago, an educator and politician named Willard Duncan Vandiver coined the saying that has defined my home state of Missouri for all time.

Speaking to an audience in blue-blooded Philadelphia, he said, "I came from a state that raises corn and cotton and cockleburs and Democrats, and frothy eloquence neither convinces nor satisfies me. I am from Missouri. You have got to show me."

We've grown a bit in Missouri since then – we have some Republicans, and we even have a nuclear plant. But some things don't change.

When I hear it said we're going to build 50 nuclear plants in the next 20 years, I say, show me – show me the designs, and then show me the hardware and the construction, and then show me you have the people and procedures in place to run those new facilities in a way that will ensure public safety and security. And by the way, show me that you're maintaining the highest standards of safety performance for the plants already in operation.

In other words, my vision is that first and foremost NRC needs to continue to be a strong regulator. We will hold our licensees accountable. My vision also is that we, the NRC, articulate our requirements clearly, and that in addition to being demanding, we are responsive.

As you know, it's become an article of faith that just about every currently operating nuclear facility will have its license extended. The process has been operating smoothly and the licenses for half the nation's reactors already have either been renewed or are under review.

But you have undoubtedly heard that the NRC rejected the license renewal application for Beaver Valley, because it was not up to standard. We'll look at it again next year, and we'll see if it passes muster then.

That action preceded my tenure at NRC, but I agree wholeheartedly, and I'm telling you here and now that you'll see more of the same unless we see submissions of consistently high quality. We all have a lot of work before us, and the NRC is prepared to hold up our end. But the industry must do the same.

NRC is gearing up – adding personnel and reorganizing. We will increase staff by a net of about 200 positions a year through 2008. We recently created an Office of New Reactors, separate from the Office of Nuclear Reactor Regulation. And since many of the announcements of new reactor activity have come from the South, we are adding a new construction office in Atlanta, with its own Deputy Regional Administrator for Construction.

We'll also look at some possible procedural changes in the review process in the future. And we're battling for a greater share of the finite resources of government to get our expanded staff adequate office space and resources to do their jobs. I would like to see the review time required for early site permits and combined operating licenses reduced, with no compromise on safety.

That is not an unrealistic goal, if industry does its job on the front end. It's a plain fact that a quality submission – COL, license renewal, design certification, or anything else – takes less time to review than a bad one. Show me quality and clarity and the NRC should show you timeliness.

We will ask hard questions, but not in a vacuum. I am a great believer in milestones – back on the farm in Missouri, we called them "chores" – and in metrics. We will do our utmost to set out our requirements, and to let the industry know – collectively and individually – where it stands at all times.

The bulk of our questions and metrics will concern technical issues – design, construction, safety, and security. But we are also very concerned about a much more basic – human – dimension. Where is the industry going to get all of the talented people to run these advanced new plants safely while shepherding today's fleet of plants through the balance of their extended lives?

I don't think I need to run the numbers for you – NEI's own surveys chronicle the tens of thousands of professional and skilled craft workers needed to keep the current fleet in operation, including the replacements for the operators, engineers, health physicists and others who are taking their invaluable knowledge with them into retirement.

And how many more professionals and craft workers will be needed for the new plants whose applications are starting to arrive at NRC?

I know that the industry is working on many fronts to address this critical need – scholarships, training programs, recruitment, and so on. But I have the sense that we're all just nibbling around the edges of an enormous obstacle to success. You know that my background is in academia, running a university nuclear engineering program, and therefore you must know that during my time in the University of Texas program I fought constantly against budget erosion and declining interest both by students and school administration.

Many of my nuclear colleagues at other universities fought the same fight – and some lost. The number of four-year nuclear engineering programs now stands at about 25, nationwide – down from 38 in the 1970s. The number of students at those and other programs is on the rise. But even the larger numbers of nuclear-trained students will, in my judgment, fall far short of needs. We need to further increase the numbers of students in the pipeline, and preserve the remaining university research and training reactors.

I would suggest to you that a major industry effort is necessary, and that it must address every level of education in this country, starting with a commitment to fostering the interest in science and engineering of elementary and middle school children.

Scholarships, training centers and recruitment efforts are commendable ways to steer the technically-inclined toward careers in the nuclear industry. So are beefed-up internship programs with meaningful work. And once they're on board, mentoring programs will help to augment training as we engage in generational knowledge transfer.

We all need to work to increase the talent pool, though, so that we are not competing for a small number of candidates. If we all spend the next 20 years waving money and benefits at the same people, there will be winners and losers. And if the industry wins and the NRC loses, or the industry wins and the A/E's lose, we all lose. This is an issue that should be addressed, urgently, at the CEO level. For instance, we ought to be talking to the University of Cincinnati right now, to head off the closure of their nuclear engineering program.

I mentioned accountability, largely in the context of new reactor licensing and license extension, but I also expect to see evidence of an even greater emphasis on accountability in existing plant operations during my tenure. I would call it self-discipline.

And the future could well be riding on the degree of self-discipline the industry can muster. A major incident or close call is not acceptable.

I know that the industry's response to the Davis-Besse reactor vessel head degradation has been far-reaching and effective – as has the response to concurrent findings of deterioration of metal components and welds in other plants. And I have been following the very effective response to the Braidwood tritium finding. But the key word here is response.

Where is the next Davis-Besse? Where is the next Braidwood? Find it, and head it off.

On the tritium issue, the industry needs to look at educating the public. I expect more reactor sites to find tritium in the ground. You need to get ahead of the curve. You need an action plan to head off unnecessary fears.

As many of you might recall, there was a tritium issue at Brookhaven in the mid-'90s that resulted in a DOE contractor getting replaced. The NRC will continue to look at the risk-based decisions, but the industry needs to be proactive to prevent negative headlines.

I've spent the last five years working for Donald Rumsfeld at the Department of Defense, and I have learned a lot from him – you'd better learn, or you won't last long. Rumsfeld used to tell us that there are things we know, and things we know we don't know, and then there are unknown unknowns. The industry has learned a lot in 40 years of running commercial reactors, but those latter two categories still exist, and we need to take a harder look at them. None of us – not the industry, and not DOE, and not NRC – has in my view put enough money in the last decade into research issues associated with operating power plants. We need to rethink that and accord that kind of research its proper priority. We need to get ahead of the unknown unknowns.

In closing, I would like to make a few more brief points:

- First, in my brief time at the NRC, I have been very impressed by both the competence and the dedication of the staff. I have been pleased with the quality of the work I have seen. They come early, stay late and focus on the job to be done.
- That said, the NRC itself places too much emphasis on process. I would like to see us concentrate more on progress, with no compromise on safety.
- We need to develop more milestones and deliverables, and articulate them clearly to those we regulate.
- I also would like to see the NRC focus more on real risk and less on risk that is simply perceptual. The tritium issue is an example of the latter.
- I want the NRC to be a strong regulator and one that merits public confidence. We should also be predictable, giving clear guidance, receiving in return quality products from the industry and responding in a timely manner.

In closing, let me just say that I have spent my career in the nuclear field, and I'm exhilarated by the possibilities ahead of us. But the possibilities will remain only possibilities unless we all work together.

Thank you, and now I'd be happy to take your questions.

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NRC NEWS

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No. S-06-018

(Slide 1)

The NRC and Grid Stability

Remarks by Jeffrey S. Merrifield, Commissioner
U. S. Nuclear Regulatory Commission

at the

ANS Executive Conference on
Grid Reliability, Stability and Off-Site Power
Denver, Colorado
July 24, 2006

THE EVENT -

(Slide 2) On August 14, 2003, I was the Acting Chairman on what I thought was going to be just another routine day at the NRC. I had a series of scheduled meetings that day, including a briefing on grid reliability, where the staff discussed the trends in loss of offsite power events at nuclear power plants. The staff informed me that the number of these events was decreasing, which was encouraging. They also mentioned, however, that the duration of individual events was tending to be longer.

Around 4:00 p.m. that afternoon, Bill Travers, the EDO at that time, came into my office and informed me that the staff was assembling in our Operations Center in response to the automatic shutdown of several nuclear plants in the Northeast and Midwest. At that time, we did not know whether it was caused by multiple operational events or, perhaps by a coordinated act of terrorism.

(Slide 3) As information continued to pour in the rest of the afternoon and into the evening hours, we came to learn that nine nuclear power plants in the U.S., as well as 11 in Canada, and a host of coal-fired power plants had been disconnected from the grid because of electrical instabilities, resulting in the blackout of major portions of the Northeast and Midwest in the U.S. and parts of Canada. **(Slide 4)** In fact, virtually every power plant east of the Mississippi experienced voltage swings of variable amplitude, though plants further from the Northeast corridor saw only minor voltage perturbations.

(Slide 5) By the next morning, after a long night at the Ops Center, we were only beginning to understand the magnitude of the blackout. I participated in several conference calls, including calls with the White House Situation Room, to discuss the causes of the event with the staff of the National Security Council as well as various Cabinet members.

Of course, as soon as the safety of the reactors was assured, the next question was how quickly could they restart to restore electrical power to the millions of people who still were without power. We received a number of calls by Friday afternoon (August 15), including some from the White House, asking when the plants would be back on line. We also had a series of phone calls with our counterparts on the Canadian Nuclear Safety Commission.

As you all know, after a nuclear power plant shuts down, it cannot just be restarted at the flip of a switch. Components in several systems must be realigned, those systems must be walked down to confirm their readiness, and the reactor operators must go through a checklist before pulling control rods to restart the nuclear reaction. It typically takes between eight and 24 hours for a reactor to restart after it trips offline. In addition, after a station blackout event, the transmission line operators must also ensure the grid is ready before the plant can close its generator output breaker and resume supplying power to the grid. There are a number of steps required to restore electrical power once the grid has gone down. That being said, most of the nuclear power plants were restarted within a few days and the grid returned to normal.

So, what caused the event? We would eventually find that poor maintenance of transmission lines including tree trimming, lack of sensor and relay repair or replacement, poor maintenance of control room alarms, poor communications between load dispatchers and power plant operators, and a lack of understanding of transmission system interdependencies were all major contributors to the domino effect that resulted in plant after plant tripping off line because of the collapse of the electrical grid.

This event was truly a wake-up call for the North American transmission system operators as well as electricity generating companies.

(Slide 6) WHY DOES NRC CARE ABOUT GRID STABILITY?

Nuclear power reactors must be cooled continuously, even when shut down. The numerous pumps and valves in the reactor cooling systems therefore must have access to electrical power at all times, even if the normal power supply from the grid is degraded or completely lost.

As a regulator, we want to minimize the time a nuclear power plant is subjected to a complete loss of offsite power, otherwise known as Station Blackout. Even though plants are designed with emergency diesel generators to supply power to pumps and valves that keep the reactor cool when normal power is lost, we do not like to challenge those diesel generators any more than is absolutely necessary.

The NRC was concerned about grid reliability long before the 2003 blackout event. On August 12, 1999, while the Callaway plant (in Missouri) was offline in a maintenance outage, the plant saw the offsite power supply voltage fall below minimum requirements for a 12-hour period. The voltage drop they observed was caused by peak levels of electrical loading and the transport of large amounts of power on the grid adjacent to Callaway. The licensee noted that the deregulated wholesale power market contributed to conditions where higher grid power flows were likely to occur in the area near Callaway. Alliant Energy had to spend ten's of millions of dollars to install new transformers with automatic tap changers to keep voltage above minimum requirements, and capacitor banks to improve the reactive power (volt-amps reactive, or VARs) factor in the Callaway switchyard.

As a result of deregulation, many electric utilities were split into electric generating companies and transmission and distribution companies. Thus, nuclear power plants now must rely on outside entities to maintain the switchyard voltage within acceptable limits. Over time, some transmission companies have become less sensitive to the potential impacts that grid voltage can have on nuclear plant operations.

A big part of our risk-informed regulatory strategy depends on plants having access to reliable offsite power. We assume that there will be very few times when a plant will be subjected to a total loss of offsite power, and when such condition exists it will be for a relatively short period of time (hours or days rather than weeks). Our strategy of allowing more on-line maintenance to be performed on certain important safety equipment such as the emergency diesel generators makes sense as long as the risk of a plant trip remains very low during the period of time that equipment is out of service. This philosophy relies on the fact that a total loss of offsite power is a rare occurrence that will be corrected in a short period of time.

(Slide 7) WHAT DID WE DO ABOUT THE BLACKOUT?

Our mission is to ensure that our nation's nuclear power plants are operated in a manner that protects the public health and safety, and promotes the common defense and security. We initially focused our attention on the nine U.S. nuclear units that automatically shut down as designed, in response to the voltage swings on the grid. Subsequently, we concluded that all of these plants responded well to the event, and their emergency diesel generators automatically started and powered the safety equipment to ensure the reactors continued to be adequately cooled after offsite power was lost.

President Bush initiated a bilateral task force with Canada to look into the causes of the blackout and develop recommendations to avoid a recurrence. Then-Chairman Nils Diaz was the NRC representative appointed to the task force.

Several key issues related to nuclear plants were examined by the task force. These issues included:

- Did grid operators understand the potential impact of voltage and frequency instability on nuclear power plants?

- Did nuclear power plant operators have the necessary protocols and equipment to communicate with the grid operators to facilitate taking action to minimize the impacts of grid instability on nuclear plants?

- Are there practices used in the nuclear power industry that could be useful to non-nuclear power producers?

The discussion of these issues significantly raised awareness of the specialized impacts on nuclear plants and led to a number of initiatives in both the industry and the government to address those concerns.

(Slide 8) As a result of our task force participation and independent reviews and assessments, the NRC has taken several actions to improve plant readiness to react to unstable grid conditions. These actions include:

- Development of special procedures otherwise known as Temporary Instructions [TIs] for our resident inspectors to review the readiness of U.S. nuclear plants for the summer peak cooling season. The resident inspectors have used these procedures each spring for the last three years to ensure all nuclear plants are prepared for potential grid problems.

- Established protocols for equipment operability assessments and maintenance rule assessments.

- Issued a Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," which asked licensees to provide information on:
  1. communication protocols between nuclear plants and grid operators;
  2. grid analysis tools used to confirm adequate offsite power;
  3. offsite power restoration procedures; and
  4. station blackout analysis on loss-of-offsite-power frequency.

The NRC is in the process of assessing the information provided by our licensees to develop a further understanding of grid issues and to determine if further NRC action is necessary.

(Slide 9) We have not only been interacting with the licensee community. As a result of discussions I had with former Chairman Pat Wood and current Chairman Joe Kelliher, the Commission held two meetings with the Federal Energy Regulatory Commission (FERC). The first was a closed meeting held on May 19, 2005, and the second was a public meeting held on April 24, 2006, to discuss the relationship between grid reliability and nuclear plant safety. As a result of these meetings, the Commission agreed to provide FERC with data collected on the frequency and duration of offsite power events, human reliability research and other information that may help FERC better understand nuclear plant sensitivity to grid conditions. I have every confidence that any action FERC takes in response to this information will help to alleviate negative impacts on nuclear power plants in unstable grid conditions.

(Slide 10) WHAT HAS THE INDUSTRY BEEN DOING DURING THIS TIME FRAME?

In addition to the many efforts underway at the government level, the industry, to its credit, has been proactive in addressing grid stability issues. According to the data from the Edison Electric Institute, by 2008 the industry plans to almost double its investment in transmission-related activities using 2000 as a baseline. This financial commitment demonstrates a recognition on the part of industry that sub par transmission equipment can have a significant impact on grid stability.

(Slide 11) In addition, the industry has initiated a number of activities to address grid reliability.

- In December 2004, the Institute for Nuclear Power Operations (INPO) issued an addendum to their Significant Operating Experience Report (SOER)-99-1 addressing offsite power concerns in response to the northeast blackout. SOER-99-1 highlights the fact that grid reliability concerns have been an outstanding issue for some time. SOER-99-1 was first issued in December 1999 following grid events in South Africa and the U.S. The addendum expanded the original recommendations and clarified others.

- The North American Electric Reliability Council (NERC) and the Nuclear Energy Institute (NEI) have conducted a number of activities to both improve communication and establish working protocols between nuclear stations and grid operators. These actions help ensure reliable offsite power and reduce risk to the grid from maintenance activities. Again, this is not the first time NERC and NEI have sponsored workshops on grid reliability. In 2001, NERC and NEI conducted workshops on grid reliability, but the event of 2003 as well as other events highlighted the need to have additional, more focused workshops.

(Slide 12) WHERE DO WE GO FROM HERE?

Although we have done considerable work to date, there is much we must continue to focus on:

- we must concentrate on nuclear power plant safety as our primary focus;

- we must ensure that communication protocols between grid operators and nuclear plant operators are in place to assess the impacts of grid disturbances on nuclear units in real time;

- we must continue our partnerships with FERC and NERC to ensure nuclear plants are in compliance with our regulations and that grid operators adhere to FERC/NERC guidelines;

- we must continue to identify best practices between the transmission organizations and the nuclear plant operators to assist in further improvements in our electrical system; and

- we will continue to encourage nuclear plant operators and grid operators to openly discuss the issue of grid operators requesting nuclear units to down power multiple times in a short period of time due to the "largest single contingency" constraint.

Aside from the issues discussed above, there are special considerations for new reactors. Two of those issues that are of particular concern include:

- Should new units be designed to withstand a 100% load reject without shutting down?

- What is the impact of bringing LARGE baseload generators (>1,200 MW) onto the grid?

Hopefully, the new communications infrastructure that has been in place since the blackout will help find the answers to these challenging questions. We are committed to pursuing answers to these and other outstanding questions and to maintaining our strong oversight of nuclear power plants.

NRC NEWS
U.S. NUCLEAR REGULATORY COMMISSION
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Web Site: http://www.nrc.gov

No. S-06-019

Stronger Support for State and Local Governments

Prepared Remarks by

The Honorable Gregory B. Jaczko, Commissioner
U.S. Nuclear Regulatory Commission

at the
Tri-State Emergency Management Meeting
Danvers, MA
July 18, 2006

Good morning. As you heard in my introduction, I have done work in physics which involved analyzing very small systems. The emergency preparedness work you do is about large and complex systems involving many different agencies and levels of government. These present very different challenges, so I have made the effort to get a fuller understanding of this subject by visiting over a dozen nuclear power plants and meeting with public groups and local officials.

I have come to believe that emergency preparedness serves as a barometer for public confidence in the U.S. Nuclear Regulatory Commission (NRC). After all, it is the area in which our agency most closely interacts with the public and with you, state and local officials. In an emergency, licensees make protective actions recommendations, state and local officials make decisions, and the public reacts. So this is an area that we have to get right. It is important work and your citizens are depending on you.

I think we need to be doing a better job of helping you do yours.

The focus of my talk today will be on one small section of the Department of Homeland Security's (DHS) regulations governing the radiological emergency preparedness roles that federal government agencies play. 44 CFR Part 351.21 describes the NRC's role of evaluating the emergency plans to ensure they are adequate and can be implemented and Section (g) reads as follows:

> "Participate with FEMA in assisting State and Local governments in developing their radiological emergency plans, evaluating exercises to test plans, and evaluating the plans and preparedness."

The NRC clearly has the primary responsibility to ensure onsite plans provide reasonable assurance that appropriate protective measures can be taken and for reviewing DHS's offsite findings to make an overall determination of adequate protection for your communities. The regulation I just quoted makes it clear that we also have an obligation to stand with you to help you develop the plans that you submit to DHS. I believe we have ceded that responsibility to DHS/FEMA and it is time for us to stop being observers, roll up our sleeves, and join with you to fulfill that mandate.

What difference would it make, you might ask.

As I mentioned earlier, I have visited over a dozen nuclear power plants. At some of the plants I have visited, I have heard serious concerns that emergency plans will not work. I have concluded that we have not done a thorough job at the federal level of figuring out exactly what it means for a plan to "work." For instance, I often hear that evacuations would take too long, but I am unable to point to a section of our regulations that explains how long they should take because there is not one.

At a May 2nd Commission meeting I asked a panel of industry, state and local government, and public interest group representatives their understanding of what working means. They all said that a working plant is one that "protects public health and safety." And of course that is the mission and our ultimate goal. But I believe emergency preparedness is mature enough that we can do a better job of adding more specificity into our regulations to define what constitutes an acceptable level of preparedness and response capabilities.

Certainly, the NRC has the 16 planning standards detailed in section 50.47 of our regulations and we have further guidance in Appendix E. And as 44 CFR 350 .5(a) states, these regulations "apply insofar as FEMA is concerned to State and Local governments." And while those regulations and the guidance contained in NUREG-0654/FEMA-REP-1 from 1980 are helpful, there is something missing.

In emergency preparedness, the NRC has requirements for developing and maintaining plans, but not for what they must be able to accomplish. In reality, we simply have procedural regulations. We need better clarity for all of the different organizations involved to be able to do their jobs. As I see it, you are the emergency management experts and you play the critical role of protecting your citizens. There will never be an NRC employee in your community, for instance, directing traffic in the event of an evacuation, but the federal government does have a responsibility to provide you with easier access to the nuclear expertise resident in the NRC to help you do your jobs in the event of a radiological emergency.

Before I continue, I want to issue my standard disclaimer: the NRC is run by a Commission of five people. I only get one vote. But here are some of the things I believe need to change to enable the federal government to better support state, local, and licensee radiological emergency preparedness efforts.

First, I propose the start of a new dialogue on this issue. I would like for us to discuss ways to develop a set of attainable radiological emergency preparedness goals and then design steps to measure how well they can actually be met. I believe the best way to do this is to embrace the development of a performance-based definition of reasonable assurance that can be implemented in a graded approach. Let me explain.

The agency has defined performance-based requirements as those that have a measurable or calculable outcome. In general, a performance-based regulatory approach focuses on results as the primary basis for regulatory decision-making. So let us have a discussion about what the standard should be, let us quantify the protection that emergency preparedness plans and procedures should result in, and let us codify them in regulations that are objective and measurable.

I do not know what these new performance-based regulations would look like. They may focus on an evacuation time standard, an amount of dose that should be prevented or a maximum dose that can be received. Because they would be performance-based, licensees and communities would have more flexibility to address their own challenges and develop their own unique solutions to meet the reasonable assurance definition.

I think this effort should also be implemented in a graded approach. We need to ensure that the same amount of protection is afforded to citizens around all nuclear power plants and to do that we need to apportion our resources and efforts based upon the size of the EPZ populations. Having the flexibility to tailor your efforts in such a fashion would be an improvement over the current system which does not adequately recognize that each plant and each community is different. Because the NRC and FEMA regulations are mostly one-size-fits all, they do not take into account one of the fundamental principles of emergency management that all disasters are local – that each community is unique and local emergency managers must have the flexibility to adopt individual solutions.

Wouldn't it be better if you had the flexibility to look at all the hazards your state faces and put the risk from a rural nuclear power plant with a small neighboring population in its proper context?

Making emergency preparedness regulations more performance-based and flexible should be really straightforward. Having this dialogue and moving our regulations in this direction will also make it more likely that we could successfully make dramatic changes to protective action recommendations, if we find that necessary in the future. I am thinking here, of course, about the Sandia evacuation and protective action recommendation studies that the NRC has funded over the past few years. The preliminary results of these studies show that in certain emergencies resulting in releases of radiological materials – such as short duration or "puff" releases and/or in communities with longer evacuation time estimates, it may be better for people to shelter in place rather than attempt to evacuate.

There is a widespread perception that radiological emergency preparedness is equivalent to evacuation. Because there is such a belief among many members of the public that evacuation is the best option for a radiological emergency, any discussion about sheltering is seen as an admission that emergency plans will not 'work' and rather than focusing on the best way to achieve our common goal of protecting the public, the dialogue ends abruptly and results in a loss of public confidence. By making clear the ultimate performance measures we strive to meet, we are more likely to be able to gain the support of the very people that we need to listen, believe, and follow instructions to shelter in place – if in fact that is the safest course of action for a given scenario.

Just the discussion of this type of proposal will be extremely valuable. Public participation in the debate will allow concerned citizens to have their views heard and considered, and it would provide them with additional information about the efforts undertaken every day by licensees, and state, local, and federal government personnel to keep them safe.

A performance-based regulatory structure would be more efficient and would free up resources that would allow the agency to take one additional step to strengthen public confidence and ensure adequate protection: performing periodic comprehensive evaluations of radiological emergency preparedness.

The NRC only issues a comprehensive affirmative finding that both onsite and offsite emergency plans are in place around a nuclear power plant, and that they can be implemented, at the time it grants an initial operating license. We do not perform periodic reviews of emergency planning around nuclear power plants for the purpose of making a new finding of a "reasonable assurance of adequate protection of the population."

The NRC and DHS do regularly assess the plans in place through exercises and reviews, but our agencies do not periodically *re*assess that initial reasonable assurance finding – even it was made decades ago – unless and until we find a serious deficiency in a biennial exercise.

This situation is not helpful for your organizations. I am absolutely certain that state and local emergency managers and first responders are entirely dedicated to protecting their citizens. But because there is a lack of specificity in our regulations and guidance, and because there are no opportunities to periodically assess how all of the pieces fit together, there is little incentive for DHS or the NRC to provide new guidance and support for you as your community and the world we live in undergoes dramatic changes.

Performing a comprehensive review of emergency preparedness at nuclear power plants, especially if it was designed to measure the new performance indicators established in performance-based regulations, would provide us all with a crucial opportunity to strengthen public confidence in those plans and procedures. Taking this step would be an acknowledgment of the importance of this capability, and it would honestly reflect the fact that the infrastructure and populations around many plants have changed dramatically in the decades since they began operation. Encouraging public participation in the review would also allow concerned citizens to have their views heard and considered.

Most importantly, it would allow the NRC to play its rightful role of assisting your agencies radiological preparedness efforts.

I am not sure what frequency such reviews would need to be conducted. Every five or ten years? More often around more densely populated plants? Based upon a trigger such as a 50% change in population size or the development of substantial new infrastructure? All of these ideas could be debated. New nuclear power plants will require you to amend your State plans to extend their coverage to the new units, and DHS's regulations require that those amended plans be reviewed in the same manner as if they were an initial plan submission. So we will be confronting this issue in some fashion in the near future. Why not take advantage of that environment to rework and improve the system?

Another logical time to perform this comprehensive evaluation during the review of a license renewal application. As you know, the process for renewing the licenses of nuclear power plants has been established in such a way that reviews of emergency preparedness are prohibited. I do not believe that was the appropriate policy decision.

I understand the argument that emergency preparedness requirements are in effect at all times. But considering emergency preparedness during the license renewal process would be good public policy and a very valuable exercise. It would provide you with a forum to raise concerns, analyze and point out the changes that have occurred in your communities over the intervening decades, and suggest improvements. It also represents a huge opportunity to improve public confidence in the licensees and all levels of government by demonstrating how seriously we take these issues.

I recognize that it is difficult to change this process now – the Commission acted some time ago and our agency has already approved many license renewal requests. But I believe this is an issue the Commission needs to reevaluate.

The vehicle to make the types of changes I have discussed already exists – a years-long comprehensive review of emergency preparedness regulations being performed by the staff that has involved everything from the previously mentioned Sandia studies to extensive and unprecedented public participation. At the conclusion of the effort in the fall, the staff intends to present the Commission with recommendations on how to improve the overall program. I am hopeful that the Commission will take action at that time to clarify and improve our regulations. And I believe that the NRC is uniquely positioned to work with DHS to take a larger onsite and offsite role as part of this reevaluation of emergency preparedness.

After all, while the Department of Homeland Security does all-hazards work with state and local emergency managers, the NRC continues to be responsible for onsite REP and for ultimately reviewing DHS offsite findings. We make the determination that the onsite *and* offsite arrangements are in place and can be implemented. If we cannot do this, the Commission has a responsibility to require a plant to cease operation.

The significant changes I have outlined will not be easy to accomplish because emergency planning is a complex and emotional issue. It will require that the NRC continue to interact with our DHS partners and with licensees, and state and local emergency management officials to continue to look for ways to make radiological emergency planning even more effective.

We must address this issue honestly, directly, and with the full participation of stakeholders to strengthen our credibility with the public and ultimately make the job each of us does a little bit easier to accomplish. Together we can make progress and I intend to help improve emergency preparedness for the current fleet of nuclear power plants and for potential future reactors.

Attending forums such as this is one of the ways I attempt to do that because in addition to sharing my ideas with you, today's sessions will give me the opportunity to hear your concerns and recommendations and engage you directly. So, again, I appreciate this opportunity to speak to you this morning. I would also welcome any questions you may have.

August 31, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - COMGBJ-06-0005 - USE OF UNSHIELDED TRANSFER CASKS IN SPENT FUEL MOVEMENT

The exemption issued for Fort Calhoun Station's transfer of spent fuel to dry storage should not be viewed as establishing a precedent that encourages future exemption requests for transferring spent fuel to dry cask storage when a crane does not have sufficient capacity to lift and transfer the approved transfer cask. The staff should issue an appropriate generic communication on this exemption to include the facts of this scenario, the insights gained, and the Commission's expectation that such issues, to the extent practicable and appropriate, be resolved well in advance of fuel movement through the normal licensing processes. The staff should make it clear that exemption requests will continue to be reviewed based on their technical merits and the standards in 10 CFR 72.7.

The staff should inform the Commission of other situations where a plant's existing crane does not have sufficient capacity to lift and transfer an approved transfer cask. The staff should also inform the Commission of receipt of exemption requests to modify transfer casks by removing shielding in order to allow for their handling using existing cranes with capacity ratings lower than would be sufficient for handling the unmodified casks. Appropriate methods for obtaining and communicating the information requested by this SRM are to be decided by the staff.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

August 23, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - SECY-06-0168 - STAFF COMMENTS ON THE DRAFT RECOMMENDATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION

The Commission has approved the staff's proposed comments on the draft 2006 recommendations of the International Commission on Radiological Protection (ICRP), subject to the comments provided below. The staff should take a lead role in discussing these comments at the NEA/ICRP Forum later this month.

1. The Commission endorses radiological protection recommendations that can enable tangible improvements in providing for adequate protection of public health and safety, and that can be implemented by practitioners and regulatory authorities in a practical, timely, and cost effective manner. Therefore, the Commission supports the Advisory Committee on Nuclear Waste's assessment that "this ICRP document does not add value to the radiation protection programs in the United States, especially those promulgated by the Commission for its licensees and for licensees in Agreement States", and the staff's assessment that since there has not been any significant change in radiation risks, there is no compelling public health and safety argument to make any changes to the recommendations, or to national regulations that implement those recommendations.

2. The Commission specifically notes (as previously documented in the SRM on SECY-04-0223) that it is not necessary to develop a framework for radiological protection of non-human species, and Section 10 of the draft recommendations should be removed. The staff should continue to express the Commission's opposition to developing standards for protection of flora and fauna to the ICRP and IAEA in the appropriate forums.

3. The Commission strongly supports the staff's view that ICRP should not propose any numerical values that could be used as the basis for terminating a pregnancy and agree that such discussion (paragraph 263) should be removed from the ICRP recommendations document. This issue should be emphasized in a standalone general comment, as follows: "The NRC believes that discussions regarding the termination of pregnancy are beyond the scope of the ICRP's mission. Such discussions should be held on case-by-case bases between competent medical practitioners and their patients, and it is therefore inappropriate for the ICRP to propose any numerical value that could be the basis for terminating a pregnancy."

4. The Commission believes that the ICRP should be encouraged to provide stronger statements to further discourage misuse of the collective dose concept and to provide recommendations on the limited appropriate uses of collective dose. The ICRP needs to provide clear guidance with numerous examples of when it is appropriate to use collective dose and, more importantly, when it is not appropriate to use collective dose.

5. The staff should continue to support the open process that ICRP is using to gather feedback from the many groups interested in the development of these recommendations. Coincident with the upcoming NEA/ICRP Forum, the staff should urge domestic stakeholders, particularly States, industry and professional organizations, and public interest groups to submit their comments on the recommendations directly to the ICRP.

6. The Commission supports the Advisory Committee on Nuclear Waste's (ACNW) view that ICRP should not adopt a new set of tissue weighting factors and nominal risk coefficients until the assessment of the atomic-bomb data is completed and published. Additionally, the staff should continue to challenge ICRP to 1) clearly describe the technical basis for its decisions and to incorporate peer-reviewed scientific information that reflects the current state of knowledge and 2) delay finalizing the draft 2006 recommendations until the ICRP stated objectives have been fulfilled.

7. The staff should continue to monitor the DOE Low Dose Radiation Research, and the ICRP and other ongoing radiation protection activities to understand the boundaries of our scientific knowledge of low dose radiation effects.

8. The staff should address the following specific comments before providing them to ICRP:

   - Comments 60 and 61 provide an adequate discussion of a specific problem(s), but they do not provide the corrections the staff desires to be made to the report. The staff needs to be clear what they want done to address the issue.

   - Comment 76 clearly indicates what the staff wants accomplished in the report but provides no justification for the action. Some type of brief justification should be provided.

   - For Comment 78, the first two sentences are clear but the third sentence needs some type of lead in phrase to connect it to the idea in the first two sentences. Otherwise the third sentence is an apparent abrupt change in thought. A possible revision of the third sentence is "As an example, U.S. materials ...".

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

July 21, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Kenneth R. Hart, Acting Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - SECY-06-0144 - PROPOSED REORGANIZATION OF THE OFFICE OF NUCLEAR REACTOR REGULATION AND REGION II

The Commission has approved the staff's recommendation to reorganize the Office of Nuclear Reactor Regulation into two offices: the Office of New Reactors (NRO) with responsibility and authority for new reactor licensing as described in SECY-06-0144 and the Office of Nuclear Reactor Regulation (NRR) with responsibility for operating reactor licensing, subject to the comments below. The Commission also approved the staff's recommendation to create a Deputy Regional Administrator for Construction in Region II. NRC management should ensure that NRR and NRO are appropriately and adequately staffed to support the activities within each office and must make every effort to obtain the office space as soon as practicable to facilitate the reorganization.

To ensure that the reorganization results in the level of accountability and effectiveness envisioned by the Commission and in order to promote continued improvement in the major activities conducted by the offices, the staff should perform periodic self-assessments, including effectiveness reviews of each office's activities, and provide the results of these assessments to the Commission. The first self-assessment should be conducted following the first year of implementation of the organizational structure. Similar self-assessments and effectiveness reviews should be performed for the organizational changes in Region II and the recent reorganization described in SECY-06-0125, "Proposed Reorganization of the Offices of Nuclear Materials Safety and Safeguards and State and Tribal Programs."

When the transition is complete, each office will have its own Program Management, Policy Development, and Planning Staff (PMAS). The combined staffing of both PMAS organizations should result in only a minimal overall staff increase beyond that which would exist if the reorganization were not approved. The Commission supports the staff's recommended approach to support both NRR and NRO initially with the NRR PMAS. At the inception of the NRO, a few key staff including the business process integrator should be moved to the NRO PMAS. As soon as practicable, the staff should complete the organizational realignment, establishing as complete a PMAS as is necessary to support NRO.

The staff should also implement the division level organization of NRO shown for FY 2008 (i.e., 5 divisions) by January 2007. The staffing of these divisions, including the number of branches and SES managers assigned to each division, should be adjusted with time, as appropriate, to address the workload.

The staff should achieve a consistent application of technical and regulatory standards, guides and requirements, for both new plant licensing and for operating plants (e.g., through use of common standards, communities of practice, steering committees, enhanced roles of senior level staff, formalized process for documenting decisions systematically, establishing a protocol between NRR and NRO for all final resolution of technical issues). The staff should continue to look for other strategies, as appropriate, to achieve and maintain the desired consistency.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA

REDACTED VERSION

June 16, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - SECY-06-0125 - PROPOSED REORGANIZATION OF THE OFFICES OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS AND STATE AND TRIBAL PROGRAMS

The Commission has approved the proposed reorganization of the Office of Nuclear Material Safety and Safeguards (NMSS) including a merger of a portion of NMSS with the Office of State and Tribal Programs (STP), subject to the comments provided below. The resulting functional alignment will provide for effective organizational focus on each of these major areas, including the Nation's evolving energy and fuel cycle strategy and the increasing contribution of the Agreement States in the regulation of radioactive materials. As proposed, the new Office of National Materials Program (ONMP) elevates the visibility of State and Tribal programs to a major program office level. The staff should remain engaged with the States to strengthen their roles in the NMP to make it a truly national program.

The office titles, organizational changes, and functional statements for the proposed new ONMP and its three divisions should better reflect the roles of the Agreement States in the NMP and the importance of intergovernmental liaison. The staff should further refine the draft revised functional statements (as attached) to clarify these points. The functional statements should be shared with State leadership in the Organization of Agreement States and the Conference of Radiation Control Program Directors to obtain their feedback on the new functional alignment.

The Commission has disapproved the proposed 17 unbudgeted positions for FY 2007. The staff should further refine organizational changes to keep the number of unbudgeted positions as close to zero as possible.

The Executive Director for Operations should provide a recommendation to the Commission on establishing a single, visible, high-level point of contact in OEDO for the Tribes.

The staff should develop a communication plan that is fully vetted with both NMSS and STP to rollout the reorganizations to staff, States, Tribes and other stakeholders. This plan should provide for engaging internal and external stakeholders, particularly the Department of Energy and the NRC's fuel cycle licensees and applicants, to emphasize the elevated importance of this dynamic area and the Commission's intent to maintain an effective, forward-looking focus on the fuel cycle. The staff should also ensure that the organization moving to the new Executive Boulevard office space will have ample communication equipment and other infrastructure support, such as video-teleconference capability, and secure video-teleconference capability, to preclude the need to travel to the White Flint Complex for such services.

The staff should inform the Commission of the feedback from the States and the status of the above actions through a Commissioner Assistants briefing.

(EDO) (SECY Suspense: 6/26/06)

Attachment: Proposed Function Statements for the Office of National Materials Program (ONMP)

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
Jesse Funches
Jack Strosnider
Janet Schlueter
James McDermott

Attachment

PROPOSED FUNCTIONAL STATEMENTS
OFFICE OF NATIONAL MATERIALS PROGRAM (ONMP)

The Office of National Materials Program (ONMP) is the program office within the U.S. Nuclear Regulatory Commission (NRC) which, in close partnership with other Federal agencies, Agreement States, Non-Agreement States, Native American Tribal governments, the public, and other stakeholders, implements NRC's responsibilities to regulate nuclear material. The creation of the ONMP reflects the changing responsibilities of the NRC and Agreement States as more states become Agreement States.

The ONMP, in cooperation with Agreement States, licensees, the public, and other stakeholders, develops and implements rules and guidance for the safe and secure use of source, byproduct and special nuclear material in industrial, medical, academic, and commercial activities, and at decommissioning, uranium recovery, low-level waste, and incidental waste sites. The ONMP also conducts rulemakings for NMSS and NSIR related to materials issues. Other specific regulatory functions within the ONMP include licensing, oversight, support for regulatory decision-making, and the resolution of safety issues resulting from assessments of operational experience. The ONMP develops policies and procedures for assessing the performance of licensing and inspection functions of NRC's Regions and Agreement States through the Integrated Materials Performance Evaluation Program. It is through this program that NRC exercises its oversight responsibility under Section 274 of the Atomic Energy Act, as amended, to ensure that the Agreement States maintain adequate and compatible radiation protection programs. The ONMP provides and updates guidance on licensing in the regions and provides direction on training priorities for materials licensing and oversight of the training program. The ONMP also presents testimony on technical and policy positions on certain matters arising before the Atomic Safety and Licensing Board (ASLBP) and supports research activities of the NRC's Office of Nuclear Regulatory Research. The ONMP coordinates environmental reviews under the National Environmental Policy Act for both it and NMSS. The ONMP is responsible for all safety and security interface issues between NRC and the Agreement States. The ONMP also coordinates with the NRC's Office of Nuclear Security and Incident Response on the necessary contingency planning and emergency response operations association with source, byproduct and special nuclear material under its purview. The ONMP fosters close coordination and cooperation between NRC, the Agreement States, non-Agreement States, local officials, other Federal agencies and Native American Tribal governments. The ONMP also participates in international activities as appropriate, in coordination with the Office of International Programs.

OFFICE OF NATIONAL MATERIALS PROGRAM (NMP)
Division of Industrial and Medical Nuclear Safety (IMNS)

Working with the Agreement States, the non-Agreement States, the NRC Regional Offices, licensees, and the public, structures and implements the National Materials Program to enable the safe use of radioactive materials in medical, industrial, and academic applications for beneficial civilian purposes. Oversees licensing, inspection, event response, allegation management, analysis of licensee performance and other regulatory activities for radioactive material licensed under the Atomic Energy Act of 1954, as amended, and the Energy Policy Act of 2005. Provides technical support and guidance to the States and Regions on licensing, inspection, and enforcement activities. Develops policy and procedures for assessing Regional performance of materials licensing and inspection activities and Agreement State adequacy and compatibility.

Coordinates closely with the Agreement States to plan and provide for compatibility in regulatory approaches. Reviews Agreement State programs for continued adequacy to protect public health and safety and compatibility with NRC's regulatory program through the Integrated Materials Performance Evaluation Program. Provides technical support for training of regional and Agreement State licensing and inspection staffs.

Incorporates information technology tools into the National Materials Program and manages the use of these tools to improve the safety and control of licensed and registered radioactive materials. Plans and coordinates all activities involving the Advisory Committee on Medical Uses of Isotopes. Manages agency programs for "exempt" use of radioactive material, generally-licensed device registration, and for evaluation of sealed sources and devices. Responds to allegations involving NRC licensees and manages allegations involving Agreement State programs.

Directs contingency and response operations dealing with accidents, events, and incidents under ONMP's responsibility. Implements the emergency preparedness and emergency response functions for materials regulated by ONMP.

Represents NRC in international activities in its area of responsibility in coordination with the Office of International Programs.

OFFICE OF NATIONAL MATERIALS PROGRAM (NMP)
Division of Waste Management and Environmental Protection (DWMEP)

Directs the NRC's program for the regulation of Decommissioning, Environmental Protection, Low-Level Waste (LLW) and Uranium Recovery in close coordination with other Federal agencies, States, Native American Tribal Governments, licensees, and the public. Oversees decommissioning and clean up of contaminated sites, safe management and disposal of LLW, uranium recovery activities, and guidance for ONMP and NMSS environmental compliance. Develops and implements the regulatory program under the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA), the Uranium Mill Tailings Radiation Control Act, the National Environmental Policy Act of 1969, the West Valley Demonstration Project Act, the Ronald Reagan Defense Authorization Act and implementation of the license termination criteria in Title 10, Code of Federal Regulations, Part 20. Serves as the focal point for implementing the NRC's materials, power reactor, and non-power reactor decommissioning programs. DWMEP through the Ronald Reagan Defense Authorization Act for Fiscal Year 2005 (NDAA) consults with DOE on its incidental waste determinations for selected sites and monitors DOE incidental waste disposal activities. Provides programmatic and technical support to Agreement States on uranium recovery issues. Serves as the NRC's lead for ensuring the safe implementation of the Department of Energy's (DOE's) Remedial Action Plans, for Title I sites under UMTRCA. Establishes policy and guidance for environmental reviews to ONMP, NMSS, and the regions. Prepares Environmental Impact Statements (EISs) for NMSS and ONMP licensing activities. Provides technical and programmatic support to the Federal Energy Regulatory Commission, Office of Energy Projects for the Federal Dam Safety Program.

Plans and coordinates activities, as appropriate, with the Advisory Committee on Nuclear Waste (ACNW) in areas of its responsibility. DWMEP provides technical support for training of regional and Agreement State staff in the areas of decommissioning, uranium recovery, and environmental compliance. Represents NRC in international waste management and decommissioning activities.

OFFICE OF NATIONAL MATERIALS PROGRAM (NMP)
Division of Intergovernmental Liaison and Rulemaking (DILR)

The Division of Intergovernmental Liaison and Rulemaking (DILR) establishes and maintains effective communications and working relationships between the NRC and States, local governments, other Federal agencies and Native American Tribal Governments. DILR serves as the primary contact for policy matters between NRC and these external groups. DILR ensures overall coordination of interactions on waste and materials. DILR keeps the NRC apprised of these groups' activities as they may affect NRC and conveys to NRC management these groups' views toward NRC policies, plans, and activities.

DILR provides guidance to states intending to become Agreement States and reviews new Agreements, in coordination with other NRC offices and the Regions, for Commission review and approval. DILR works in cooperation with Federal, State, and local governments, interstate organizations and Native American Tribal Governments to ensure that NRC maintains effective relations and communications with these organizations and promotes greater awareness and mutual understanding of the policies, activities, and concerns of all parties involved, as they relate to NRC and Agreement State regulated facilities. DILR coordinates information exchange to and from the NRC's Regional State Liaison Officers (RSLO) and Regional State Agreement Officers (RSAO) in support of the activities of the Office of the NMP. DILR maintains coordination and communication with the Governor-appointed State Liaison Officers in all 50 States on materials, waste, security and reactor program issues.

DILR develops, in consultation with the Agreement States, where appropriate, needed regulations for ONMP, NMSS and NSIR . DILR coordinates the review and planning of all rulemaking activities related to waste, materials, spent fuel transportation, storage or disposal and security in these subject areas and monitors and schedules rulemaking to ensure that rules are developed in the time frame specified by the Commission.

August 28, 2006

The Honorable Michael Chertoff
Secretary
U.S. Department of Homeland Security
Washington, D.C. 20528

Dear Mr. Secretary:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I want to take this opportunity as we approach the fifth observance of the 9/11 terrorist attack to share with you NRC's perspective on security improvements made over the past five years. In short, we have significantly enhanced the security and emergency preparedness programs for NRC-licensed nuclear facilities and have further improvements planned for the future. I believe that our two agencies have established a close and highly productive working relationship not only with each other but with other Federal, State, and local agencies and with the nuclear sector, and I want to reaffirm the NRC's commitment to work collaboratively with the Department of Homeland Security (DHS) to achieve our common goal to protect the homeland and the American public.

The state of security at our nuclear facilities is strong and will get stronger as the NRC, DHS, other agencies, and those in the nuclear field continue their excellent cooperation. As someone with extensive experience in nuclear matters at the Department of Defense and having seen the security arrangements at commercial nuclear power plants and Category 1 fuel cycle facilities, I am confident that these plants are very secure.

Let me share with you some of the highlights of recent years:

- The NRC's budget for nuclear security has increased more than ten-fold since 9/11.

- The defenses of nuclear plants are being tested through the force-on-force program nearly three times as often as before and in a much more realistic fashion; these defenses are robust. We are on track to complete the full cycle of initial tests by December 2007.

- The DHS comprehensive review of the nuclear sector has yielded additional improvements in plant security.

- The Nation has a substantially better system to secure risk-significant radioactive material as reflected in the Radiation Source Protection and Security Task Force Report just sent to Congress and the President.

As you know, following the events of 9/11, the NRC took a series of actions designed to enhance the already strong security at commercial nuclear facilities in the United States. In developing these enhancements, the Commission drew on its previous experience, on a robust research program on potential vulnerabilities at commercial nuclear facilities, and on the means to mitigate those vulnerabilities. Initially, the Commission focused on the most important facilities and materials, issuing orders, for example, for additional security measures at power reactors and major fuel cycle facilities in 2002. I am pleased to report that all facilities for which the Commission believed additional security requirements were needed now have those measures in place.

NRC has significantly increased its security inspection activities. In fiscal year (FY) 2001, NRC security inspection hours at reactor facilities for baseline and follow-on inspections totaled approximately 1600 hours. In FY 2006, NRC baseline and follow-on inspections are projected to be over 8000 hours. Force-on-force inspection activity totaled approximately 2000 hours in FY 2001 and is projected to be approximately 7700 hours in FY 2006. We have conducted 37 force-on-force exercises at power reactors and Category 1 fuel cycle facilities since November 2004, and as noted earlier, are on schedule to complete exercises at all 66 sites by December 2007. NRC has similarly increased its inspections at materials facilities.

More importantly, NRC has greatly enhanced the quality of its security inspections. In force-on-force exercises at power reactor and Category 1 fuel cycle facilities, NRC uses highly trained adversary forces. In addition, NRC expert advisors deeply knowledgeable about the security of these facilities oversee the design of the attack scenarios. Evaluation of these exercises has been made far more objective through the use of MILES (Multiple Integrated Laser Engagement System) gear. We continue to identify and implement enhancements to these exercises. NRC also plans to incorporate the JCATS (Joint Conflict and Tactical Simulation) modeling system, a tool widely used by DOD, DOE and the FBI, into force-on-force exercise planning.

The NRC has also benefitted from insights derived from our involvement in the DHS Comprehensive Review process for power reactors. The nuclear sector was the first to volunteer for such comprehensive reviews, which have now been carried out at more than half of the 64 power reactor sites. Federal, State, and local first responders have been actively involved in the reviews, and gaps/shortfalls identified are being addressed. The Commission fully concurs with DHS Under Secretary for Preparedness George Foresman's comments at Calvert Cliffs on July 19, 2006, that security at nuclear power reactors is unmatched in the critical infrastructure.

The NRC has also worked with other Federal partners to enhance the security at power reactors. Most notable is our partnership with NORAD/NORTHCOM (North American Aerospace Defense Command/United States Northern Command) to provide advance warning of commercial aircraft diversions that could potentially affect power reactor facilities. NRC has utilized the insights from its classified research on security assessments to direct that appropriate imminent threat procedures be developed at each power reactor. Implementation of these procedures significantly enhances mitigation capabilities. These procedures have been inspected at all 64 power reactor sites. NRC has also utilized the insights from its security assessments to enhance spent fuel pool security and mitigation capabilities. More broadly, as a result of NRC initiatives, all power reactor licensees are in the process of formalizing extensive damage mitigation guidelines that will provide unprecedented capability to cope with damage caused by potential terrorist attacks. These guidelines will provide for the use of all reasonably available resources in the event of extensive damage to the site.

The Commission has three important rulemakings regarding power reactor and Category 1 fuel cycle facility security currently underway. The first, among other things, would make generically applicable the security requirements previously imposed by the Commission's April 29, 2003 Design Basis Threat (DBT) Orders, consistent with insights gained since then and consideration of specific factors as directed by Congress in the Energy Policy Act of 2005. A final rule will be completed later this year. The second concerns generally applicable security requirements for power reactors. The proposed rule is being prepared for publication in the Federal Register for public comment, and a final rule is targeted for completion at the end of 2007. The third rule will establish the security analysis to be submitted for new reactor designs. The staff will submit a proposed rule in September with the goal of a final rule by next summer. All of these rulemakings are important to establishing a stable regulatory framework for both existing reactors and for new reactor license reviews. DHS, pursuant to Section 657 of the Energy Policy Act of 2005, will also have a prominent role in the new reactor licensing process. The NRC looks forward to working with DHS to ensure timely DHS consultation on security and emergency preparedness matters in the licensing process.

NRC and DHS continue to work together to develop and improve emergency response initiatives for power reactor facilities. Our combined efforts have resulted in specific enhancements to security-related drills focused on licensee/first responder coordination and Federal agency support activities under the National Response Plan. The Commission anticipates further improvement in the capabilities of licensees and off-site response organizations to respond to a spectrum of events through our joint review and revision of emergency preparedness exercise evaluation criteria.

While I have focused above on power reactor and Category 1 fuel cycle facilities, NRC has put in place similar risk-informed security enhancements for other classes of licensees and for the transportation of spent fuel and risk-significant radioactive material. NRC has had in place since August 2002 a graded security framework that parallels the Homeland Security Advisory System for various classes of NRC licensees.

The Commission has taken a leadership role in establishing an enhanced security framework for risk-significant radioactive sources, not just for the United States, but also for the international community. Working with the Departments of Energy and State, NRC greatly influenced the International Atomic Energy Agency (IAEA) Code of Conduct on the Safety and Security of Radioactive Sources, which was completed in September 2003. Since then the Commission has used the Code of Conduct as the organizing principle for the security enhancements for licensees possessing risk-significant sources as defined in the Code. The full details are in the Radiation Source Task Force report submitted to the President and the Congress on August 15, 2006. In short, we have today an interim data base on all risk-significant radioactive sources and will next year have a National Source Tracking System that meets the needs of NRC, DHS, DOE, DOJ, and the States. Since the start of the year, NRC has required export and import licenses for all risk-significant sources. So far, 83 nations have either implemented or have stated their intention to implement the Code, and the Department of State, NRC, and DOE are providing support to IAEA to accelerate implementation around the globe. The U.S. has taken a leadership role through the G-8 summit process.

The Commission could not have achieved these results without the strong support of the Congress. Congress has fully supported (and often augmented) NRC's security requests. The Energy Policy Act of 2005 included a series of provisions on nuclear security, many long sought by the Commission, that the Commission is implementing.

In short, Mr. Secretary, the state of security in the nuclear sector as a result of the efforts of our two agencies; our Federal, State, and local partners; and NRC and Agreement State licensees is strong and will become stronger once the initiatives I have described are fully implemented.

Sincerely,

/RA/

Dale E. Klein

cc: See attached list

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

COMMITTEE ON ENERGY AND NATURAL RESOURCES

UNITED STATES SENATE

CONCERNING

S. 2589

NUCLEAR FUEL MANAGEMENT AND DISPOSAL ACT

PRESENTED BY

MARTIN J. VIRGILIO
DEPUTY EXECUTIVE DIRECTOR FOR
MATERIALS, RESEARCH, STATE AND COMPLIANCE PROGRAMS
OFFICE OF THE EXECUTIVE DIRECTOR FOR OPERATIONS

SUBMITTED: AUGUST 3, 2006

Introduction

Mr. Chairman and Members of the Committee, it is a pleasure to appear before you today to discuss S. 2589, the Nuclear Fuel Management and Disposal Act, which has several provisions that affect the Nuclear Regulatory Commission (NRC).

It is important to make clear at the outset that, because of the NRC's licensing and adjudicatory role in the national repository program, the NRC is not taking a position on most of the provisions in the legislation, which appear to be aimed at facilitating eventual operation of the proposed repository at Yucca Mountain.

However, some of those provisions, if enacted, could adversely impact the NRC's ability to meet its statutory obligations with respect to radioactive high-level waste. The Commission offers the following comments on provisions in the bill that would affect the timing of the Commission's review of a Department of Energy (DOE) application for a license to receive and store waste at the proposed Yucca Mountain high-level waste repository. These provisions are the subject of a letter we sent the Committee on June 30, 2006, and the points we are going to make here today are the points that we made in that letter.

Time Needed for Adequate Review

The Commission fully understands the importance of addressing the storage and disposal of high-level radioactive waste in a manner that is both safe and timely. The Commission has a record of moving responsibly and promptly to meet its obligations under the Nuclear Waste Policy Act. We continue our preparations for conducting an independent safety review of a Yucca Mountain application. We are confident that we will be ready to receive an application that DOE now says it will submit to us in 2008. We are also confident that we will reach a decision on the application within the time constraints set forth in the Nuclear Waste Policy Act assuming DOE submits a high-quality license application.

At the same time, our long experience in dealing with applications for major nuclear projects has made us keenly aware of the level of effort required to conduct a thorough licensing review that meets our statutory obligations to protect public health and safety, and to promote the common defense and security. Our main concern here is that the NRC be given sufficient time to conduct a comprehensive review of DOE's applications.

Accordingly, we are concerned with Section 4(b) because it appears to give the NRC insufficient time to review an application to license receipt and possession of waste at the proposed repository. Section 4(b) imposes a 1-year limit (with the possibility of a six-month extension) on the NRC's licensing decision. This deadline does not appear achievable to us for at least three reasons.

First, the NRC staff's technical, environmental, and legal reviews are likely to take more than a year, particularly because the staff is almost certain to ask questions about the application, and to ask for additional information in support of the application. Even the staff's reactor renewal reviews, which are widely recognized as efficient, have required about two years for each application (22-30 months, depending upon whether a hearing is requested and granted), and yet those reviews focus on a relatively narrow range of issues at facilities we have regulated for several decades.

Second, even the informal adjudicatory proceeding called for in the bill would contain certain necessary processes that cannot be carried out quickly. For example, the bill provides for limited discovery; add to this the Commission's own default proceedings, which, though less formal than trial-type proceedings, nonetheless call for written testimony, allow for questioning by the presiding officer, and allow for appeal of the presiding officer's decision to the Commission. The NRC cannot complete, in one year, both the staff's safety review and the adjudicatory proceeding.

Third, another provision in Section 4 might increase the scope of the licensing decision, and thus the time needed to make the decision: Section 4(a) of the bill provides that an application for construction authorization "need not contain information on surface facilities other than surface facilities necessary for initial operation of the repository." This provision might be read simply to place certain surface facilities outside the NRC's jurisdiction, in which case the provision would reduce the time licensing might take; on the other hand, the provision might be read to provide for staged consideration of surface facilities. Under this latter interpretation, the NRC would review certain facilities as part of its decision on construction authorization, but review others during the later receipt and possession phase, with the result that Section 4(a) would increase the scope of the receipt and possession review, and yet Section 4(b) would decrease the time allowed for that review. The intent of this provision needs to be clarified. Section 4(b) also should be revised to make clear whether the use of informal proceedings in hearings is intended to apply to the multiple amendments to the license to receive and possess that are envisioned with a phased approach for the potential repository.

For these reasons, the NRC would urge that the time for deciding on the application to receive and possess waste be increased to two years after the docketing of the application, with the possibility of an extension of six months.

We appreciate the opportunity to appear before you today, and the Commission looks forward to continuing to work with the Committee on this proposed legislation. We welcome your comments and questions.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

July 25, 2006

EA-06-071

Virginia Electric and Power Company
ATTN: Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Innsbrook Technical Center - 2SW
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (Surry Power Station - NRC Inspection Report Nos. 05000280/2006010 and 05000281/2006010)

Dear Mr. Christian:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving the failure of Surry Nuclear Station's full-scale exercise critique to identify a weakness associated with a risk-significant planning standard (RSPS) which was determined to be a drill/exercise performance (DEP) - performance indicator (PI) opportunity failure. The finding was also determined to be an apparent violation associated with emergency preparedness planning standards 10 CFR 50.47(b)(14) and 10 CFR 50.47(b)(4), and the requirements of 10 CFR Part 50, Appendix E, Section IV.F.2.g. The finding was documented in NRC Integrated Inspection Report No. 5000280,281/2006008 issued on May 5, 2006, and was assessed under the significance determination process as a preliminary White issue (i.e., an issue of low to moderate safety significance which may require additional NRC inspection).

The cover letter to the inspection report informed Virginia Electric and Power Company (VEPCO) of the NRC's preliminary conclusion and provided VEPCO an opportunity to request a regulatory conference on this matter. In lieu of a regulatory conference, VEPCO provided a written response dated June 6, 2006.

In its written response, VEPCO disagreed with the NRC's conclusion that the issue resulted in a preliminary White finding. VEPCO contended that the NRC's determination did not fully consider the information available regarding the Site Area Emergency (SAE) classification made by drill participants during the exercise or subsequent deliberations that formed VEPCO's basis for its exercise critique conclusions. VEPCO also noted its differing view regarding compliance with the applicable regulatory requirements that were noted in the NRC's inspection report.

After carefully considering the information developed during the inspection and the information provided in VEPCO's response of June 6, 2006, the NRC has concluded that the final inspection finding is appropriately characterized as White in the Emergency Preparedness cornerstone. The NRC's response to the points made by VEPCO and the bases for our conclusions are provided in an attachment to this letter. In summary, the NRC concluded that VEPCO's SAE event classification during the exercise was an inaccurate classification. VEPCO's critique failed to identify that the SAE declaration was made using EALs (indications) that were not exceeded at the time of the declaration. Based on this and in accordance with NRC Inspection Manual Chapter 0609, Appendix B, Emergency Preparedness Significance Determination Process, the NRC has concluded that the significance of the finding is appropriately characterized as White.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also has determined that VEPCO's failure to identify the above weakness during its exercise critique is a violation of 10 CFR 50.47(b)(14), 10 CFR 50.47(b)(4), and the requirements of 10 CFR Part 50, Appendix E, Section IV.F.2.g. The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 0500280,281/2006010, and the above violation is identified as VIO 0500280,281/2006010-01, White Finding Involving Failure to Identify a Weakness During an Emergency Exercise Critique Associated with an RSPS. Accordingly, Apparent Violation AV 0500280,281/2006008-01 is closed.

Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you by separate correspondence of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Significant Enforcement Actions**.

Should you have any questions regarding this letter, please contact Mr. Brian Bonser, Chief, Security and Emergency Preparedness Branch, Division of Reactor Safety, at (404)562-4653.

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket Nos. 50-280 and 50-281
License Nos. DPR-32 and DPR-37

Enclosures:
1. Notice of Violation
2. Basis for NRC's Final Significance Determination

cc w/encls:
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X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☐ NON-SENSITIVE

ADAMS: ☐ Yes ACCESSION NUMBER:________________________

OFFICE	ORA	DRS	NRR	NSIR/EP	OE		
SIGNATURE	/RA/	/HC/ FOR	/RA/	/RA/	/RA/		
NAME	CEVANS	VMCCREE	RPASCARELLI	RKAHLER	DSOLORIO		
DATE	7/13/06	7/13/06	07/14/06	07/18/06	07/20/06		
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NOTICE OF VIOLATION

Virginia Electric and Power Company
Surry Nuclear Station
Units 1 and 2

Docket Nos. 50-280 and 50-281
License Nos. DPR-32 and DPR-37
EA-06-071

During an NRC inspection completed on March 29, 2006, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.47(b)(4) requires, in part, that a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.
>
> 10 CFR 50.47(b)(14) requires, in part, that periodic exercises be conducted to evaluate major portions of emergency response capabilities and deficiencies identified as a result of exercises be corrected.
>
> 10 CFR Part 50, Appendix E, Section IV.F.2.g, requires that all training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified shall be corrected.
>
> Contrary to the above, the licensee's formal critique of an emergency preparedness exercise conducted on February 7, 2006, failed to identify weak or deficient areas. Specifically, the exercise critique failed to identify that the Station Emergency Manager's Site Area Emergency event classification was an inaccurate classification.
>
> This violation is associated with a White significance determination process finding for Units 1 and 2 in the Emergency Preparedness cornerstone.

Pursuant to the provisions of 10 CFR 2.201, Virginia Electric and Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-06-071" and should include: (1) the reason for the violation or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Enclosure 1

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 25th day of July 2006

NRC'S BASIS FOR FINAL SIGNIFICANCE DETERMINATION

The NRC's inspection report of May 5, 2006, documented the preliminary significance determination for a finding involving the failure of Surry Nuclear Station's full-scale exercise critique to identify a weakness associated with a risk-significant planning standard (RSPS) which was determined to be a drill/exercise performance (DEP) - performance indicator (PI) opportunity failure. The finding was also determined to be an apparent violation associated with emergency preparedness planning standards 10 CFR 50.47(b)(14) and 10 CFR 50.47(b)(4), and the requirements of 10 CFR Part 50, Appendix E, Section IV.F.2.g. The finding was assessed under the significance determination process (SDP) as a preliminary White issue (i.e., an issue of low to moderate safety significance which may require additional NRC inspection).

In lieu of a regulatory conference, Virginia Electric and Power Company (VEPCO) provided a written response dated June 6, 2006. VEPCO's written response documented its disagreement with the NRC's preliminary determination that the finding rises to a level of significance of a White finding. VEPCO concluded that its drill critique correctly determined that drill personnel responded to entry criteria for classifying the event in a reasonable and conservative manner and in accordance with the Emergency Action Level (EAL) procedure in effect. To support its conclusions, VEPCO provided the following four considerations:

1. The "failure" determination reached by the NRC appears to be based on an overly narrow construct regarding the use and application of EALs for Site Area Emergency (SAE) classification. This in turn has resulted in an associated overly narrow application of the SDP.

2. A "failure" determination is not consistent with NRC regulatory action taken by the NRC in evaluations of other licensees.

3. A "failure" determination is not consistent with NRC endorsed guidance.

4. A detailed critique that does not find an event classification to be a failure, because the classification is made conservatively and is consistent with the EAL's entry criteria, is not an indication that a weakness exists in the effective implementation of the Emergency Plan. Such a discrepancy is certainly not a weakness as defined in the SDP; therefore, this issue should not meet the intent nor rise to the level in the SDP process of an actual programmatic weakness.

The NRC's response to each of the points made by the licensee is provided in the following paragraphs:

<u>Licensee Comment No. 1</u> – The "failure" determination reached by the NRC appears to be based on an overly narrow construct regarding the use and application of EALs for SAE classification. This in turn has resulted in an associated overly narrow application of the SDP.

To support its view, VEPCO noted that the NRC's inspection report made the following three inappropriate assumptions/implications after which VEPCO provided its basis for why these assumptions were inappropriate:

a. Without a second seismic event of design-basis earthquake (DBE) magnitude, the EAL was not usable.

b. The earthquake was not validated.

c. Knowledge of the 0.13g acceleration should have caused the Emergency Director to ignore other EAL entry conditions.

NRC Response to Licensee Comment No. 1 – In the NRC's view, a key issue is whether the damage to the safety-grade auxiliary building emergency ventilation system (damage to the 1-VS-F-58 A and B fans) was the result of the seismic event that occurred over an hour earlier and had been terminated. VEPCO's position is that the EAL (indication) for the Notification of an Unusual Event (NOUE) could be applied for the determination of the SAE which occurred 1 hour and 50 minutes later. The licensee used the transition from the NOUE to the Alert as support for using the initial NOUE EAL as meeting one of the SAE EALs (indications).

The NRC's position is that the earthquake was a discrete (discontinuous) event. This conclusion is supported by NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Rev. 2, which provides an earthquake as an example of a discrete (discontinuous) event. In this case, the EAL (indication) used to declare the NOUE did not exist at the time the SAE was declared.

Classification of the NOUE was based on meeting the EAL (indication) confirmed earthquake which activates the event indicator on the strong motion accelerograph. With the event indicator, the operators entered Procedure 0-AP-37.00, Seismic Event. When the data from the strong motion accelerograph was analyzed, the classification was upgraded to an Alert. The earthquake confirmation and data collection occurred at the same time, and only the analysis of the data delayed the declaration of the Alert.

After entry into Emergency Plan Implementing Procedure (EPIP)-1.01, Emergency Manager Controlling Procedure, the applicable procedures in effect included EPIP-1.02, Response to Notification of Unusual Event; EPIP-1.03, Response to Alert; EPIP-1.04, Response to Site Area Emergency; and EPIP-1.05, Response to General Emergency. The Station Emergency Director is directed to review the EAL table and determine if the current classification is correct and to return to EPIP-1.01 for escalation and de-escalation of the emergency classification as required.

EPIP-1.01, Step 1, directs the user to evaluate EALs in the following manner:

a. Determine event category using Attachment 1, Emergency Action Level Table 1 Index.

b. Review EAL tab associated with event category.

c. Use control room monitors, process computer system (PCS), and outside reports to get indications of emergency conditions listed in the EAL table.

d. Verify EAL - CURRENTLY EXCEEDED.

Each evaluation for emergency classification takes into account the classification considered, the conditions/applicability, and the existing indications for that classification at the time of the classification. If the indications for the classification are not met at that time, then the classification or change in classification cannot be made. Based on the procedures in effect and the fact that indications for an SAE were not met at the time, the NRC concluded that VEPCO's change in classification to an SAE was inaccurate.

Regarding the inspection report assumptions, two of the three statements the licensee identified as NRC assumptions are properly referenced in the following statements taken from the NRC report:

a. Without a second seismic event of DBE magnitude, the correct classification of the turbine blading failure and damage to safety-related structures and equipment would have been at the Alert level. Since the facility was already in an Alert status, no change in the emergency response level was necessary. The inspectors determined that the EAL used to make the classification by the exercise participants for SAE was an incorrect EAL classification based on the event conditions and the indications available.

The licensee's analysis of the first assumption states that ...

> This assertion implies that the only correct way to use an EAL is for a unique event that would be classifiable at the moment the event occurs. This perspective is employing an event evaluation method where all of the information is revealed at the same time; however, the evaluation of a flow of events that are revealed over time is also an appropriate method for event classification.

In response, the NRC notes that both EPIP-1.01 and NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 3, clearly state that to make an event classification, the EAL (indications) must currently be exceeded for that classification. NEI 99-02 also provides guidance on actions that must be taken if the licensee discovers an event or condition had existed that exceeded an EAL, but no classification had been made, and the EAL is no longer exceeded at the time of discovery. Based on this, the NRC concluded that the earthquake was a discrete (discontinuous) event and that the EAL (indication) used to declare the NOUE did not exist at the time the SAE was declared, approximately 1 hour and 50 minutes later.

Regarding the assumption that the earthquake was not validated, VEPCO properly referenced this assumption as indicated by the following statement from the NRC inspection report:

b. The Station Emergency Manager (SEM) assumed that a second seismic event occurred without validating the information from the control room alarms. The inspectors based the SEM assumption on hearing the SEM's statement during the exercise prior to the SAE declaration. The SEM made the statement after receiving reports that vibrations were felt coming from the floor/ground. Significant floor vibration is expected in the event of a turbine blading failure that penetrates the turbine casing. As event conditions changed that could meet emergency classification escalation criteria, the SEM should have evaluated the event category and selected the proper EAL tab associated with the event category.

The licensee's written response stated that ...

Furthermore, the EAL construction for the SAE did not require another earthquake to occur even though the SEM thought one had occurred. This action would only serve as a replication of the action that was taken to transition from the NOUE to the Alert classification.

The NRC notes that the crew had received information that no damage and no flooding in the Unit 1 and 2 turbine buildings occurred as a result of the earthquake. The report of no damage detected was consistent with the facilitator interface for mini-scenario No. 1. Procedure 0-AP-37.00, Seismic Event, had been exited, and the event indicator on the strong motion accelerograph had been reset. Both actions were completed prior to the SAE declaration.

Classification of the NOUE was based on meeting the EAL (indication) (confirmed earthquake) which activates the event indicator on the strong motion accelerograph. With the event indicator, the operators entered Procedure 0-AP-37.00, Seismic Event. When the data from the strong motion accelerograph was analyzed, the classification was upgraded to an Alert. The earthquake confirmation and data collection occurred at the same time, and only the analysis of the data delayed the declaration of the Alert.

When the turbine failure occurred, there was no earthquake that activated the event indicator on the strong motion accelerograph, and there was no safety-related system significantly degraded by the earthquake. Procedure 0-AP-37.00 was not entered, and the required EALs (indications) for L-1, Earthquake Greater than DBE Levels, were not met. NEI 99-02 states that if an event has occurred that resulted in an emergency classification where no EAL was exceeded, the incorrect classification should be considered a missed opportunity. EPIP-1.01, step 1.c, stated, "Use control room monitors, PCS, and outside reports to get indications of emergency conditions listed in the EAL table." Based on this, the NRC concluded that the SAE declaration was made using EALs (indications for L-1) that were not exceeded.

The third assumption identified in the licensee's written response of June 6, 2006, has no specific tie to the NRC inspection report that can be found.

c. The assumption that knowledge of one indication should shade or influence the use of another indication in the EAL structure; however, this is not the logic of many of the EAL classification schemes.

Licensee Procedure DNOS-0101, Nuclear Safety and Conservative Decision Making, provides at least four standards that address this concern:

- Human performance tools and group input shall be utilized to avoid inappropriate actions and unexpected responses when reaching operating decisions.

- Operators shall recognize when degraded conditions exist that could challenge plant safety or reliability.

- Information shall be gathered and analyzed from relevant sources and appropriate personnel in order to clearly define and provide options for resolution of operational concerns.

- When faced with time-critical decisions, operators:

  - Question and validate available information.

  - Utilize available alternate indications to validate information.

  - Assume the available indications are valid until proven otherwise.

  - Use all available resources, including people offsite, if necessary.

Both EPIP-1.01 and NEI 99-02 state that to make an event classification, the EAL (indications) must currently be exceeded for that classification. Each evaluation for emergency classification takes into account the classification considered, the conditions/applicability, and the existing indications for that classification. If the EALs (indications) for the classification are not met, then the classification or change in classification cannot be made.

Licensee Comment No. 2 – A "failure" determination is not consistent with NRC regulatory action taken by the NRC in evaluations of other licensees.

The licensee provided descriptions of two events which were classified as emergencies that were later found to have used entry criteria to classify an event that led to an overly conservative classification.

NRC Response to Licensee Comment No. 2 – Based on the NRC's followup review of the two events in question and the information provided by VEPCO, the NRC has concluded that regulatory action in these cases was in accordance with Inspection Manual Chapter (IMC) 0609, Appendix B, Emergency Preparedness Significance Determination Process. The information provided by VEPCO was not sufficient to warrant a reconsideration of the NRC's conclusions in these two previous matters. Should additional or new information become available, the NRC would be amenable to reconsideration of these matters within the context of the criteria provided in NRC ICM 0609, Attachment 2.

The NRC notes that the conclusions in the instant VEPCO matter are consistent with a recent enforcement action involving a White finding and associated NOV that was issued to another utility on December 16, 2005 (EA-05-192, ADAMS Accession No. ML053530049).

Licensee Comment No. 3 – A "failure" determination is not consistent with NRC endorsed guidance.

Based on NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 3, the licensee stated that they reevaluated indications provided to the participants and the method of interpretation and implementation of the EALs that was used. The determination of PI opportunity success was based on the fact that the indications provided were usable as supportive of an escalation to an SAE classification in this scenario.

NRC Response to Licensee Comment No. 3 – NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 3, states that ...

> During drill performance, the emergency response organization may not always classify an event exactly the way that the scenario specifies. This could be due to conservative decision making, Emergency Director judgment call, or a simulator driven scenario that has the potential for multiple "forks." Situations can arise in which assessment of classification opportunities is subjective due to deviation from the expected scenario path. In such cases, evaluators should document the rationale supporting their decision for eventual NRC inspection. Evaluators must determine if the classification was appropriate to the event as presented to the participants and in accordance with the approved emergency plan and implementing procedures.

The NRC observed the deviation during the graded exercise and was knowledgeable of the events leading to the deviation. The NRC reviewed the deviation from the expected scenario path and the licensee's rationale used to reach their decision. Additional information provided by the licensee was reviewed and incorporated into the inspection report. The NRC disagrees with the licensee's conclusion that the classification was appropriate and in accordance with the approved emergency plan and implementing procedures, as noted previously.

Licensee Comment No. 4 – *A detailed critique that does not find an event classification to be a failure, because the classification is made conservatively and is consistent with the EAL's entry criteria, is not an indication that a weakness exists in the effective implementation of the Emergency Plan. Such a discrepancy is certainly not a weakness as defined in the SDP; therefore, this issue should not meet the intent nor rise to the level in the SDP process of an actual programmatic weakness.*

NRC Response to Licensee Comment No. 4 – IMC 0609 states, in part, that ...

> As applied to emergency preparedness, a weakness is a level of performance demonstrated during a drill or exercise that could have precluded effective implementation of the Emergency Plan in the event of an actual emergency. Weaknesses are not confined to performance problems that result in a loss of planning standard (PS) function. For example, an inaccurate or untimely classification, notification, or Protective Action Recommendation (PAR) development is a weakness associated with an RSPS (i.e., a DEP PI opportunity failure) ... The NRC staff expects licensees to identify and critique this performance problem as a weakness associated with an RSPS. Failure to correct a weakness should be analyzed against the compliance criteria in planning standard 10 CFR 50.47(b)(14) and the Emergency Plan. A failure to identify and/or correct a weakness associated with an RSPS function represents a loss of PS function 10 CFR 50.47(b)(14) for which Section 5.0 of IMC 0609, Appendix B, provides guidance regarding the correction of weaknesses. For purposes of this SDP, this includes a deficiency, as the term is used in planning standard 10CFR 50.47(b)(14) and Section IV.F.2.g of Appendix E to 10 CFR Part 50.

If the licensee's critique fails to identify an inaccurate or untimely classification, notification, or PAR development (i.e., a DEP PI opportunity failure), it is considered a loss of PS function (white finding). This is because the licensee's capability to observe and evaluate the process associated with an RSPS is questionable.

It is the NRC's conclusion that the SAE event classification was an inaccurate classification. The licensee's critique failed to identify that the SAE declaration was made using EALs (indications) that were not exceeded at the time of the declaration. This determination is consistent with IMC 0609.

The response of the offsite response organizations (ORO) to a radiological emergency is highly dependent on the quality of the information that the licensee provides the OROs in emergency classification, PARs, and notifications. Conservative decision-making is highly encouraged but not when the decision may result in the public being placed at unnecessary risk due to over-conservative classifications or PARs. As such, the NRC expects licensee emergency classifications, PARs, and notifications to be accurate and timely. NEI 99-02 defines accurate as: "Classification and PARs appropriate to the event as specified by the approved plan and implementing procedures ..."

The exercise scenario provided no valid bases for plant personnel to conclude that the turbine failure and the consequential safety-grade ventilation system damage was the result of the seismic event which had occurred and terminated over an hour earlier. This conclusion is confirmed by the facts that the scenario developers did not envision the SAE being called under EAL L-1 and that the operators exited the seismic abnormal procedure before the SAE was declared. The turbine failure was not a progression from the earlier seismic event but rather a new discrete event. As such, the NRC continues to believe that the SAE classification was inaccurate and, therefore, a PI opportunity failure, a deficiency that was not identified in the critique.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 17, 2006

NRC INFORMATION NOTICE 2006-18: SIGNIFICANT LOSS OF SAFETY-RELATED ELECTRICAL POWER AT FORSMARK, UNIT 1, IN SWEDEN

ADDRESSEES

All holders of operating licensees for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to alert addressees of a significant incident that occurred at the Forsmark Nuclear Power Station, Unit 1 (Forsmark-1), in Sweden involving the loss of several safety-related electrical busses. It is expected that addressees will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Forsmark-1 is a 1020 Megawatt electric boiling-water reactor designed by ASEA-Atom which began commercial operation in 1980. On July 25, 2006, a significant incident occurred at Forsmark-1 in which, through a complex series of events, a short circuit in the switchyard led to the loss of two out of the four trains of safety-related alternating current (AC) and direct current (DC) power due to a common mode failure. This event is significant in that it could have caused the common mode failure in all four trains and therefore, could have resulted in the loss of all four trains of safety-related AC and DC power.

The event began when an arc and a two phase short circuit occurred when a breaker was opened in the 400 kV switchyard to support maintenance. The electrical transient dropped the voltage to about 30 percent of nominal voltage and the unit was disconnected from the grid. In addition, the electrical transient caused a brief increase in voltage on the main generator. This sudden overvoltage caused two of the four electrical inverters to fail and consequently disabled two emergency diesel generators (EDGs) from powering the corresponding buses as expected. The remaining two EDGs were able to start automatically and provide power to the batteries.

The reactor successfully scrammed and all control rods inserted. The control room staff were challenged by the absence of control room indications associated with the two trains of power supply that were lost. The event was further complicated by the actuation of the containment spray and emergency cooling systems. After restoring power, the operators were able to secure the containment spray and emergency cooling systems.

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Investigation is currently in progress by the licensee for Forsmark-1 regarding the cause of the switchyard electrical transient and its resulting complex effects on the plant. The Swedish Nuclear Power Inspectorate categorized the event under the International Nuclear Event Scale (INES) as a level 2 event.

DISCUSSION

Abnormal overvoltage conditions from the grid or other sources could lead to failures of critical electrical and electronic components including electrical inverters unless they are protected. The sensitivity and the response of the components to overvoltage condition could vary depending upon the characteristics of the electrical transient and the source of the overvoltage. Capability to identify such potential vulnerabilities and preparations to implement compensatory actions could reduce the challenges for the control room operators.

CONTACT

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

July 31, 2006

NRC INFORMATION NOTICE 2006-17: RECENT OPERATING EXPERIENCE OF SERVICE WATER SYSTEMS DUE TO EXTERNAL CONDITIONS

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of operating experience within the past few years affecting the operability of the service water system at several nuclear power plants. The NRC expects that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

During 2004 through 2005, 15 events occurred related to blockages in service water systems. These events were primarily self-revealing. The various blocking agents included silt, sand, small rocks, grass or weeds, frazil ice, and small aquatic fauna, such as fish. All these events were of low safety significance but illustrate the susceptibility of the safety-significant service water system. For instance, in September 2005, NRC inspectors identified a condition at Fort Calhoun that allowed small rocks to regularly enter the raw water system, contribute to tripping of a pump and strainer motors, and interfere with traveling screen operation (NRC Inspection Report 50-285/2005-11, Agencywide Documents Access and Management System (ADAMS) Accession No. ML052920543). In June 2005, NRC inspectors found a portion of a service water accumulator outlet line at Salem to be nearly full of silt (NRC Inspection Report 50-272/2005-03, ADAMS Accession No. ML052090344).

Salem - Hope Creek Nuclear Power Plants

On December 2, 2004, crude oil was found leaking from a ship (Athos I) on the Delaware River upstream of the Salem and Hope Creek Generating Stations. To mitigate the potential for oil intrusion into the cooling water systems, the licensee placed booms around the intake structures at both stations. The booms are effective at controlling oil that is at or near the

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surface; however, the effectiveness of the booms was lessened because the spilled oil was "heavy" crude and was suspended at varying depths in the river. On December 3, 2004, the licensee commenced shutdown of both Salem units due to the conditions on the river. There were no issues associated with the shutdowns. Hope Creek was already shut down for a refueling outage. The licensee restarted both Salem units after review of heat exchanger performance and monitoring of the oil spill.

Cooper Nuclear Station

On November 20, 2004, the service water system was clogged with sediment, resulting in an unexpected pressure drop in both loops of service water, high differential pressure alarms on both strainers, and isolation of the nonessential service water loads. Both trains exceeded the differential pressure operability limit of 15 psid. Backwash automatically initiated and successfully cleaned the Loop A strainer, but the analogous action for Loop B did not succeed in cleaning the strainer. Operators opened the strainer bypass valve to restore service water flow and subsequently cleaned both strainers.

On October 20, 2005, while preparing for online maintenance of the service water system, operators started a fourth service water pump and received high differential pressure alarms on both Loops A and B. The automatic backwash did not sufficiently decrease the differential pressure, and operators bypassed the strainer. Following these actions, the service water system header pressures returned to normal. During the event, operators declared both loops of service water inoperable. Both loops exceeded the strainer differential pressure structural integrity limit of 15 psid. The high differential pressure across the strainers was the result of debris (small rocks and sediment) introduced by the start of the fourth pump. With both loops of service water inoperable, operators declared both emergency diesel generators inoperable.

In 2005, the NRC Region IV office organized a special inspection based on the repetitive nature of this type of event (NRC Inspection Report 50-298/2005-15, ADAMS Accession No. ML061160027).

Watts Bar Nuclear Plant

On November 22, 2004, while performing a manual valve exercising procedure, the licensee identified that a centrifugal charging pump backup cooling line from the essential raw cooling water system was completely blocked with silt. Approximately 2.5 gallons of muddy paste passed through the 1-inch drain valve before the valve became blocked. The line had to be cleared mechanically. This line is significant in that this is the only high head pump with a backup source of cooling water (NRC Inspection Report 50-390, 391/2004-05, ADAMS Accession No. ML050280344).

DISCUSSION

Cooper Nuclear Station

In both events, for a few minutes service water flow was lost to the nonessential header and greatly reduced to the essential headers. In each case, the successful Loop A automatic backwash precluded the need for a manual scram, which would have been required if the loss of turbine equipment cooling water had been prolonged. In each event, the Loop B filtering function was overwhelmed by the inrush of sediment. The Loop B automatic backwash function failed due to the lack of downstream pressure, which provides the motive force for the backwashing operation. The licensee believes that the contributing external factor was the low level of the Missouri River, the source of the service water system. Both of these events occurred during autumn, following the navigation season. A weir wall is installed in the river in front of the intake structure. The low river level caused an increased portion of the water that flows into the intake structure to go around (rather than over) the weir wall and jet into the service water bay. This circuitous flow entrained more sand due to the high flow and deposited it in the intake structure near the service water pump intakes in the low-flow areas.

At the time of the October 2005 event, the licensee had not completed its actions to modify the setpoint for automatic backwash of the strainer, alter the strainer intermittent backwash frequency, modify the strainer differential pressure alarm setpoint, and implement weir-wall and traveling-screen modifications.

NRC inspectors noted that the licensee had not performed certain actions committed to in its response to NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," specifically to periodically monitor silt levels and to periodically examine the intake structure basin for silt, debris, and deterioration (including corrosion), using divers or by dewatering the intake structure bay. At the time of the event, the licensee had not examined the intake structure bay to assess its condition.

Watts Bar Nuclear Plant

The licensee generated 13 problem evaluation reports from early 2002 through late 2005 for blockages identified in raw cooling water lines. The licensee identified silt accumulation in portions of systems providing raw cooling water for both essential and nonessential purposes and for high pressure water for fire protection. These accumulations were identified in both stagnant and active cooling water lines, typically in system low points and in piping with low water velocity. In 1999 and 2002, clam accumulations resulted from missed biocide treatments. The licensee implemented periodic ultrasonic testing and flushing to identify and minimize blockages due to silt and clam accumulations. The initial frequency of ultrasonic testing was every 6 months, later shortened to every 3 months. However, the licensee determined that this program did not cover all susceptible lines and components.

The centrifugal charging pump backup cooling line was not included in the ultrasonic testing monitoring program. In 2000, a maintenance rule panel review left the flushing frequency for this line at 18 months, not recognizing the consequences of silt accumulation. This conclusion was consistent with the general site perception that silt accumulation was not a significant problem. The blockage was found by means of an 18-month manual valve test. Most other lines were being flushed or tested every 3 months. This issue resulted in a White finding in the NRC's Significance Determination Process.

Raw water systems draw from a section of the Tennessee River downstream of the Watts Bar dam. The suspended solids count in the river water increases after periods of heavy rains upstream. The suspended solids are transported into the affected systems where they settle at points with low fluid velocities.

The licensee's corrective actions for the violation included increasing the frequency of ultrasonic testing, developing higher velocity flush procedures, and modifying systems to improve flushing. Lessons learned included the following observations:

- Silt accumulation in smaller diameter lines may not flush as readily as in larger diameter lines.
- Silt accumulates in stagnant lines off the main headers.
- Lines with a vertical drop off the main headers are more susceptible to silt accumulation than lines with horizontal legs off the main headers.

RELEVANT GENERIC COMMUNICATIONS

NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment"

NRC GL 89-13 lists the following five recommendations for licensees:

- Significantly reduce the incidence of flow blockage problems resulting from biofouling.
- Conduct a test program to verify the heat transfer capability of all safety-related heat exchangers cooled by service water, including initial and periodic retesting.
- Ensure by a routine inspection and maintenance program for open-cycle service water system piping and components that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water.
- Confirm that the service water system will perform its intended function in accordance with the licensing basis for the plant.
- Confirm that maintenance practices, operating and emergency procedures, and training that involves the service water system are adequate for ensuring that safety-related equipment cooled by the service water system will function as intended and that operators of this equipment will perform effectively.

NRC Information Notice 2004-07: "Plugging of Safety Injection Pump Lubrication Oil Coolers with Lakeweed"

NRC IN 2004-07 also discusses operating experience related to service water system susceptibilities due to external events.

CONCLUSION

The above events involve instances in which sediment and debris has blocked flow in one or more service water lines. A number of the events described above involved the failure to take adequate and timely corrective actions that could have prevented the event from occurring. Often there were multiple previous occurrences that could have alerted licensees to take more aggressive or broader corrective actions.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July 27, 2006

INFORMATION NOTICE 2006-15: VIBRATION-INDUCED DEGRADATION AND FAILURE OF SAFETY-RELATED VALVES

ADDRESSEES

All holders of operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to alert addressees of vibration-induced degradation and failure of valves supplied by Fisher Controls and other manufacturers. The agency expects that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, the suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

During a plant startup in October 2003, Vogtle Electric Generating Plant, Unit 1, experienced a failure of an auxiliary feedwater (AFW) discharge control valve. The valve was a 4-inch, model SS-120, manufactured by Fisher Controls. Although the valve indicated full open, operators noted that AFW loop flow was reduced and did not change during valve throttling. The valve degradation was attributed to the flow-induced metal fatigue failure of a cotter pin designed to secure the pilot plug assembly retaining nut to the valve stem. Consequently, the retaining nut backed off completely, releasing the pilot plug spacer and a washer from the pilot plug, allowing them to be transported downstream and block flow through a restricting orifice. A similar failure of an AFW discharge control valve occurred in 1989 at Vogtle.

The valve vendor issued an advisory regarding this failure mechanism in 1988. In this advisory, the vendor stated that failures of Fisher Controls type AP, EP, EWP, and SS-120 valves may occur and recommended that all valves affected by the advisory be disassembled to inspect the main-plug/pilot-plug restraining nut assembly. The assembly is held together by a large nut which is restrained from turning by either a star lockwasher with bend-up tabs, or a single cotter pin design. The vendor stated that the hex nut may unscrew because of improper installation of either type locking mechanism. Specifically, reuse of the star lockwasher has resulted in fatigue and subsequent breaking of the tabs, and the cotter pin design has failed from improper replacement or reuse which has allowed the pin to vibrate and fail through fatigue.

ML061790443

CONTACT

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

In response to this failure event, the licensee performed visual inspections of 15 similar valves and identified that all of the motor-driven AFW discharge control valves were missing cotter pins. The cotter pins associated with the turbine-driven AFW discharge control valves appeared unaffected, probably because of the much lower operational flow time. The licensee repaired the valves by staking the threads on the valve stem against the retaining nut, instead of securing the nuts with cotter pins.

DISCUSSION

Over the years, nuclear power plants have experienced vibration-induced degradation of plant equipment during operation at the original licensed power and under power uprate conditions. The NRC has issued several INs on this subject, including NRC IN 83-70, "Vibration-Induced Valve Failures," dated October 25, 1983 (http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1983/in83070.html), to alert nuclear power plant licensees of instances of valve failures and system inoperability that occurred as a result of normal operational vibration.

In January 2004, NRC IN 2002-26, Supplement 2, "Additional Flow-Induced Vibration Failures After a Recent Power Uprate," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML040080392) was issuedwhich described that increased steam and feedwater flow can increase the vibration of plant equipment, including valves and valve actuators. The higher vibration levels can impact the appropriate inspection intervals for some plant components.

San Onofre Nuclear Generating Station, Units 1 and 2, experienced degradation of butterfly valves in 2003, as discussed in NRC IN 2005-23, "Vibration-Induced Degradation of Butterfly Valves," dated August 1, 2005 (ADAMS Accession No. ML051740299). The failures resulted from lost taper pins used to connect the valve disc to the valve stem. These valves were manufactured by Fisher Controls. Problems have been attributed to failed taper pin connectors in butterfly valves supplied by other manufacturers. In 1989, Turkey Point Nuclear Plant, Unit 4, lost taper pins in a 36-inch intake cooling water isolation valve manufactured by the Henry Pratt Company. In 2003, Davis-Besse Nuclear Power Station, Unit 1, lost taper pins in a 10-inch decay heat removal cooler valve with the brand name Valtek marketed by the Flowserve Corporation.

In June 2005, the licensee at Hope Creek Generating Station shut down the unit and entered into its emergency plan because it exceeded limits for unidentified leakage inside primary containment. This event was discussed in "Hope Creek Nuclear Generating Station - NRC Integrated Inspection Report 05000354/2005005 and Exercise of Enforcement Discretion," dated January 26, 2006 (ADAMS Accession No. ML060270171). The licensee identified an approximately 285-degree circumferential crack in the position-indicating tube for the "A" residual heat removal shutdown cooling return testable check valve. This through-wall leak was caused by vibration of the attraction sleeve (located at the end of the actuator rod), in the presence of the switch magnetic force, resulting in the attraction sleeve fretting and wearing through the position-indicating tube. Licensee corrective actions included modifying both the "A" and "B" train check valves by removing the position indicator tubes. Six additional check valves that use the same position indicator tube underwent ultrasonic testing, which revealed no similar wear indications.

In summary, operating experience associated with vibration-induced valve degradation shows that certain valve sub-components (such as yoke-to-bonnet hold-down studs and nuts, stem-to-disc connectors, valve stem clamp setscrews) may be more susceptible to failure. Changes to system flow characteristics and vibrational harmonics may serve as indicators that further evaluation of these effects on system components is needed. Initiatives to preclude valve failures may include identifying components that could be subjected to vibration-induced stress and wear, fully understanding the long-term effects that vibration-induced stress may have on these components (including sub-components that may be prone to early failure), and thoroughly evaluating and inspecting components on a schedule consistent with the overall risk significance associated with a failure.

CONTACTS

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below.

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Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

OMB Control No.: 3150-0011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 24, 2006

NRC REGULATORY ISSUE SUMMARY 2006-17 NRC STAFF POSITION ON THE REQUIREMENTS OF 10 CFR 50.36, "TECHNICAL SPECIFICATIONS," REGARDING LIMITING SAFETY SYSTEM SETTINGS DURING PERIODIC TESTING AND CALIBRATION OF INSTRUMENT CHANNELS

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) on the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical Specifications," with respect to limiting safety system settings (LSSSs) assessed during periodic testing and calibration of instrumentation. This RIS discusses issues that could occur during testing of LSSSs and which, therefore, may have an adverse effect on equipment operability. This RIS also presents an approach, found acceptable to the NRC staff, for addressing these issues for use in licensing actions that require prior NRC staff approval. Methods and approaches different from those in this RIS may also be acceptable to the NRC staff. The approach presented in this RIS is intended for use by licensees in developing content for license amendment applications. This RIS requires no action or written response from addressees.

BACKGROUND INFORMATION

Over the past several years during public meetings and as part of license amendments, the NRC staff has been discussing its perspective on the efficacy of using technical specification (TS) allowable values to meet the requirements of 10 CFR 50.36(c)(1)(ii)(A) for LSSSs. The industry Technical Specifications Task Force (TSTF) submitted its recommendations for standard technical specifications (STS) changes as TSTF-493, Revision 0, "Clarify Application of Setpoint Methodology for LSSS Functions" on January 27, 2006, for NRC staff review. TSTF-493 was intended to address seven concepts proposed by industry for developing model content for TSs and TSs Bases for LSSSs instrumentation functions. TSTF-493 was provided as a readily adoptable approach to ensure that the TSs conform to the requirements of 10 CFR 50.36. The background information that follows cites regulations, identifies guidance documents, and

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defines terms important to understanding this RIS. The information also provides discussion of the relationship between the STSs and TSTF-493, Revision 0 to set the framework for the Summary of the Issue that follows.

Regulations and guidance documents

The requirements for plant TSs are stated in 10 CFR 50.36, "Technical Specifications":

- Section 50.36(a) states: "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

- Section 50.36(c)(1)(i)(A) states: "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

- Section 50.36(c)(1)(ii)(A) states: "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor."

- Section 50.36(c)(3) states: "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The RG endorses Part I of ISA -S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation," subject to NRC staff clarifications. Part I defines a framework for ensuring that setpoints for nuclear safety-related instrumentation are established and maintained within specified limits. The RG does not address or endorse Part II of ISA-S67.04-1994, "Methodologies for the Determination of Setpoints for the Nuclear Safety-Related Instrumentation." Part II provides recommended practices and guidance for implementing Part I.

For the purpose of this RIS, the NRC staff is providing the following definitions of *limiting trip setpoint, nominal trip setpoint,* and *allowable value*:

Limiting trip setpoint (LSP)
The LSP is the limiting setting for the channel trip setpoint (TSP) considering all credible instrument errors associated with the instrument channel.

The LSP is the limiting value to which the channel must be reset at the conclusion of periodic testing to ensure the safety limit (SL) will not be exceeded if a design basis event occurs before the next periodic surveillance or calibration.

Nominal trip setpoint (NSP)
The NSP is the TSP value selected by the licensee for plant operations.

The NSP must be equal to or more conservative than the LSP.

Allowable value (AV)
An AV is a limiting value of an instrument's as-found trip setting used during surveillances.

Technical Specifications: Limiting Trip Setpoints and Resetting Requirements

Many licensees use an AV as as-found LSSSs. This means that licensees perform periodic surveillances and use the AV to verify that the SL is protected and that the channel is operable. If the AV is exceeded during a surveillance, the instrument is declared inoperable because there is not adequate assurance that the instrument will perform its safety function, and appropriate TS-required action must be taken.

10 CFR 50.36(c)(1)(ii)(A) requires that the TSs include LSSSs for variables that have significant safety functions. For variables on which a SL has been placed, the LSSS must be chosen to initiate automatic protective action to correct abnormal situations before the SL is exceeded. Many licensees have TSs that specify an AV as the LSSSs. During periodic surveillances, no actions are required by TSs (e.g., resetting) as long as the results indicate that the as-found TSP is conservative with respect to the AV. Many licensees rely on administrative controls to reset the instrument TSP to the LSP or to a value more conservative than LSP at the conclusion of periodic testing, but these controls are given in documents other than the TSs. However, if the instrument TSP is not left at a value that is conservative with respect to the LSP, then there may not be assurance that the SL will be protected until the next periodic surveillance because instrument drift and other changes in setpoint can occur. These uncertainties are accounted for in the calculation of the LSP. It is the NRC staff's position that the LSP protects the SL.

Technical Specifications: Automatic Safety Systems Function to Protect the SL

In addition, 10 CFR 50.36(c)(1)(ii)(A) requires a licensee to take appropriate action if it is determined that the automatic safety system does not function as required to protect the SL. If the channel is set to a NSP that is more conservative than the LSP then abnormally large changes in the setpoint have to occur between surveillance test intervals to indicate the channel is malfunctioning. Such setpoint changes may not exceed the AV because of the added conservatism between the LSP and the NSP. Under these conditions, operators consulting the TSs might conclude that the instrument is operable because the as-found TSP is more conservative than the AV, even though the instrument is not functioning as predicted by the instrument setpoint methodology and may not be capable of protecting the SL.

As one measure of instrument operability, the NRC staff expects licensees to verify during testing or calibration that the change in the measured TSP since the last test or calibration is within predefined limits (double-sided acceptance criteria band) and to take appropriate actions if the change is outside these limits. The acceptance criteria band should be derived from the licensee's setpoint methodology, including use of generic or plant-specific data. If the as-found TSP exceeds the AV in TSs the channel is inoperable and the associated action requirements are followed. If the change in the measured TSP exceeds the predefined limits but the measured TSP is conservative with respect to the AV, and the licensee determines during the surveillance that the instrument channel is functioning as expected and can reset the channel to within the setting tolerance (amount by which as-left setting value is permitted to differ from NSP) of the NSP, then the licensee may restore the channel to service and the condition is entered into the licensee's corrective action program for further evaluation. However, if during the surveillance the change in the measured TSP exceeds the predefined limits and the licensee cannot determine that the instrument channel is functioning as required, then the instrument is declared inoperable and the associated TS actions are followed. It is NRC staff's position that verifying that the as-found TSP is within the acceptance band limits during test or calibration is part of the determination that an instrument is functioning as required.

10 CFR 50.36(c)(1)(ii)(A) also contains requirements for a general class of LSSSs; LSSSs related to variables having significant safety functions but which do not protect SLs. All plant operating licenses have TSs for LSSSs that are not related to SLs. For these LSSSs, 10 CFR 50.36(c)(1)(ii)(A) also requires that a licensee take appropriate action if it is determined that the automatic safety system does not function as required. Additionally, 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires safety-related structures, systems, and components must also perform satisfactorily in service, i.e, the settings must initiate automatic protective actions consistent with the design basis. Following surveillance testing, resetting the TSP to within the setting tolerance of the LSP or to a value more conservative than the LSP would ensure that LSSSs for instrument functions not related to SLs perform their specified safety functions. Additionally, when evaluating the as-found TSP, operability should be determined based on the plant-specific setpoint methodology, (including consideration of the expected uncertainties in the instrument setpoint determination) to ensure that automatic protective devices will perform their specified safety function. The NRC staff recognizes that other methods and approaches different from those described above may also be acceptable and would be evaluated on a plant-specific basis.

SUMMARY OF THE ISSUE

Standard Technical Specifications

The TSTF-493, Revision 0 traveler submitted by industry addressed many of the 10 CFR 50.36 requirements identified above. However, although the TSTF discussed a plant-specific process for identifying the LSSSs instrument functions, it did not provide a list of functions that would resolve the issues for most plants. The NRC staff believes that a generic list of functions is needed in the final TSTF to avoid significant resources being expended by both industry and NRC as part of plant-specific reviews.

In accordance with 10 CFR 50.36(c)(1)(ii)(A), the following guidance is provided for identifying a list of functions to be included in TS as the subset of LSSSs specified for variables on which SLs have been placed. The SLs are those limits defined in STS Section 2.1.1, Reactor Core SLs and 2.1.2, Reactor Coolant System Pressure SLs. This subset includes automatic protective devices in TSs for specified variables on which SLs have been placed that: (1) initiate a reactor trip; or (2) actuate safety systems. As such these variables provide protection against violating reactor core safety limits, or reactor coolant system pressure safety limits. The NRC staff notes that these generic criteria represent one method the NRC staff would find acceptable for identifying LSSSs in its reviews of plant-specific license amendments. If licensees make submittals which do not follow this guidance, they should provide a plant-specific analysis to justify excluding instrument functions within these criteria.

Additionally, the TSTF did not sufficiently address the NRC staff concern with the practice of using NSPs for establishing the test acceptance criteria band for as-found instrument values. The NRC staff concern was that excessive changes in the TSP could go undetected and also that a high incidence of false detections could result from such a practice. Subsequently, the NRC staff investigated the acceptability of basing operability determinations for as-found instrument values on NSP values. The NRC staff review concluded that if specific conditions are met, then the NRC staff would find a NSP-based assessment of as-found values acceptable. Those conditions are: (1) the setting tolerance band is less than or equal to the square root of the sum of the squares of reference accuracy, measurement and test equipment, and readability uncertainties; (2) the setting tolerance is included in the total loop uncertainty, and (3) the pre-defined test acceptance criteria band for the as-found value includes either, the setting tolerance or the uncertainties associated with the setting tolerance band, but not both of these.

The NRC staff intends to incorporate this setpoint issue guidance in the final approved TSTF. The NRC staff believes that this will establish a uniform, satisfactory resolution that addresses the industry's and the staff's concerns with instrument settings, conforms to Inspection Manual Chapter Part 9900 guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded and Nonconforming Conditions Adverse to Quality or Safety" (RIS 2005-20), and ensure compliance with 10 CFR 50.36. The NRC staff intends to issue TSTF-493 as a consolidated line item improvement process (CLIIP). The CLIIP package will include a model application and safety evaluation to support using TSTF-493 for plant-specific license amendment applications.

The NRC staff believes that for current plant operation, addressing these instrument setpoint issues is not an immediate safety issue since most plant procedures require reset of instruments. In the case where an instrument channel has deviated from its trip setpoint by a small amount a reactor trip and safety system actuation would still occur. Finally, diverse instrumentation for reactor trip and the actuation of safety systems exist and are expected to function. In addition, most licensees assure operability of instrument channels when they periodically compare the as-found setpoint value during periodic surveillances with a predetermined value other than the AV of the TS, and adjust the instrument channel to within a calibration tolerance band. If the trip setpoint exceeds this predetermined value, licensees take corrective actions per plant procedures.

BACKFIT DISCUSSION

This RIS presents generic criteria that represents one method the NRC staff would find acceptable for identifying LSSSs in its reviews of plant-specific license amendments. This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because it is informational and does not depart from current regulatory requirements and practices. NRC intends to work with the Nuclear Energy Institute, industry representatives, members of the public, and other stakeholders in developing final guidance and revising related guidance documents.

CONGRESSIONAL REVIEW ACT

The NRC has determined that this action is a rule subject to the Congressional Review Act. Office of Management and Budget (OMB) has determined this is a minor rule.

PAPERWORK REDUCTION ACT STATEMENT

This RIS contains information collection requirements that are subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the OMB, approval number 3150-0011, which expires February 28, 2007.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

CONTACT

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF ENFORCEMENT
WASHINGTON, DC 20555-0001

July 31, 2006

NRC REGULATORY ISSUE SUMMARY 2006-13 INFORMATION ON THE CHANGES MADE TO THE REACTOR OVERSIGHT PROCESS TO MORE FULLY ADDRESS SAFETY CULTURE

ADDRESSEES

All holders of operating licenses for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issues summary (RIS) to provide information to addressees and their contractors regarding changes made to the Reactor Oversight Process (ROP) to more fully address safety culture. No specific action or written response is required.

BACKGROUND INFORMATION

The staff submitted to the Commission, SECY-04-0111, "Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture," dated July 1, 2004. This paper sought Commission direction with regard to the development of possible options for enhancing oversight of safety conscious work environment and safety culture. The paper noted that a weak safety culture was identified as a root cause of the reactor vessel head degradation at the Davis-Besse nuclear power plant. The NRC's Davis-Besse Lessons Learned Task Force report recommended that the staff review NRC inspections and plant assessment processes to determine whether sufficient processes are in place to identify and appropriately disposition the types of problems experienced at Davis-Besse. On August 30, 2004, the Commission provided direction in a staff requirements memorandum (SRM) on SECY–04-0111 that included the following:

- Enhance the ROP treatment of cross-cutting issues to more fully address safety culture.
- Continue to monitor industry efforts to assess safety culture.
- Include, as part of the enhanced inspection activities for plants in the degraded cornerstone column (referred to as Column 3) of the ROP action matrix, a determination of the need for a specific evaluation of the licensee's safety culture and develop a process for making the determination and conducting the evaluation.
- Continue to monitor developments by foreign regulators.

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The staff submitted to the Commission, SECY-05-0187, "Status of Safety Culture Initiatives and Schedule for Near Term Deliverables," dated October 19, 2005. This paper updated the Commission on the staff's plans and activities to enhance the agency's oversight of operating reactors to more fully address safety culture. The Commission provided direction in an SRM on SECY-05-0187, dated December 21, 2005, that included the following:

- Continue to interact with external stakeholders and build from enhancements already made to the ROP in response to the Davis-Besse Lessons Learned Task Force.
- Develop a process for determining if an evaluation of safety culture is warranted when a plant falls into the degraded cornerstone column of the ROP action matrix.
- Document significant changes to the ROP addressing safety culture in the ROP guidance documents and/or basis documentation.
- Ensure that the resulting modifications to the ROP are consistent with the regulatory principles that guided the development of the ROP.

Following receipt of SRM/SECY-05-0187, the staff held frequent public meetings with external stakeholders and, with the full participation of these stakeholders, developed an approach to enhance the ROP to more fully address safety culture. This resulted in modifications to selected inspection manual chapters (IMCs) and inspection procedures (IPs).

The staff submitted to the Commission, SECY-06-0122, "Safety Culture Initiative Activities to Enhance the Reactor Oversight Process and Outcomes of the Initiative," dated May 24, 2006, which described the status of the staff's activities and plans to enhance the ROP to more fully address safety culture. The staff implemented the changes to the ROP on July 1, 2006.

SUMMARY OF THE ISSUE

Discussion

During the November and December 2005 public meetings, the staff, with the full participation of external stakeholders, used a systematic approach to identify proposed changes to the ROP to more fully address safety culture. As a result of these meetings, the NRC and stakeholders reached alignment regarding the following:

- the definition of safety culture[1]
- those attributes or elements that are important to safety culture (i.e., safety culture components)
- needed enhancements to more fully address safety culture
- proposed changes to the ROP based on the identified needed enhancements

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[1] The NRC adopted the International Atomic Energy Agency's International Nuclear Safety Advisory Group's (INSAG) definition of safety culture provided in Safety Series No. 75-INSAG-4, "Safety Culture," issued 1991, as "that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear safety issues receive the attention warranted by their significance."

At subsequent public meetings, the staff and stakeholders discussed the details of the proposed changes and descriptions of the safety culture components. As a result of stakeholder feedback, the staff eliminated certain components and revised others, as appropriate, to provide terminology similar to that used by the industry, thereby supporting a common understanding of the safety culture components. The NRC made the draft IPs and IMCs reflecting changes to incorporate safety culture features available to stakeholders through the safety culture web page. The staff considered stakeholder recommendations and suggestions in finalizing the IPs and IMCs.

The changes to the ROP are within the ROP framework and are consistent with the regulatory principles that guided the development of the ROP. Therefore, the agency's oversight activities and their outcomes remain mostly transparent, understandable, objective, predictable, risk informed, and performance based.

The NRC intends the changes to the ROP to achieve the following:

- Provide better opportunities for the NRC staff to consider safety culture weaknesses and to encourage licensees to take appropriate actions before significant performance degradation occurs.
- Provide ~~the NRC~~ staff with a process to determine the need to specifically evaluate a licensee's safety culture after performance problems have resulted in the placement of a licensee in the degraded cornerstone column of the action matrix.
- Provide the NRC staff with a structured process to evaluate the licensee's safety culture assessment and to independently conduct a safety culture assessment for a licensee in the multiple/repetitive degraded cornerstone column of the action matrix.

Key Features of the Modified ROP

The ROP, as modified, continues to provide a graded approach to plant performance issues so that the regulatory response increases as performance degrades and licensees move to the right in the ROP action matrix. The key features of the revised process include the following:

- Inspector development of findings and the assessment of performance deficiencies for cross-cutting aspects are consistent with current practice.
- The staff revised the existing cross-cutting areas of human performance, problem identification and resolution, and safety conscious work environment to incorporate components that are important to safety culture.
- The staff revised IMC 0612, "Power Reactor Inspection Reports," to reference IMC 0305, "Operating Reactor Assessment Program," to ensure that, when the NRC identifies findings with cross-cutting aspects, the agency uses language that parallels the descriptions of the cross-cutting area components in IMC 0305.
- The staff revised IP 71152, "Identification and Resolution of Problems," to modify the existing guidance for inspectors to assess the effectiveness of the corrective action program, the use of operating experience information, and the results of independent and self-assessments. The revised procedure allows inspectors to have the option of reviewing licensee self-assessment of safety culture if performed and directs inspectors

to be aware of safety culture components when selecting samples. The staff also revised the suggested inspector questions in Appendix 1 to better assess the licensee's safety conscious work environment.

- The NRC revised the event response procedures in IP 71153, "Event Follow-up," IP 93812, "Special Inspection," and IP 93800, "Augmented Inspection Team," to direct inspection teams to consider contributing causes related to the safety culture components as part of their efforts to fully understand the circumstances surrounding an event and its probable causes.

- For performance deficiencies that appear to have a safety conscious work environment aspect as a contributor, the staff has provided additional guidance to inspectors on inspecting and documenting these issues. Appendix F to IMC 0612 provides examples.

- The staff revised the assessment process and expected NRC and licensee actions as provided for in the action matrix in response to inspection and performance indicator results as follows:

  - For the third consecutive assessment letter identifying the same substantive cross-cutting issue with the same cross-cutting theme, the staff modified IMC 0305, "Operating Reactor Assessment Program," to provide an option for the NRC to request that the licensee perform an assessment of safety culture.

  - For licensees in the regulatory response column, the staff modified IP 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," to verify that the licensee's root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components.

  - For licensees in the degraded cornerstone column, the staff modified IMC 0305, "Operating Reactor Assessment Program," to provide the expectation that the licensee's evaluation of the root and contributing causes will determine whether deficient safety culture components caused or significantly contributed to the risk-significant performance issues. The revised IMC 0305 will allow the NRC to request the licensee to complete an independent assessment of safety culture if the NRC determines that the licensee did not recognize that safety culture components caused or significantly contributed to the risk-significant performance issues. The staff also modified IP 95002, "Supplemental Inspection Procedure for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," to require inspectors to independently determine whether any safety culture components caused or significantly contributed to the individual or collective (multiple white inputs) risk-significant performance issues.

  - For licensees in the multiple/repetitive degraded cornerstone column, the staff modified IMC 0305 to provide the expectation that the licensee will perform an independent assessment of its safety culture. The staff is modifying IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstone or Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," to require the staff to (1) assess the licensee's independent evaluation of its safety culture and (2) independently perform an assessment of the licensee's safety culture.

The enclosure provides a full description of the changes to the ROP, including the safety culture components and specific enhancements to the IPs and IMCs.

Implementation Phase-In

The NRC implemented the revised ROP documents on July 1, 2006, except for IP 95003. The ROP uses an annual assessment cycle, with input from inspections that are conducted at preestablished periods that vary based on IPs or in response to identified performance deficiencies or events. Therefore, the NRC is phasing in the ROP changes effective July 1, 2006, as follows:

General

- All event response inspections performed after July 1, 2006, will use the revised IPs (IP 71153, IP 93800, and IP 93812). If an inspection began before July 1, 2006, the inspector would use the existing procedure; if the inspection began after July 1, 2006, the inspector will use the revised procedures.

- If the biennial inspection based on IP 71152 began before July 1, 2006, the inspector would use the existing procedure. If the inspection began after July 1, 2006, the inspector will use the revised procedure.

- The NRC will document cross-cutting aspects of findings in accordance with the revised process as provided in IMC 0612 for inspections that began after July 1, 2006.

- If at the time of the mid-cycle review meetings in August 2006, the licensee has a third consecutive assessment letter with the same substantive cross-cutting issue with the same cross-cutting theme, the NRC will not consider the option of requesting a licensee to conduct an assessment of safety culture. However, if at the end-of-cycle assessment in February 2007, a licensee has a substantive cross-cutting issue with the same cross-cutting theme for three or more consecutive assessments, the staff will have the option of requesting that the licensee conduct an assessment of safety culture.

- When evaluating licensee performance during the mid-cycle and end-of-cycle reviews, the staff considers all information that has been documented through the inspection program. If a licensee has voluntarily conducted a self-assessment of safety culture and the staff has reviewed it using IP 71152 or another procedure, the staff will use the information obtained as it evaluates the cross-cutting criteria provided in IMC 0305, including the possibility of closing a substantive cross-cutting issue.

Regulatory Response, Degraded Cornerstone, and Multiple/Repetitive Degraded Cornerstone Columns of the ROP Action Matrix

- For licensees in the regulatory response column of the action matrix that did not receive supplemental inspection IP 95001 as of July 1, 2006, the NRC will follow the guidance in the revised IMC 0305 and perform the revised inspection. Those licensees in this column of the action matrix that have already received supplemental inspection IP 95001 will not receive an additional IP 95001 inspection using the revised guidance.

- For licensees in the degraded cornerstone column of the action matrix that did not receive supplemental inspection IP 95002 as of July 1, 2006, the NRC will follow the guidance in the revised IMC 0305 and perform the revised inspection. Those licensees in this column of the action matrix that have already received supplemental inspection IP 95002 will not receive an additional IP 95002 inspection.

- For licensees in the multiple/repetitive degraded cornerstone column of the action matrix that did not receive supplemental inspection IP 95003 as of July 1, 2006, the NRC will expect that the licensee will independently assess its safety culture, and the NRC will perform the revised IP 95003 inspection to both review the licensee's independent assessment of its safety culture and to conduct an independent evaluation of the licensee's safety culture. Those licensees in this column of the action matrix that have already received supplemental inspection IP 95003 and are under a confirmatory action letter will not receive an additional IP 95003 inspection using the revised guidance.

Other Implementation Phase-In Issues

- The staff will not revisit inspection results for recently completed inspections or request licensees to take actions to meet the revised inspection or assessment guidance for past assessment cycles.

- If a licensee commits or is requested by the NRC to perform a safety culture assessment, the licensee will typically provide the results of the requested safety culture assessment to the NRC. The NRC will then make the assessment results publically available. At a minimum, the NRC will document its reviews of licensee safety culture assessments in NRC inspection reports.

As in the past, the staff will continue to have a process available to deviate from those actions described above on a case-by-case basis, consistent with the deviation guidance/criteria in IMC 0305.

Assessment of the ROP during the Implementation Period

The staff implemented the revised guidance on July 1, 2006. The staff will assess the changes to the ROP consistent with the current ROP assessment process in IMC 0307, "Reactor Oversight Process Self-Assessment Program," to determine that the revisions continue to meet the ROP regulatory principles of being objective, understandable, predictable, transparent, risk informed, and performance-based. The assessment will also determine whether the revisions have met the intended objectives and outcomes. The staff will seek opportunities for stakeholders to provide feedback on the implementation of the changes to the ROP (e.g., through the ROP monthly public meetings, external surveys, and regional utility group meetings).

BACKFIT DISCUSSION

The RIS requires no action or written response and is, therefore, not a backfit under Title 10, Section 50.109, "Backfitting," of the *Code of Federal Regulations* (10 CFR 50.109). Consequently, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

The NRC did not publish in the *Federal Register* a notice of opportunity for public comment on the RIS because the RIS is informational and pertains to staff actions that do not depart from current regulatory requirements and practices.

FEDERAL REGISTER NOTIFICATION

The NRC did not publish in the *Federal Register* a notice of opportunity for public comment on the RIS because the RIS is informational and pertains to staff actions that do not depart from current regulatory requirements and practices.

CONGRESSIONAL REVIEW ACT

The NRC has determined that this action is not subject to the Congressional Review Act.

PAPERWORK REDUCTION ACT STATEMENT

The RIS references information collection requirements that are subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) approval number 3150-0011.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

CONTACT

The RIS requires no specific action nor written response. If you have any questions about this summary, please contact one of the technical contacts listed below.

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Enclosure: Summary of the Reactor Oversight Process Safety Culture Approach

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

Inside NRC

Volume 28 / Number 18 / September 4, 2006

Safety of existing fleet to remain the top priority at NRC, Klein says

Even as NRC prepares for a stream of new plant license applications and looks for ways to accelerate the estimated 42-month review, the agency's attention will not stray from the current fleet of operating reactors, NRC Chairman Dale Klein said last week. Klein said the agency's top priority will remain focused on ensuring that the existing plants operate safely. "We do not need to become distracted on the new plants and forget about the existing plants," he told reporters August 30 at a Platts Energy Podium event.

It's a message he also has stressed internally, including at a meeting later that day with the entire staff of the Office of Nuclear Security and Incident Response. "He made it clear [we've] got to get it right in the operating side," said Scott Morris, chief of the reactor security branch within NSIR.

NRC is in the process of reviewing four early site permit applications and one design certification request, and is involved in pre-application reviews with two other reactor designs. But the real test of NRC's ability to handle the new plant workload will begin next year when the first of the combined construction permit-operating license (COL) applications is filed.

Klein said NRC is expecting some 13

applications with requests to build up to 27 new units. That list includes TXU, which publicly announced plans August 31 to possibly build between 2,000 and 6,000 megawatts of new nuclear generating capacity in Texas and other states (see story below). More COL requests could follow the initial wave.

Five years ago, the Nuclear Energy Institute set a goal of constructing 50,000 MW of new nuclear generating capacity by 2020 (Nucleonics Week, 31 May '01, 1). Several companies need to have more power online by 2014 or 2015 and plan to make a decision in the next couple years whether to build.

In an August 29 address at a conference hosted by the Goizueta Directors Institute at Emory University in Atlanta, Klein said he is taking a wait-and-see approach on the industry's stated aspirations.

"When I hear it said we're going to build 50 nuclear plants in the next 20 years, I say 'show me' — show me the designs, and then show me the hardware and construction, and then show me you have the people and procedures in place to run those plants in a way that protects the public health and safety," Klein said, invoking the unofficial "show me" motto from his home state of Missouri.

"And as importantly, show me that you are maintaining the capability of running the current fleet of plants at the same high level," Klein said.

The public, nuclear industry, Congress, and others will be watching how NRC handles the workload. In fact, the House Appropriations Committee is planning to hold a hearing September 13 on impediments to nuclear power. Former NRC chairman Nils Diaz is one of the invited witnesses, said one source.

Industry officials and some in the financial community have expressed concern that a lengthy NRC licensing review could threaten new plant construction ambitions. Klein told reporters last week that the agency has been hiring more staffers in anticipation of processing the large number of applications and that the agency would manage the workload "in a safe and reliable fashion." He did not rule out the possibility of the agency establishing some type of prioritization system.

Shorter review

Klein, who joined the agency July 1, said he would like to see an acceleration of the staff's estimated 42-month COL review schedule, which includes 30 months for a technical review and another 12 months for a licensing proceeding. A former assistant secretary at the Department of Defense, Klein said it would have been unacceptable for him to tell Defense Secretary Donald Rumsfeld that he had a "great idea and in about three and a half years I can give you an answer." Klein said, "The question is not whether the door hits me on the way out, but how hard it hits me."

The chairman said he believed the staff could "curtail" the 42-month review — with "no compromise on safety" — after the first few COL applications.

Responding to a question about how much the industry would save if the review were trimmed by several months, Klein said he was not looking at the schedule from a financial perspective. "To me, dollar savings is not as important as being predictable and stable," he said.

"If it takes 48 months to build a plant, but it takes 42 months to license it, it just doesn't sound right," Klein said. "Particularly if it's a standardized plant." Privately, Klein is said to have challenged the staff to look at the feasibility of a 12-month schedule for reviewing a COL application.

Although Klein seems confident that NRC can cut substantial time off the period for reviewing COLs, there is apparently concern on the part of some NRC staffers that the agency not move too aggressively to reduce review times. "We're not building doughnut factories," one NRC staffer said.

Klein is also said to be interested in other ways that the COL review schedule could be curtailed, including looking at an idea proposed by former Commissioner James Curtiss, now a partner with the Washington law firm Winston & Strawn, to pass legislation to eliminate the need for a mandatory hearing before an Atomic Safety and Licensing Board in uncontested proceedings (INRC, 7 Aug., 1).

At the Platts Energy Podium, Klein said another one of his goals is to wipe out from the books some of the outdated regulations. Comparing NRC to Congress, Klein said both bodies "pass regulations and laws, but we never unpass them." The result, he said, is that nothing is taken away. "So I'd like to see the agency take a look at the concept Lean Six Sigma" in reviewing NRC's work processes, he said. "The management tool would focus on how the agency could work faster and better, or more accurately," Klein said. "I would like for us to be more efficient, yet more accurate," he said.

IT upgrades

Klein plans to usher in a new era of information technology at the NRC. He said some of the NRC's systems, both software and hardware, needed an overhaul. Holding up his BlackBerry, a wireless device that syncs to a desktop e-mail account and other online services, Klein said he was surprised to find the agency did not use the technology before his arrival.

"If you look at the way modern communications occur, this is just the way we are able to stay in communication — and communicate better and more efficiently," he said. An NRC spokesman later told Platts that the agency now has distributed 38 Black Berries to employees such as NRC office directors "who have a critical role in incident response." The agency plans to expand distribution of the devices to a larger group of staffers on the incident response teams, he said.

Klein said he was concerned that if NRC did not upgrade to more widely used software systems, the agency would have trouble attracting IT talent. The NRC uses Novell GroupWise for e-mail and Corel WordPerfect for its office suite, including word processing. Klein said that while undergoing his confirmation process, he was unable to open documents sent to him by the NRC. And he learned his experience was not unique. "I heard that same story from congressional staffers, NEI and (others in the) industry," he said.

Klein also has his sights on the financial system that NRC uses for billing services, which he described as "bulky" and inefficient.

"We're embarking on a major improvement on the IT side of the house so we can be a little more modern and correspond to people in the electronic world," he said.

In mid-July NRC signed a new one-year service contract for 112 iridium satellite phones, with one-year options to renew. An NRC spokesman said the agency also signed for service for five mobile phones. The 117 phones go to every resident inspector site, the headquarters operations center, each regional response center, the Office of Information Services, NSIR, the Office of Nuclear Material Safety and Safeguard, the Office of Administration, the chairman, commissioners, the executive director for operations and deputy executive directors; the spokesman said. He said the phones are strictly for use in emergencies.—***Jenny Weil, Washington***

NRC setting up new program to make best use of lessons learned

NRC is instituting a new program to maximize the benefits of its lessons-learned efforts and to ensure that "major organizational problems identified by lessons learned will not recur," the agency said in documents issued last month.

Management Directive 6.8 and an August 4 paper (Secy 06-175) sent to the commissioners by Executive Director for Operations Luis Reyes provided details on the new program, including a pilot test that the Secy paper said was successful. The program had its origins in the lessons learned task force (LLTF) assembled after the discovery in 2002 of severe corrosion in the reactor vessel head at Davis-Besse. The work of that group led to a review of the effectiveness of recommendations from previous task force groups, and, in turn to a recommendation for the new program on how to institutionalize lessons learned.

Speaking at an August 30 Platts Energy Podium event, NRC Chairman Dale Klein emphasized the importance of incorporating the lessons learned into the agency's procedures and practices. Referring to the Davis-Besse situation, he said: "Whenever you find things of that nature, you do lessons learned." But, he added, "More importantly, you do lessons implemented so that it does not happen again."

When Loren Plisco, the leader of the lessons-learned team and the deputy regional administrator for Region II, briefed the commission last November on the team's work, Commissioner Peter Lyons said one potential difficulty would be in determining which lessons-learned findings should merit the special attention (INRC, 14 Nov. '05, 3). A July 14 summary report from Plisco to Reyes said that most lessons-learned reports issued to date have not established priorities among their findings.

To make sure that only the most important items receive the added attention of the new program, the management directive establishes criteria that a "potential lessonslearned item" must meet. For example, the item must have "significant organizational, safety, security, emergency preparedness, or generic implications," the directive said. Also, there must be a root cause that "exists — or can be identified," it said.

During the pilot test of the program by the NRC's 2005 Hurricane Season Task Force, the threshold criteria were found to be "sufficiently clear and discriminating," the Secy paper said. The paper was publicly released August 21.

According to the directive, an "ideal" lessons-learned program includes several "basic processes," including a "configuration management process that provides assurance that the changes made to incorporate corrective actions taken for lessons learned will not be subsequently altered or removed without adequate review." Also highlighted is a "knowledge management process" that "disseminates the lessons learned to appropriate personnel and ensures that a library of historical lessons-learned information is maintained." The lessons learned program is tied to a broader knowledge-management effort, which is a high priority at the agency (INRC, 7 August, 1).

John Lamb, a senior assistant in the EDO office's technical and regional programs section, has been tapped to manage the new program, the Secy paper said. There also will be a Lessons-Learned Oversight Board, which "should include at least one Senior Executive Service representative from each major program office," the management directive said.

Web tracking

The new program also includes a web-based system to capture and track lessons learned. The system, to be called the Agency Lessons-Learned System, or ALLS, will contain information about a specific topic, including any related reports, recommendations, corrective action plans or closeout documents. The system is expected to be operational in June 2007 but could be pushed back, the Secy paper said. Until then, the staff can use other electronic systems — NRC's document system Adams and, soon, the EDO's action tracking system, or EDATS.

—Daniel Horner and Jenny Weil, Washington