United States Nuclear Regulatory Commission

Protecting People and the Environment

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

December 4, 2009

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

OVERVIEW

MARIO V. BONACA

Accomplishments

- Since our last meeting with the Commission on June 4, 2009, we issued 20 Letter Reports:
- Topics included:
  - License Renewal Applications
  - ITAAC Closure Process
  - North Anna COL Application and SER with Open Items

- 3-Dimensional Finite Element Analysis of the Oyster Creek Drywell Shell
- TRACE Thermal-Hydraulic System Analysis Code
- Fire Protection Issues
- Steam Generator Action Plan Items
- Cyber Security Programs for Nuclear Plants

Containment Accident Pressure Issue

- Issued a letter on March 18, 2009, describing ACRS position and making several recommendations to facilitate resolution of the differences between the ACRS and the staff on the containment accident pressure (CAP) issue, and briefed the Commission on our recommendations on June 4, 2009

- In its June 4, 2009, response to our March 18, 2009, letter, the EDO stated:
  - The staff is evaluating some of the ACRS recommendations which entail generic implementation, e.g. revising Regulatory Guide 1.82. But, this evaluation will take some time
  - In the near term, the staff is evaluating and factoring ACRS questions and suggestions into its ongoing review of the extended power uprate application for Browns Ferry Units 1, 2, and 3

- In September 2009, the staff informed the licensees of Browns Ferry and Monticello plants that, until additional regulatory guidance is developed for dealing with the CAP credit issue, completion of the review of the EPU applications for these plants will be delayed

- We will meet with the staff to discuss additional regulatory guidance to address the CAP credit issue, when available

New Plant Activities

- Completed review of the draft SER Chapters for the ESBWR design certification application
  - Provided six interim letters on 20 Chapters
  - Reviewing the resolution of open items and the ACRS issues
  - Will review the final SER

- Reviewed draft SER on North Anna, Unit 3, COL application referencing the ESBWR design. Issued letter dated October 23, 2009
- Reviewing design certification application and draft SER associated with the US-APWR design
  - Issued a letter on June 19, 2009, on the Topical Report, “Defense in Depth and Diversity,” related to US-APWR design

- Reviewing amendment to the AP1000 Design Control Document
- Reviewing draft SER on the EPR design certification application
- Reviewing the Reference COL Application for the AP1000 design, and the draft SER
- Continuing to interact with the NRO staff to establish schedule for review of design certification and COL Applications to ensure timely completion of ACRS review

Major Review Activities

- Design Certification applications
- Combined License applications
- License Renewals
- Extended Power Uprates
- Fire Protection
- Digital I&C / Cyber Security
- Safety Culture

- Rules and Regulatory Guidance
- Safety Research Program
- SOARCA
- Containment Accident Pressure Credit Issue
- PWR Sump Performance
- Reactor Fuels
- Radiation Protection and Materials Issues

ACRS Reviews of New Reactor Applications

- We conducted a Mini-Retreat on November 7, 2009, which was focused on optimizing our reviews of amendments to previously-certified designs
- Several operational items were identified for enhancement. We have initiated discussion with NRO on these items and are preparing a memorandum to the EDO with specific conclusions and recommendations

Observations on Recent License Renewal Reviews

- Recent License Renewal reviews that are the subjects of two of the subsequent presentations (Beaver Valley and Oyster Creek) demonstrate that the License Renewal Program continues to provide safety benefits
- ACRS will continue to focus on lessons learned from our reviews that may have generic implications for other facilities

Closure of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)/ Design Acceptance Criteria (DAC)

Dennis C. Bley

Background

- ITAAC is defined in 10 CFR 52.47(b)(1), with the closure requirements specified in 10 CFR 52.99, “Inspection During Construction”
- SRM on SECY-90-377 stated that applications for design certification should reflect a complete design except to accommodate as-procured hardware characteristics

- SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews"
  - introduced DAC
  - identified need
  - identified potential pitfalls
- ACRS issued three reports addressing ITAAC and DAC

ACRS Reports

- 1990 Report on SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52"
  - Agreed with process and recommended that the staff focus the scope on that needed for safety
- 1992 Report "Use of Design Acceptance Criteria (DAC) during 10 CFR Part 52 design certification reviews"
  - Supported DAC for limited applications
  - Extensive use of DAC may be adverse to safety

ACRS July 24, 2009, Report on RG 1.215, “Guidance for ITAAC Closure under 10 CFR Part 52”

- RG 1.215 identifies three options for the closure of DAC:
  - amendment of the design certification rule
  - COL application review process
  - ITAAC after COL issuance

- The third option especially needs clarification
- RG 1.215 provides an acceptable approach for closing ITAAC
- RG 1.215 should be revised to specify where the detailed closure process guidance for DAC will be provided

- The DAC closure process guidance should include an in-depth review comparable to the usual design certification process to ensure adequacy of the design
- The DAC closure process guidance should be provided for ACRS review

- Staff has formed a Task Working Group to develop DAC resolution process
- October 16, 2009, SRM directs staff to complete the proposed revisions to the regulatory guidance by the end of 2010

Amendment to the AP1000 Design Control Document

Harold Ray

ACRS Review in 2009

- Full Committee briefings in May and November
  - Amendment changes to DCD presented to ACRS on a Safety Evaluation Report (SER) chapter-by-chapter basis
- Three two-day subcommittee meetings to date
  - July, October, and November
  - July meeting also included Bellefonte RCOLA

Status of Review

- Review is current with available SER Chapters
  - 15 of 19 chapters with open items
  - One partial chapter with open items
  - Approximately 100 of 130 open items are not yet closed by NRC staff
  - A meeting is scheduled in January when additional chapters are expected to be available

Amendment Reflects Extensive Changes to DCD

- As identified by the applicant, the purpose of the amendment is to:
  - Replace COL information items with specific design
  - Replace Design Acceptance Criteria (DAC) with specific design
  - Respond to NRC requirements
  - Enhance standardization
  - Reflect design maturity
  - Incorporate design improvements

Amendment Overview

- As identified by the applicant, key review issues include:
  - Response to developing security requirements
  - Specific designs to replace DAC for
    - Instrumentation & Control
    - Human factors engineering
    - Piping
  - Containment sump and downstream effects

- Structural design and seismic analyses
- Control room ventilation
- Enhanced integrated head package
- Automated Statistical Treatment of Uncertainty Method (ASTRUM)
- Non plant-specific technical specification changes

• The amendment is supported by over 100 technical reports submitted by the applicant

Potential ACRS Concerns

- No items of potential concern have been identified to date that were not previously identified by staff and remain under staff review

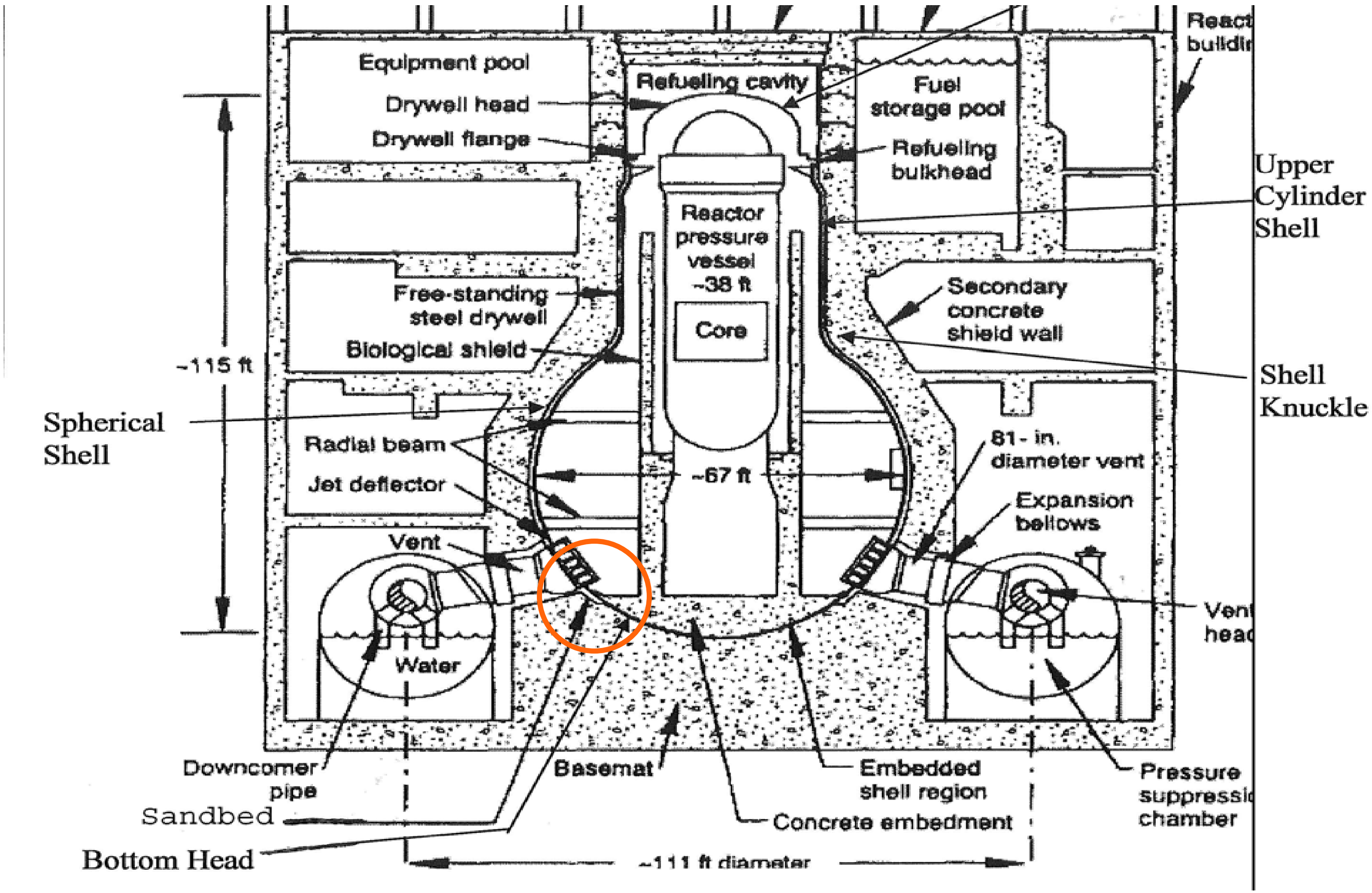
Oyster Creek Drywell Shell 3-D Finite Element Analysis

William J. Shack

Background

- Corrosion identified late 1980s
  - Lower spherical portion of the shell – "sandbed" region
  - Unevenly distributed within the 10 bays
- 2/1/07 ACRS Meeting - Exelon committed to perform a 3-D FEA
- 2/8/07 ACRS Report on the Oyster Creek LRA recommended a license condition

Oyster Creek Containment



Drywell Shell Analyses

- General Electric 1992
  - Assumed uniform reduction in shell thickness (sandbed region)
  - Current licensing basis analysis
- Sandia 2007
  - 3D analysis, but included conservative assumptions for thickness & capacity reduction factor
  - Confirmed current configuration meets licensing basis

- Structural Integrity Assoc. (SIA) 2009
  - More realistic analysis
  - Used modified capacity reduction factor to account for biaxial stresses
  - Performed base case and sensitivity analyses to address measurement uncertainty
- SIA results suggest actual margins significantly larger than ASME Code minimums (e.g, 3.4 vs 2.0 for buckling during refueling)

Finite Element Analysis

- Detailed model using shell elements
  - All penetrations greater than 3-in. diameter were included
  - Over 400,000 elements
  - Mesh Sensitivity: approximately 1,000,000 elements

Primary Sources of Uncertainty in 3-D FEA

- The characterization of the thickness of the sandbed region
- The calculation of the capacity reduction factors

Characterization of Thickness

- Licensee estimates based on UT thickness data from grids at Elevation 11' 3"
  - Supplemented by the grids in the trenches in Bays 5 and 17
  - Supported by visual examination and engineering judgment
- Sandia estimates based on individual UT measurements of locally thinned areas; more conservative, but generally consistent with licensee's estimates

Modified Capacity Reduction Factor

- FEA buckling analysis assumes perfect shell geometries
  - Capacity reduction factors introduced to account for imperfections
- Primary justification for capacity reduction factors are experimental results formalized as ASME Code Case
  - ACRS consultant provided independent, analytical assessment; Code Case results are slightly more conservative

ACRS Report

- Analysis has been reviewed by the staff, the ACRS, our consultants, and by Becht Nuclear Services for New Jersey. General agreement that analysis was performed using good engineering practices and judgment
- Analysis fulfills licensee's commitment to provide a more realistic analysis that better quantifies the available safety margin for the current drywell shell configuration

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Beaver Valley License Renewal and Containment Liner Corrosion

J. Sam Armijo

Background

- In our letter of Sept 16, 2009, we recommended approval of the application for license renewal of BVPS Units 1 and 2
- Critical issue in the renewal was additional evaluation of localized corrosion of the Unit 1 carbon steel containment liner

2006

- During a steam generator replacement, pitting corrosion was discovered at the containment liner-to-concrete interface
- The pits were found in three areas but did not penetrate through the liner
- Two areas were repaired and one is being monitored for evidence of continued corrosion
- These pits were attributed to corrosion early in plant life

2009

- A paint blister was observed during a Unit 1 IWE visual inspection
- Investigation of the blister revealed a 1 in. x 3/8 in. through wall hole in the liner
- A decomposed piece of wood, embedded in the concrete wall, was found at the location of hole
- The wood was a construction spacer that should have been removed prior to concrete placement

Observations

- The mechanism responsible for the through-wall liner penetration in Unit 1 is reasonably well understood
- The localized corrosion was caused by moisture at the wood-to-steel interface
- When Unit 2 was constructed, welded angle irons were used as spacers between the liner and the first row of re-bar rather than wood

Future Inspections

- Near term visual inspection of all accessible liner surfaces will be performed
- Focused, non-random, UT inspections will be performed to determine whether additional localized corrosion is occurring
- 75 or more randomly selected areas will be examined by UT to evaluate the condition of a representative portion of the liner

- Inspections of the Unit 1 liner will be completed in time for corrective actions prior to entering the period of extended operation
- Although no liner corrosion has been observed in Unit 2, similar visual and UT Inspections will be performed prior to entering the period of extended operation

ACRS Conclusions

- The proposed inspection programs and related commitments provide reasonable assurance that liner integrity will be adequately maintained during the period of extended operation

Future Activities

- ACRS is expecting a briefing/update from NRR in 2010 regarding containment liner corrosion issues and actions taken by the staff to address them generically for operating plants
- NRC staff activities include:
  - Supplementing IN 2004-09
  - Potential changes to the NRC's outage inspection procedures for additional guidance on containment walkdowns

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Cyber Security for Nuclear Power Plants

George E. Apostolakis

RG 5.71, “Cyber Security Programs For Nuclear Facilities”

- 10 CFR 73.54 requires that the licensees produce policies and plans for cyber security by November 23, 2009
- RG 5.71 should be issued to support compliance with the rule
- RG 5.71 adapts NIST Standards for the development of plans but does not provide guidance to evaluate their adequacy

- After the initial implementation of the cyber security plans, RG 5.71 should be revised to include the resulting insights and provide guidance regarding the adequacy of cyber security plans and policies

- Longer-term research projects should be initiated in the following areas:
  - Exploration of the use of PRA insights, in particular those regarding accident sequences, in cyber security
  - Development of better guidance on the interaction between cyber security and safety
  - Investigation of the possibility of supply chain attacks

Abbreviations

3-D	3-Dimensional
ACRS	Advisory Committee on Reactor Safeguards
AP1000	Advanced Passive 1000
ASME	American Society of Mechanical Engineers
ASTRUM	Automated Statistical Treatment of Uncertainty Method
CAP	Containment Accident Pressure
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
COL	Combined License
DAC	Design acceptance criteria
DCD	Design Control Document
EDO	Executive Director for Operations
EPU	Extended Power Uprate
EPR	Evolutionary Power Reactor
ESBWR	Economic Simplified Boiling Water Reactor
FEA	Finite Element Analysis
I&C	Instrumentation & Control
IN	Information Notice
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
IWE	Subsection in the ASME Code XI, Division 1, dealing with primary containment inspection programs
LRA	License Renewal Application
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors
NRR	Office of Nuclear Reactor Regulation
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCOLA	Reference Combined License Application
RG	Regulatory Guide
SECY	Office of the Secretary
SER	Safety Evaluation Report
SIA	Structural Integrity Assoc
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRM	Staff Requirements Memorandum
TRACE	Thermal-Hydraulic System Analysis Code
U.S.	United States
US-APWR	United States Advanced Pressurized Water Reactor
UT	Ultrasonic testing