Westinghouse Presentation for Fuel Performance Update Meeting Management Licensing Overview (Slide Presentation for April 12, 2006)

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AGENDA
Westinghouse Semi-Annual Fuel Performance Update
April 12, 2006
Westinghouse Office
Rockville, Maryland

Wednesday, Apr 12	**Licensing Review (Westinghouse & NRC)**	
8:30 – 8:40 am	Westinghouse Organization	[ ]^{a,c}
8:40 – 9:00 am	NRC Update New Organization	NRC
9:00 – 9:30 am	PWR/BWR Topicals and Schedule	[ ]^{a,c}
9:30 – 10:00 am	Fuel Input on AP1000 COL	[ ]^{a,c}
10:00 – 10:15 am	Break	
10:15 – 11:30 am	General Licensing Issues [SRWM F_Q Surveillance CE Seismic Issue Damaged Annular Axial Blanket Pellet Standard Review Plan Updates High Burnup Licensing Personnel Changes]^{a,c} Other	All
11:30 - noon	Wrap-up Next meeting Agenda Format	[ ]^{a,c}
Noon – 1:00 pm	Lunch/Informal Discussion between NRC & Westinghouse	
1:00 – 3:00 pm	GSI-191 Impact on Fuel	[ ]^{a,c}

Dress is Business Casual

Westinghouse Organization - (NRC Interface)

a, c

PWR/BWR Topicals and Schedules

Fuel Performance Update Meeting
Rockville, MD
April 10-12

PWR Topical Schedule

a, c

PWR Topical Schedule (cont.)

a, c

[ ]

BWR Topical Schedule

a, c

BWR Topical Schedule (cont.)

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[ ]

General Issues

Fuel Performance Update Meeting
Rockville, MD
April 10-12

SRWM

Conditions that are specified in the SERs for WCAP-8846-A and WCAP-13749-P-A were not specifically considered in topical report WCAP-16260-P-A; however, Westinghouse's interpretation is that compliance to these SER requirements is maintained

In WCAP-8846-A, page 7 of the SER, the NRC stated:

- "An acceptable program might include several rod reactivity checks during the first core cycle and rod measurements of all the rod banks at refueling outages thereafter."

In WCAP-13749-P-A, pages 6 and 8 of the SER, the NRC stated:

- "The BOL MTC measurement is made at Hot-Zero-Power (HZP) conditions and is an accurate measurement characterized by relatively small uncertainties."
- "2) <u>Changes in Core/Fuel Designs and the Measurement Data Base</u>
  The predictive correction should be reevaluated if changes in core/fuel designs or the MTC calculation-to-measurement data base have a significant effect on the MTC predictive correction (Section-3)."

SRWM (cont.)

Westinghouse issued a letter to the NRC documenting its interpretation of continued compliance with the SER requirements of WCAP-8846-A and WCAP-13749-P-A using "The Spatially Corrected Inverse Count Rate (SCICR) Method for Subcritical Reactivity Measurement" testing methodology documented in WCAP-16260-P-A (also referred to as Subcritical Rod Worth Measurement – SRWM)

The letter to the NRC requested acknowledgement from the NRC on Westinghouse's interpretation

The letter to the NRC and any response from the NRC will be incorporated into WCAP-16260-P-A as a revision to the topical to capture this interaction

F_Q Surveillance

The surveillance exclusion zone is an axial zone of the core where the power distribution surveillance requirements do not apply

Historically, for most plants, the surveillance exclusion zone has been defined to be the top and bottom 15% of the core

The original reason for having a surveillance exclusion zone is as stated in WCAP-10216-P-A, Appendix A:

- "the difficulty in making a precise measurement for this region and the low probability that this region would be more limiting than the central 70% of the active core."

F_Q Surveillance (cont.)

Recent trends in core designs have resulted in an increased frequency of cases where the peak transient F_Q (or minimum transient F_Q margin for plants with an axially varying F_Q limit), is predicted to occur in the surveillance exclusion zone over a portion of the operating cycle.

There have been cases where Westinghouse has recommended reductions in the size of the surveillance exclusion zone, in the interest of assuring that the power distribution surveillance remains effective at detecting potential violations over portions of an operating cycle.

F_Q Surveillance (cont.)

A 10 CFR 50.59 screening was prepared that addresses a change in the surveillance exclusion zone to a value less than the current 15% top and bottom of the core (i.e. 8% top and bottom or no exclusion zone at all)

The 10 CFR 50.59 screening is based on the assumption that plants have relocated the exclusion zone outside of the Technical Specifications (i.e., no Technical Specification changes are required for those plants that have the exclusion zone outside of the Tech Specs. – moved to BASES)

For those plants that need to make a Tech Spec change, the information provided in this 10 CFR 50.59 screen can be used by the specific plant to support a license amendment request

CE Seismic Issue

During recent analyses performed for a licensee, Westinghouse discovered that the fuel assembly preload value used in the original reactor vessel internals (RVI) and the fuel vertical seismic analyses of record (AOR) performed in 1976 was higher than the revised value used in an asymmetric loads analysis performed in 1981

Further investigation determined that the lower value was correct (i.e., the 1981 vintage)

Because of the larger, incorrect value for fuel hold-down spring preload used in the original analyses, fuel assemblies were not calculated to lift off the core support plate during seismic excitations

The fuel was calculated to lift off when the analysis was re-run using the correct preload value, resulting in an impact between the fuel and the core support plate with an increase in vertical loads in the RVI and fuel

CE Seismic Issue (cont.)

While addressing the fuel pre-load issue, Westinghouse discovered that seismic loads (horizontal) were not properly incorporated into the evaluation of the fuel assembly

Plants having a Westinghouse designed NSSS are unaffected by these issues

Evaluations performed for CE NSSS plants demonstrated that a safety concern does not exist

Evaluations were performed to determine the stresses in the flange and end fitting regions of the guide tubes under seismic conditions

CE Seismic Issue (cont.)

Ten CE NSSS plants were found to be potentially affected by this issue (i.e., these plants operate with CE design fuel assemblies for which the subject guide tube regions were not properly analyzed)

For these CE NSSS plants, revised evaluations demonstrated that stresses in the guide tube flange/weld region and the lower end fitting/weld region are acceptable for Safe Shutdown Earthquake (SSE) + LOCA conditions

Because these guide tube regions satisfy SSE + LOCA stress limits, the fuel continues to meet safe shutdown requirements under all postulated design basis seismic conditions, i.e., core coolability is maintained and the capability of control rod insertion is assured

CE Seismic Issue (cont.)

While assessing the guide tube flange stresses, it was discovered that the Operating Bases Earthquake (OBE) allowable stress limits were exceeded for five CE-NSSS plants

Although the allowable stress limit was exceeded, it has been concluded through additional detailed analyses that no significant, permanent guide tube deformation will occur under OBE conditions (i.e., the mechanical integrity of the fuel assembly is maintained, and control rod insertability, insertion time requirements, and core coolability are not impacted)

Exceeding the OBE stress limits is a non-conforming condition, but it is not a safety issue

Licensee have been notified

A Nuclear Safety Advisory Letter (NSAL) will be issued in April 2006

Damaged Annular Axial Blanket Pellet

Damaged annular axial blanket (AAB) pellets were discovered in a limited number of fuel rods which contained zirc diboride integral fuel burnable absorber (IFBA) coating in the enriched fuel region (note: AAB pellets are not coated with IFBA)

This damage ranged from slightly chipped pellets to badly fragmented pellets

A root cause analysis concluded that the damaged AAB pellets are limited to the top end annular pellet added to IFBA rods using the automatic plenum adjustment equipment to adjust the stack length

The automatic plenum adjustment equipment has been used since the introduction of AAB pellets in 1991 and is only used with the IFBA fuel rods

Damaged Annular Axial Blanket Pellet (cont.)

Non-IFBA fuel may or may not contain AAB pellets; however, the automatic plenum adjustment equipment is not used with these rods

After reviewing the potential scenarios, the conclusion is that leaking fuel rods, as a result of the annular blanket pellet damage, are not credible

This conclusion is supported by a review of the leaker database which shows that the rate of leaking IFBA rods from unknown causes (which could potentially be associated with this issue) decreased at the time the number of IFBA rods with plenum adjustment using the automatic equipment was increasing

After extensive review, Westinghouse can find no evidence of a fuel performance or safety issue associated with the potentially damaged annular blanket pellet

Damaged Annular Axial Blanket Pellet (cont.)

A 10 CFR 50.59 screening has been issued to all Westinghouse licensees to address this issue

A review of the impact of the damaged AAB pellet indicates no impact on the reload safety analysis checklist (RSAC) parameters or the safety analyses, therefore it does not represent a safety concern

Based on these results, the fuel continues to function as it was designed and is acceptable for operation without restriction

Standard Review Plan Updates

General discussion topic:

What is the status of Standard Review Plan (SRP) updates?

Based on Westinghouse's detailed review and understanding of SRP Chapter 4, it is expected that only minor editorial changes are really needed to bring the SRP up to date (i.e., no major revisions should be needed)

High Burnup Licensing

General discussion topic:

Where do we (industry and NRC) stand?

- Reactivity Insertion Accident (RIA) issues with ultimate regulatory limits
- Incremental steps to high burnup
- Data substantiation (WCAP-15604-NP-A was suppose to do this, but fell short)
- Status of EPRI High Burnup Topical to support industry
- Other?

Personnel Changes

General discussion topic:

Both NRC and industry face loss of senior experienced personnel

- How do we ensure technical experience is not lost
- How do we ensure that the next generation does not fall into a "re-inventing the wheel syndrome" (i.e., not accepting precedence, re-opening analyses that have been accepted for years, etc.)
- Changes in management is resulting in loss of continuity and corporate knowledge
- Reviewer appear to stray from regulations and try to invoke new regulatory requirements outside the normal licensing process

Other Changes

General discussion topic:

- 10 CFR 50.46 Rulemaking
- Labs closing
- etc.

GSI-191 Long-Term Core Cooling Evaluation of Post-Accident Flows and Flow Paths into the Reactor Core

Discussion with US NRC
April 12, 2006

Purpose of this Meeting

- Present an approach to demonstrate sufficient long term core cooling is achieved to satisfy requirements of 10 CFR 50.46

- Program proposed to be conducted under PWROG sponsorship

General Approach

- Demonstrate sufficient long term core cooling provided
  - Remove decay heat, and,
  - Maintain a coolable core geometry

- Pursue a "Defense in Depth" Position
  - Flow to fuel is non-uniform
    - Non-uniform flow leads to non-uniform debris bed
  - Multiple flow paths for fluid to get to the core
    - Assess the flow paths
  - Address other concerns
    - Hot spots
    - Chemical Plate-out
  - Sensitivity calculations

General Approach (cont.)

- Evaluate using Chapter 6, "Alternate Evaluation" method of NEI 04-07
  - "Use highly conservative analysis methodologies for pipe breaks up to an alternate break size ≤ 14 inches"
  - "Use risk insights and more realistic analysis methodologies for pipe breaks larger than the alternate break size > 14 inches"

Assembly-to-Assembly Flow Variation

- It is known that:
  - There are assembly-to-assembly variations in fuel assembly inlet flow
    - Non-uniform flow from lower plenum into bottom of fuel
    - Complex geometry of fuel bottom nozzle
  - These variations are accounted for in the design

- These factors also operate under ECCS condition to provide for assembly-to-assembly flow variation

- Based on above, a non-uniform bed is formed

Evaluate Flow Paths to Core

- Identify the possible flow paths into the core, i.e.
  - Pressure relief holes in baffle/barrel design (some designs)
  - Flow from lower plenum through baffle region to top of core
  - Fill-up of intact cold legs, into steam generators and overflow into intact hot legs (for a hot-leg break scenario)
  - Others, as appropriate for specific design
- Evaluate geometry, hydraulic resistance of flow paths
- Assess head available to drive flow into the core
- Calculate total flow into the core
- Calculate decay heat to be removed
- Demonstrate flow is sufficient to remove decay heat for long-term cooling

Evaluate Other Concerns

- Fuel Hot Spots
  - At and behind grid straps due to collection of debris
    - Not at high-power locations
  - Assess axial conduction and flow rates
  - Demonstrate that conditions provide for
    - Decay heat to be removed, and,
    - Coolable core geometry maintained

- Chemical Plate-out
  - Account of scenario effects
  - Impact on heat transfer at rod surface (boiling)

- Sensitivity Calculations (Examples)
  - Flow Through Gaps
  - Axial Conduction

Summary

- This presentation outlines an approach to demonstrate sufficient long term core cooling will be provided to achieve:
  - Removal of decay heat, and,
  - A coolable core geometry is maintained

- The approach uses a "Defense in Depth" strategy:
  - Consider use of Chapter 6, "Alternate Evaluations" methods of NEI 04-07 as approved by NRC SER
  - Identify and evaluate flow paths for fluid to get to the core
  - Assessment of non-uniform flow to fuel on debris bed formation
  - Address other concerns
  - Sensitivity calculations
  - Show 10 CFR 50.46 requirements are met
    - Removal of decay heat
    - Maintain core in a coolable geometry