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

April 5, 2012

LICENSEE: Carolina Power & Light

FACILITIES: Shearon Harris Nuclear Plant, Unit 1

SUBJECT: SUMMARY OF JANUARY 11, 2012, MEETING WITH CAROLINA POWER & LIGHT TO DISCUSS REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT (TAC NO. ME6999)

On January 11, 2012, the U.S. Nuclear Regulatory Commission (NRC) staff conducted a closed meeting with Carolina Power and Light (the licensee) at NRC Headquarters, 11555 Rockville Pike, One White Flint North, Rockville, Maryland. The purpose of the meeting was to continue the discussion on the NRC staff concerns related to the realistic large break loss-of-coolant accident (LOCA) methodology amendment request for Shearon Harris Nuclear Plant, Unit 1 (HNP). Proprietary information was discussed during the meeting so it was closed to the public. To facilitate discussion, the NRC provided suggested discussion topics (Enclosure 2). The licensee presented non-proprietary version of a slide presentation (Enclosure 3).

DISCUSSION

In a letter dated August 22, 2011 (Agencywide Documents Access and Management System Accession No. ML11238A077), the licensee submitted an amendment request. The proposed request would revise the HNP technical specifications (TSs) to add a plant-specific methodology that implements AREVA's NRC-approved topical report, EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors, Revision 0" and add EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher upon approval of the specific revision by the NRC.

The licensee described the basis and intent of the submittal. This amendment request supports the ongoing efforts by the licensee to allow the use of the AREVA fuel cladding alloy designated as M5™. The licensee indicated that the approval of the use of the realistic LOCA methodology topical reports supports a migration away from the existing legacy methodologies to a more updated methodology. These activities are in preparation for the use of the new clad during the HNP Cycle 18 refueling outage, currently scheduled for early in the second quarter of 2012.

During the meeting, the licensee and the NRC staff continued the discussion of the December 11, 2011, closed meeting topics. These topics covered the droplet shattering model and its impact on heat transfer, the packing factors, the swell/rupture data considerations, and the application of the Sugimoto/Murao correlation. The licensee responded many of the NRC staffs concerns during the discussion. These responses are to be submitted to the NRC staff formally by the licensee. Minor changes were made to the discussion topics. No commitments or regulatory decisions were made by the NRC staff during the meeting.

Please direct any inquiries to me at 301-415-3302.

Sincerely,

Araceli T. Billoch Colón

Araceli T. Billoch Colón, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. List of Attendees
2. Discussion Topics
3. Licensee's Presentation

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LIST OF ATTENDEES

JANUARY 11, 2012, CLOSED MEETING WITH CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

REALISTIC LARGE BREAK LOCA LICENSE AMENDMENT REQUEST

U. S. NUCLEAR REGULATORY COMMISSION
Douglas Broaddus
Anthony Mendiola
Araceli T. Billoch Colón
Benjamin Parks
Eva Brown
Leonard Ward

Carolina Power & Light Company
John Caves
Dean Tibbitts
Dave Corlett
Mike Blom

AREVA
Bob Baxter
Bert Dunn
Nithian Nithianandan
Mireille Cortes
Gayle Elliot

ENCLOSURE 2

SHEARON HARRIS NUCLEAR POWER PLANT

DOCKET NO. 50-400

DISCUSSION TOPICS

FOR REALISTIC LARGE BREAK LOCA

LICENSE AMENDMENT REQUEST

SUGGESTED DISCUSSION ITEMS

LARGE BREAK LOSS-OF-COOLANT-ACCIDENT SUBMITTAL

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1

PROGRESS ENERGY

DOCKET NO. 50-400

1. For droplet break up model, show the drop sizes produced by the model for several low reflood rate data. Present the clad, vapor temperature and total heat transfer coefficient versus time at the measured axial locations. Show the heat transfer coefficient for all of the components comprising the dispersed flow film boiling (DFFB) heat transfer, including the interfacial heat transfer coefficient.

2. Since RELAP5 is one-dimensional the vapor temperature and droplets are distributed evenly across the hot channel. The code computed cross-section average quantities appears to fail to properly capture the very high temperature gradient in the vapor phase boundary layer near the wall so that the distribution of the evaporating water droplets play a fundamental role in the heat transfer process. In particular, interfacial heat transfer is over predicted. This appears to be a major limitation for all one-dimensional codes. Test data shows that the channel is three-dimensional with accumulation of drops in the central region and a highly superheated region near the walls. Modeling this multi-dimensional behavior leads to a substantial reduction in the interfacial heat transfer and limiting of the droplet de-superheating to the central core and not the highly superheated layer near the walls.

   Explain what adjustments are made to the DFFB model components to overcome this major discrepancy. That is, the sink temperature is not the average channel temperature for computing single phase heat transfer, an interfacial heat transfer between the drops and the vapor is control by the lower vapor temperature in the central core where the drops reside.

3. Due to the simplified one-dimensional averaging of thermodynamic quantities in RELAP5 and the limited data, it is difficult to quantify all of the component contributions to DFFB.

   a. Address how the magnitude of the droplet contribution is verified in the RELAP5 model.

b. Without detailed knowledge of the magnitude of all of the components to DFFB, validation of this model against reflood data may result in including other phenomena/effects that are not pertinent to the heat transfer benefits from the droplet break up model. Explain and justify the magnitude of the impact on DFFB heat transfer with this new model.

c. Describe the interfacial heat transfer model and the impact on interfacial heat transfer coefficient with the new droplet model. In comparing the DFFB against data with the new droplet model, show all of the contributions to the total heat transfer coefficient versus time at the peak clad temperature (PCT) location.

4. The packing fraction of 50 percent does not appear to capture all of the test data. Packing fraction as a function of burst strain varies in the range 52 to 80 percent based on data from Broughton, J. M, 1981, "[Power Burst Facility] PBF [loss-of-coolant accident] LOCA Test Series, Test LOC-3 and LOC-5 Fuel Behavior Report," NUREG/CR-2073. The Nuclear Energy Agency (NEA) Organization for Economic and Co-operation and Development (OECD) Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions State-of-the-art Report identifies 55.5 and 61.5 percent fill fraction for the FR2 reactor test E2. Values for the high burnup fuel in IFA-650.4 are expected to be higher than 70 percent, consistent with the bounds for PBF/LOCA gamma scanning and micrographies and FR-2. (See Grandjean, C "IRSN Calculation of the IFA-650.4 and .5 LOCA Tests ISRN, Cadadache, Fr. EHPG Meeting, Storefjell, March 12-15, 2007 meeting).

   Show the impact on PCT for fill fractions up to and including 80 percent. Please also describe how the fill fraction is sampled.

5. Address whether the use of a nominal decay heat curve has ever been applied to decay heat test data over the range of applicability to show that this approach captures all decay heat conditions. The discussion should also address the uncertainty in generating this nominal curve and demonstrate that use of the nominal curve does not capture the decay heat for the first two seconds. Provide a multiplier which appropriate captures the decay heat behavior during this first two seconds of the curve.

ENCLOSURE 3

SHEARON HARRIS NUCLEAR POWER PLANT

DOCKET NO. 50-400

LICENSEE'S PRESENTATION

FOR REALISTIC LARGE BREAK LOCA

LICENSE AMENDMENT REQUEST

Harris Realistic LBLOCA Question Response Meeting

January 11, 2012

Rockville, MD

~~Slides contain material proprietary to AREVA~~

NGG Think Fleet

Progress Energy

AGENDA

- Introductions
- Overview of Status
- Response to Individual questions from December 13th Meeting
- Schedule
- Concluding Remarks

Participants

Progress Energy

- Mike Blom
- John Caves
- Dave Corlett
- Dean Tibbitts

AREVA

- Bob Baxter
- Bert Dunn
- Mireille Cortes
- Nithian Nithianandan
- Gayle Elliott

PGN Reload Summary

	Current	Cycle 18
Feed Assembly Clad	Zircaloy 4	M5
LBLOCA Method	EMF-2087	ANP-3011 based on EMF-2103
PCT	2081 °F	1919 °F
Transient Oxidation	7%	<3%
Safety Analysis Core Power (MW)	2958	2958

LOCA History

Method	Method Type	Core Power (MW)	Clad	PCT	Transient Local Oxidation (%)
EMF-2087	Appendix K	2958	Zircaloy 4	2081	7.0
EMF-2103 Rev 0	Best Estimate	2993	Zircaloy 4	1887	2.02
EMF-2103, Rev 0 + Trans Pkg	Best Estimate	2958	Zircaloy 4	1930	1.95
ANP-3011	Best Estimate	2958	M5	1919	2.94

Overview of Question Response

- Additional Sensitivity Cases Run (Response to Question 1)
- Discussions of Questions 2 to 6 refined and responses prepared

Question #1

- Characterize droplet shattering model without modeling the fuel relocation
- Show the sensitivity of the fuel relocation to fuel relocation packing factor
- Consider including a range of packing factors 30 - 80%
- Utilize the Harris Nuclear Plant (HNP) limiting-peak clad temperature (PCT) case

Question #1
Droplet Shattering w/o Relocation

Question #1
Expand Sensitivity Study

Table 1: 0.5 Packing Fraction Cases with Hot Assembly Rupture

Question #1
Expand Sensitivity cont'd

Table 2: 0.6 Packing Fraction Cases with Hot Assembly Rupture

Question #1
Expanded Sensitivity cont'd

Table 3: 0.7 Packing Fraction Cases with Hot Assembly Rupture

Question #1
Summary of Expanded Sensitivity Study

Question #1 cont'd

Histogram of PCT with Droplet Shattering Activated - 0.7 Packing Fraction

Question #1 cont'd

Rupture node cladding temperature response for limiting Case-57

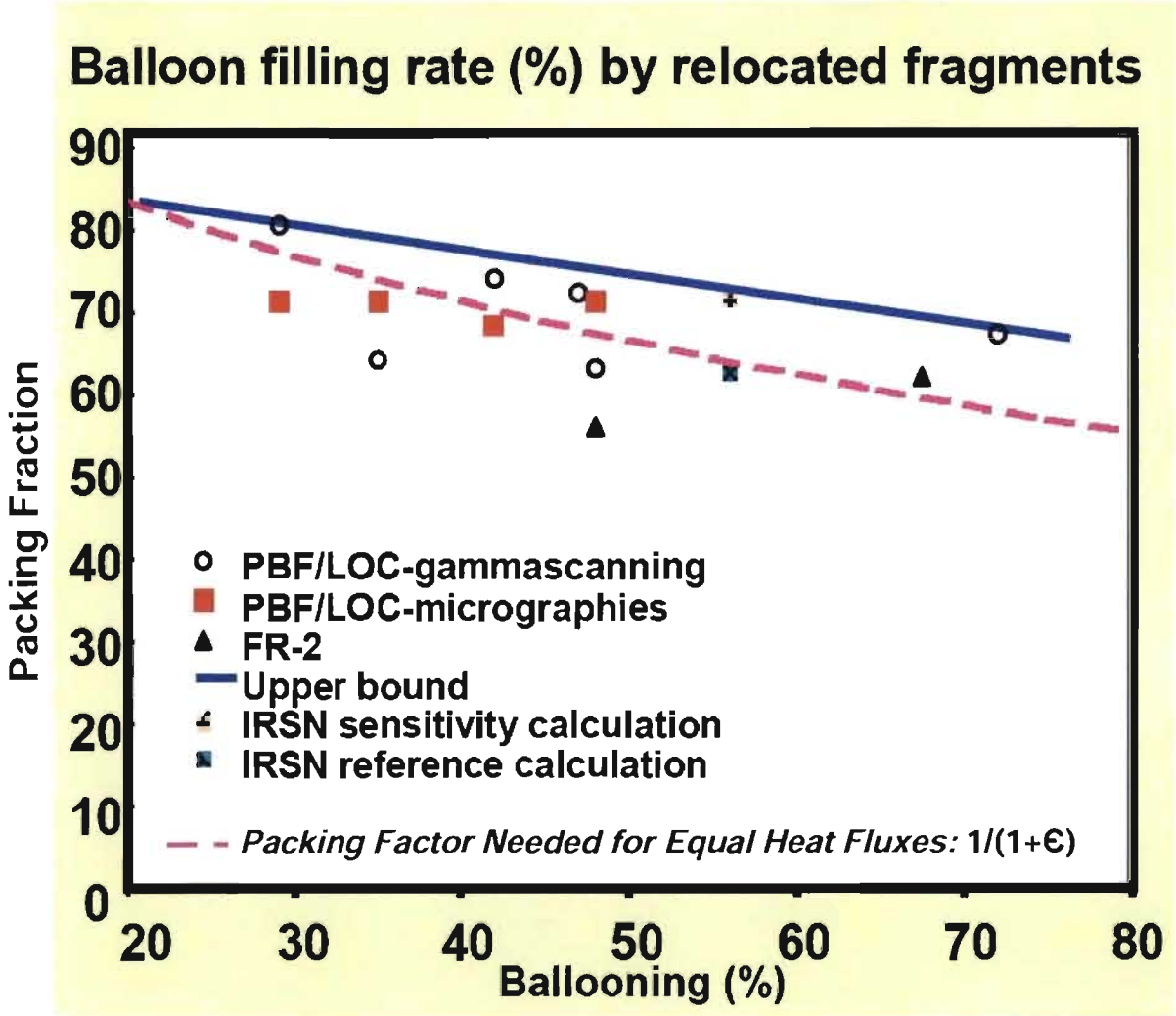
Question #1
Maximum Packing Factor [PF] Considerations

PBF – Power Burst Facility

- Only 1 gamma scanning data point at 80% packing fraction and it is at 30% strain
- Data results are in question due to material movement during the handling of the test rods
- Micrographies measurement is more accurate compared to gamma scanning
- PBF micrographic data shows 70% PF for rupture strain below 50%
- AREVA estimates 45% PF for strains near 70%

Question #1
Maximum PF Considerations cont'd

Packing Fraction vs. Rupture Strain



Note: The balloon filling rate is interpreted as packing fraction.

Question #1
Maximum PF Considerations cont'd

Test	Measured PF	Comments
FR-2 (E-5)	61.5%	Test maximized rupture strain and had very small rupture opening
Halden IFA-650.4	53%	92 GWd
Halden IFA-650.9	Not measured, but observed as similar to 650.4 (NEA report)	~ 90 GWd
Halden Test IFA-650.10	No relocation (no strain)	60 GWd

Question #1
Maximum PF Considerations cont'd

- Studsvik Tests – Provide results for burn-ups at 70 GWd, show no fuel in the ruptured region as it was all lost out of the rupture

- KfK/FR2 Tests – Only rod E5 showed a PF of 61.5%. During the swelling and rupture of the rod, the cladding expanded to make a seal around the ID of the container tube, thus invalidating any other results

Question #1
S-RELAP SRR Conservatism

Question #1
HNP Rupture Strain vs. Rupture Temperature

Question #1
S-RELAP Faster Cladding Heatup

Question #1 Summary

- 50%, as in ANP-3011P, is an appropriate PF based on the Halden test results, which are the most reliable information available
- 70% PF is an upper limit (based on older PBF data) for the strain range seen in the HNP plant cases
- Sensitivity study concludes base case is conservative and no bias is proposed (with droplet shattering)

Question #2
Droplet Shattering Model

Address whether droplet shattering is calculated on all flow blockage (non-vertical) surfaces in the S-RELAP5 calculation. If not, provide the flow blockage surfaces which are assumed to cause droplet shattering.

Question #2
Droplet Shattering Model cont'd

Question #3

Page 122 of ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis" states, In the present model, the rupture blockage ratio [which is correlated to the number of droplets to yield a maximum atomization factor], Є, is taken from the swelling and rupture correlation.

- Address whether droplet shattering is calculated only against the hot pin rupture, or the additional flow blockage areas (i.e., balloon/burst regions, spacer grids, etc.) assumed to be present upstream of the hot pin rupture location.
- If the additional flow blockage areas are not based on pre-transient core geometry, discuss how the locations and sizes of flow blockages are distributed.

Question #3 cont'd

Question #3 cont'd

Question #3 cont'd

Question #4
Benchmark to Swell/Rupture [SR] Data

On page 123 of ANP-3011(P), it is stated, It can be seen that the code predicted the peak cladding temperature variation well. The data is so tightly clustered that the degree of agreement is difficult to ascertain. Please tabulate the data to provide a more quantitative indication. Address how well the S-RELAP5 modification predicted the data.

Question #4
Benchmark to SR Data cont'd

- An Excel spreadsheet that contained blockage Test 61607 data has been placed in PGN FTP site for NRC to retrieve for further evaluation
- Figures 6-14 and 6-15 in ANP-3011P show S-RELAP5 conservatively predicted PCT compared to blockage test data for the FLECHT-SEASET and REBEKA-6 tests
- Additional FLECHT-SEASET test benchmarks, 61509 and 61607, are presented on the next two slides
- COBRA-TF (NUREG/CR-4166) modeled all expected multi-dimensional flow phenomena at rupture location, including flow diversion
- The figures show that S-RELAP5 conservatively predicts cladding response compared to the COBRA-TF results

Question #4
Benchmark to SR Data cont'd

Question #4
Benchmark to SR Data cont'd

Question #5
Application of Sugimoto/Murao correlation

Comparative data demonstrate the global effects of the droplet shattering phenomena; however, the correlation as implemented discriminates between large and small droplets and the behavioral differences between the two. Validate droplet size distribution as implemented in model. Explain how the Sugimoto/Murao correlation applies to the scenario in which it is applied.

Question #5
Application of Sugimoto/Murao Correlation cont'd

Question #6
Droplet shattering impacts on Heat Transfer

Explain how droplet shattering model incorporates the following droplet-dependent heat transfer effects:

- Inter-phase heat transfer
- Fluid-structural interactions including cladding, balloon, and spacer heat transfer to coolant
- Validate heat transfer modeling for these separate effects

Question #6
Droplet shattering impacts on Heat Transfer cont'd

Question #6 cont'd

Question #6 cont'd

Question #6 cont'd

REBEKA-6 Test: Rupture rod cladding temp. & pin pressure

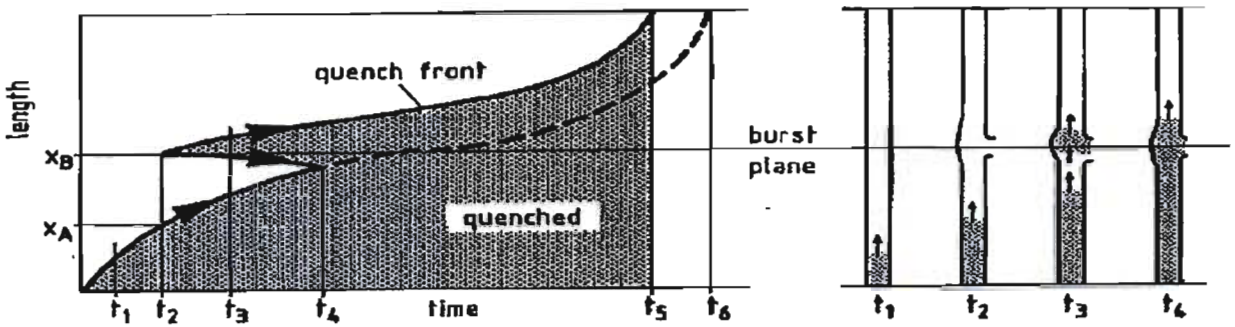
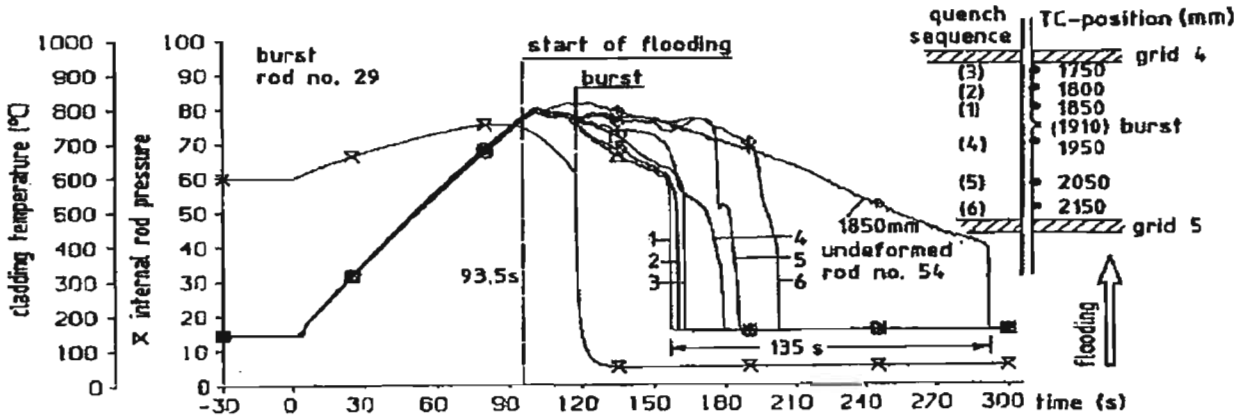


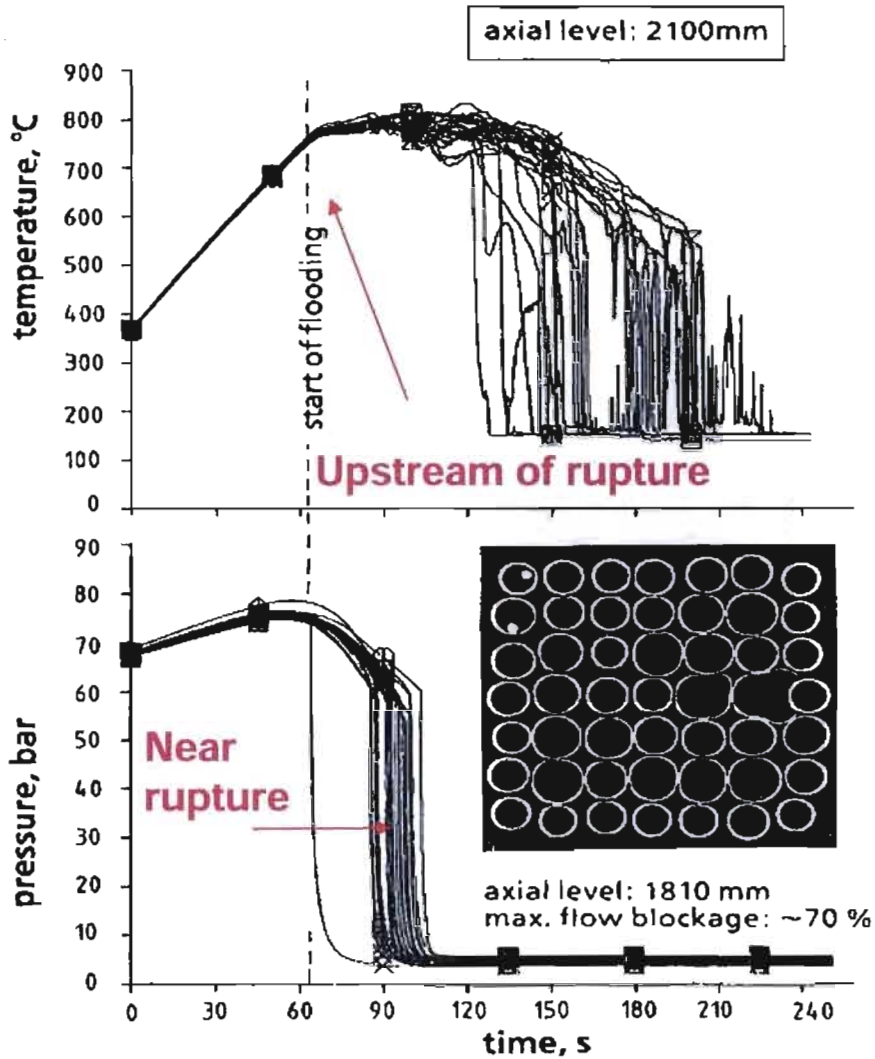
Fig. 7 Premature quenching of burst Zircaloy claddings (schematic)



Question #6 cont'd

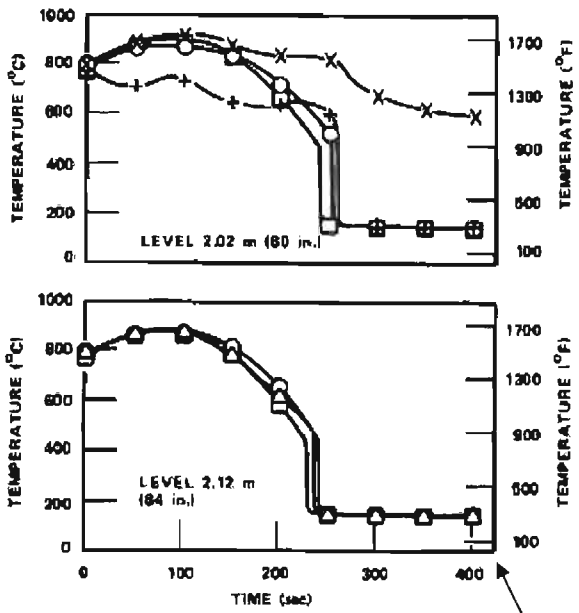
REBEKA-7 Results:

Maximum blockage ~ 70 %



Question #6 cont'd

FEBA Tests Results



FLOODING RATE = 0.038 m/sec (1.5 in./sec)
PRESSURE = 0.40 MPa (58 psia)

SERIES II:

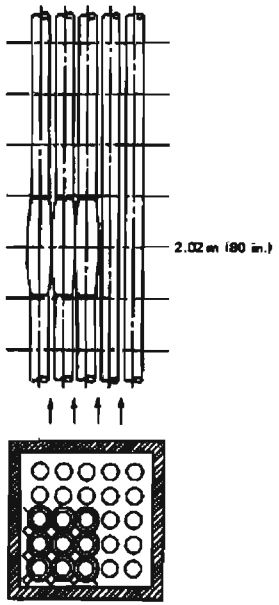
TEST 229
6 GRID SPACERS
UNBLOCKED BUNDLE

□ CLADDING

SERIES III:

TEST 239
6 GRID SPACERS
BLOCKED BUNDLE (3 X 3 RODS)
BLOCKAGE AT LEVEL 2.02 m (80 in.)
BLOCKAGE RATIO 80%

○ BYPASS REGION
△ BLOCKED REGION
+ SLEEVE
X UNDERNEATH SLEEVE



Upstream of blockage

Summary of Questions 2 - 6

- Droplet shatter model is implemented in a conservative manner
- Benchmarking demonstrates that sensitivity study model provides conservative predictions

Conclusions

- Best estimate to upper bound packing factor is 50% to 70%
- The response of PCT over a range of PF shows a smooth response to changes
- AREVA's sensitivity method has been successfully benchmarked against applicable research
- AREVA's treatment of the SRR phenomenon is applied in a conservative manner within ANP-3011
- A PCT of 1919 from base case is therefore defensible

New Round of Questions

Discussion of questions received on January 10th, 2012.

Schedule

- Progress Energy will docket the responses
- Projected docket date is January 20, 2012
- Target date for new round of questions to be docketed is the week of February 13th

Supplemental Slides

Supplemental Slides

Supplemental Slides Question #5

Supplemental Slides Question #5

Please direct any inquiries to me at 301-415-3302.

Sincerely,

/RA/

Araceli T. Billoch Colón, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Discussion Topics
2. Licensee's Presentation
3. List of Attendees

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