September 13, 2004

MEMORANDUM TO: Laura A. Dudes, Section Chief
New Reactors Section
New, Research and Test Reactors Program
Division Regulatory Improvement Programs, NRR

FROM: John Segala, Senior Project Manager ***/RA/***
New Reactors Section
New, Research and Test Reactors Program
Division Regulatory Improvement Programs, NRR

SUBJECT: AP1000 THERMAL-HYDRAULICS TELEPHONE CONFERENCE CALL SUMMARY (MULTIPLE DATES)

Telephone conference calls were held with Westinghouse Electric Company (Westinghouse) representatives and Nuclear Regulatory Commission (NRC) staff on the following dates:

August 25, 2003
October 28, 2003
November 6, 21, 2003
December 4, 12, 18, 2003
January 8, 12, 15, 20, 22, 28, 30, 2004
February 3, 6, 9, 12, 25, 26, 2004
March 1, 2, 2004

The purpose of these calls were to discuss AP1000 issues related to draft safety evaluation report (DSER) open items (OI) 15.2.7-1, "Long Term Cooling and Boron Precipitation;" 21.5-1, "Confirmatory Sensitivity Analysis;" 21.5-2, "Upper Plenum Liquid Entrainment Testing;" and 21.5-3, "Core Level Swell." A list of participants for each meeting is included in Attachment 1. A list of the Westinghouse letters responding to the AP1000 draft safety evaluation report thermal-hydraulics open items is included in Attachment 2. NRC staff comments provided to Westinghouse are included in Attachment 3.

Docket No. 52-006

Attachments: As stated

MEMORANDUM TO: Laura A. Dudes, Section Chief September 13, 2004
New Reactors Section
New, Research and Test Reactors Program
Division Regulatory Improvement Programs, NRR

FROM: John Segala, Senior Project Manager ***/RA/***
New Reactors Section
New, Research and Test Reactors Program
Division Regulatory Improvement Programs, NRR

SUBJECT: AP1000 THERMAL-HYDRAULICS TELEPHONE CONFERENCE CALL SUMMARY (MULTIPLE DATES)

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March 1, 2, 2004

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Docket No. 52-006

Attachments: As stated

<u>Distribution</u>:
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ACCESSION NUMBER: ML042470041

OFFICE	RNRP:PM	RNRP:SC
NAME	JSegala	LDudes
DATE	09/8/04	09/10/04

AUGUST 25, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	M. Corletti
Y. Hsii	J. Scobel
W. Jensen	B. Kemper
	T. Schulz
	E. Cummins

OCTOBER 28, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
J. Starefos	K. Ohkawa
N. Lauben	E. Cummins
S. Bajorek	B. Brown
W. Jensen	C. Frepoli
Y. Hsii	A. Gagnon
L. Ward	M. Corletti
J. Uhle	R. Wright
J. Colaccino	B. Kemper
	T. Schulz

NOVEMBER 6, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Starefos	B. Brown
Y. Hsii	B. Kemper
L. Ward	R. Wright
	T. Schulz

NOVEMBER 21, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission

J. Starefos
E. Throm
S. Bajorek
W. Jensen
Y. Hsii
L. Lois
J. Uhle

Westinghouse

C. Frepoli
K. Ohkawa
B. Kemper
B. Brown
A. Gagnon
R. Ofstun
R. Wright
E. Cummins
T. Schulz
R. Vijuk

DECEMBER 4, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Starefos	R. Vijuk
W. Jensen	T. Schulz
E. Throm	R. Ofstun
Y. Hsii	R. Wright
S. Bajorek	
J. Uhle	

DECEMBER 12, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	E. Cummins
Y. Hsii	K. Ohkawa
L. Ward	C. Frepoli
J. Uhle	G. Saiu
S. Bajorek	
J. Starefos	

DECEMBER 18, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	E. Cummins
Y. Hsii	K. Ohkawa
L. Ward	C. Frepoli
	B. Brown
	A. Gagnon
	R. Wright

JANUARY 8, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	B. Brown
S. Bajorek	B. Kemper
J. Colaccino	T. Schulz
J. Starefos	E. Cummins
L. Dudes	C. Frepoli
E. Throm	K. Ohkawa
J. Uhle	A. Gagnon
L. Ward	

JANUARY 12, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	E. Cummins
L. Ward	T. Schulz
J. Colaccino	

JANUARY 15, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	R. Wright
S. Bajorek	B. Kemper
W. Jensen	T. Schulz
J. Starefos	E. Cummins
J. Uhle	C. Frepoli
E. Throm	A. Gagnon
L. Ward	

JANUARY 20, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	T. Schulz
Y. Hsii	
J. Colaccino	
J. Uhle	
L. Ward	

JANUARY 22, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	B. Brown
S. Bajorek	B. Kemper
J. Colaccino	T. Schulz
E. Throm	E. Cummins
W. Jensen	C. Frepoli
L. Ward	K. Ohkawa
J. Uhle	A. Gagnon

JANUARY 28, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	
S. Bajorek	
W. Jensen	
J. Uhle	
L. Ward	

JANUARY 30, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Colaccino	R. Vijuk
Y. Hsii	A. Gagnon
S. Bajorek	B. Kemper
W. Jensen	B. Brown
L. Ward	T. Schulz
L. Lois	E. Cummins
	C. Frepoli
	K. Ohkawa

FEBRUARY 3, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission

J. Segala
Y. Hsii
J. Uhle
L. Ward

Westinghouse

R. Vijuk

FEBRUARY 6, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
W. Jensen	J. Winters
J. Uhle	E. Cummins
J. Colaccino	

FEBRUARY 9, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission

J. Segala
Y. Hsii
S. Bajorek
J. Colaccino
W. Jensen
S. Bloom
E. Throm
J. Uhle
L. Ward

Westinghouse

R. Vijuk
M. Corletti
R. Wright
T. Schulz
E. Cummins
C. Frepoli
K. Ohkawa
A. Gagnon

FEBRUARY 12, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Colaccino	E. Cummins
Y. Hsii	T. Schulz
J. Uhle	R. Wright
L. Ward	

FEBRUARY 25, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
J. Uhle	T. Schulz
J. Colaccino	C. Frepoli
	A. Gagnon
	K. Ohkawa

FEBRUARY 26, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
J. Wilson	E. Cummins
J. Uhle	M. Corletti
J. Colaccino	

MARCH 1, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	M. Corletti
J. Uhle	B. Kemper

MARCH 2, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Segala	R. Vijuk
Y. Hsii	E. Cummins
J. Uhle	

WESTINGHOUSE LETTERS RESPONDING TO THE
AP1000 DRAFT SAFETY EVALUATION REPORT THERMAL-HYDRAULICS OPEN ITEMS

DSER Open Item	Westinghouse Response Letter	
	Date	Accession Number
15.2.7-1	08/04/03	ML032190557
	08/15/03	ML032310154
	09/08/03	ML032530326
	09/29/03	ML032760084
	10/06/03	ML032950139
	10/07/03	ML032820430
	10/13/03	ML031250168
	10/14/03	ML032900103
	10/31/03	ML033090106
	12/23/03	ML033630760
	12/31/03	ML040050443
	01/05/04	ML040080880
	01/09/04	ML040220575
	01/14/04	ML040160800
	01/16/04	ML040210451
	01/21/04	ML040260437
	01/26/04	ML040290529
21.5-1	08/04/03	ML032190557
	09/29/03	ML032760084
	10/28/03	ML033030412
	11/13/03	ML033210561
	12/31/03	ML040050443
	01/05/04	ML040080880
21.5-2	08/07/03	ML032230268
	08/13/03	ML032270388
	09/29/03	ML032760084
	10/07/03	ML032820430
	10/13/03	ML032901130
	12/22/03	ML040070211
	12/31/03	ML040050443
	01/02/04	ML040070082
	01/05/04	ML040080880
	01/09/04	ML040220575
	01/13/04	ML040150441
	01/14/04	ML040160800
	01/15/04	ML040210416
	01/16/04	ML040210451
	01/21/04	ML040260437
	01/28/04	ML040330261
	02/02/04	ML040400095
	02/06/04	ML040420120
	07/30/04	ML042160219

DSER Open Item	Westinghouse Response Letter	
	Date	Accession Number
21.5-3	08/04/03	ML032190557
	09/23/03	ML032690344
	09/29/03	ML032760084
	01/09/04	ML040220575
	07/30/04	ML042160219

AUGUST 25, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

OCTOBER 28, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

NOVEMBER 6, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

NOVEMBER 21, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

NRC staff comments provided to Westinghouse via e-mail on November 21, 2003 (ML040120802).

DECEMBER 4, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

NRC staff comments provided to Westinghouse via e-mail on December 2, 2003 (ML040120795).

DECEMBER 12, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

Thermal-hydraulic graphs faxed to Westinghouse on December 12, 2003 (ML042460041).

DECEMBER 18, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

JANUARY 8, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff comments (regarding Westinghouse's submittals dated December 22, 2003, December 23, 2003, December 31, 2003, and January 2, 2004) were provided to Westinghouse via e-mail on January 8, 2004:

DSER OI 15.2.7-1 Item 7:

Original Issue:

The revised DCD Section 15.6.5.4C (DSER OI 15.2.7-1P Page 14) states that the LTC phase analysis uses the NOTRUMP DEDVI case at 25 psia containment pressure reported in Section 15.6.5.4B as initial conditions, and the WGOTHIC analysis of this event as boundary conditions.

Please describe the model used to develop the containment backpressure and demonstrate that it represents a bounding and conservative estimate of containment pressure following a small break LOCA. Discuss any differences that may exist between this model and that used in the large break LOCA analyses. Please discuss how water spillage from a broken DVI line is mixed with the containment atmosphere and justify that the treatment is consistent with the Westinghouse ECCS evaluation model. Discuss the conservative treatment of non-safety related containment sprays and containment coolers in reducing containment pressure. Please also clarify if the 25 psia initial condition is consistent with the WGOTHIC analysis of the containment pressure as a function of time.

Remaining Issue:

In its December 23, 2003, submittal, Westinghouse provided its response (in Attachment A to OI 15.2.7-1 Item 7 Revision 2) to the NRC assessment of the AP1000 WGOTHIC containment backpressure calculation for a small break LOCA (SBLOCA). Attachment A provides the AP1000 containment backpressure sensitivity with the NRC recommendations in Figure A-1, and states that the NOTRUMP small-break LOCA response, and the WCOBRA/TRAC long term cooling response will be re-analyzed using the NRC case for the AP1000 DEDVI break.

1. Please clarify whether the SBLOCA and LTC analyses will use the time-dependent containment backpressure of the NRC case in Figure A-1.

2. Please explain why the containment pressure in the NRC case increases after 9000 seconds as shown in Figure A-1, and would containment pressure decrease in the long term.

3. As commented in the December 17, 2003 meeting, when the containment pressure issue is settled, Westinghouse needs to re-analyze long-term cooling performance including those aspects which could impact for the PRA using acceptable models.

DSER OI 21.5-2 Item 19:

Original Issue

As mentioned in the ACRS Meeting in Monroeville in July, 2003, the APEX test facility contains an oversized downcomer. The oversized downcomer will produce high liquid inventories for extended periods of time which will maximize the liquid and two-phase levels in the core and upper plenum. This suggests the APEX facility cannot be used to simulate the minimum liquid and two-phase levels in the inner vessel that could occur following small breaks in the AP1000 plant. With a larger downcomer, more liquid mass will be retained in the vessel for small breaks. The statements in the Westinghouse August 13, 2003 letter (DCP/NRC1611) that the APEX-1000 facility is well scaled to AP1000 and the two-phase level remains in the upper plenum while the core remains covered for all phases of the simulated accident may not be appropriate and is misleading.

Please discuss the impact of the larger downcomer on the relevant APEX tests and explain why the facility test results can be used to demonstrate that significant amounts of inventory in this facility apply to the anticipated AP1000 response. Please also explain the statement that the APEX tests show the insensitivity of the AP1000 system behavior to entrainment is unaffected in lieu of the excessive amounts of liquid in the inner vessel during the tests referred to in the August 13, 2003 letter.

Remaining Issue

In its December 22, 2003, submittal, Westinghouse provided Revision 2 of its response. Westinghouse concluded that the APEX facility is adequately scaled for downcomer inventory depletion relative to AP1000 during a potential situation in a SBLOCA where only the liquid inventory in the downcomer is available for cooling. This conclusion was based on its scaling analysis showing that the ratio of the downcomer drain time constant between the APEX test facility and the AP1000 to be approximated ½. However, this time constant ratio was based on a value of the core mass flow ratio between the APEX and AP1000, which the staff finds to be based on the results of the "Simple Model" as opposed to using the mass flow ratio for which the APEX-AP1000 facility was scaled. Using the core flow scale ratio of 1/96, the staff calculated the downcomer depletion time constant ratio to be close to a value of 1, which indicates that the APEX downcomer is oversized since the APEX facility was designed to operate with a ½ time scale. This conclusion is consistent with the report OSU-APEX-03001,"Scaling Assessment for the Design of the OSU APEX-1000 Test Facility," May 12, 2003, which indicated that the APEX downcomer is oversized.

1. Please provide an evaluation of the effect of oversized downcomer on the test DBA-2. That is, assuming the DBA-02 test was performed with the APEX facility having a properly sized downcomer, what would be the expected core collapsed liquid level compared to Figure 21.5-2.19 in the August 13, 2003, submittal.

2. Westinghouse needs to address the "APEX-AP1000 Scaling Report Questions" provided by the staff in the December 17, 2003 meeting.

<u>OI 21.5-2P Item 22</u>:

Original Issue:

For the NOTRUMP simulation of both DBA-02 and DBA-03 tests, Figures 21.5-2.19 and 21.5-2.44, respectively, show NOTRUMP over-predicts the core collapsed liquid level for the time period between about 100 to 400 seconds. Figures 21.5-2.20 and 21.5-2.45, respectively, indicate NOTRUMP under-predicts core average void fraction between the same time period. Westinghouse attributes the non-conservative NOTRUMP calculations to its lack of two-dimensional downcomer modeling and heating of DVI injection flow. The sensitivity study performed with higher injection temperature still show non-conservative, though reduced, NOTRUMP predictions (Figures 21.5-2.17 and 18).

Please explain why, with the same downcomer modeling deficiency, the NOTRUMP calculations of the core liquid level and void fraction are comparable with the tests after 400 seconds.

NRC Comment from 12/17/03 meeting:

Provide justification of the acceptability of NOTRUMP SBLOCA analysis in light of its over-prediction of core collapsed liquid level in the early portion of the transient and propagation effects on the later stage of DVI line break transient as seen in (1) the NOTRUMP simulation of the APEX test DBA-2 and DBA-3 test data, and (2) the NOTRUMP-RELAP5 comparison.

Remaining Issue:

In Attachment 21.5-2 Item 22 R1-1 of January 2, 2004, submittal, Westinghouse provided the justification for the acceptability of NOTRUMP SBLOCA analysis despite its over-prediction of core collapsed liquid level in the early portion of the event and propagation effects on the later stage of the DEDVI break transient.

1. The early ADS-1-2-3 blowdown period of the DVI line break transient is critical as it has the minimum core collapsed liquid level (CLL) indicated by the DBA-02 test. The over-prediction of the core CLL during this period by NOTRUMP should be carefully addressed. Please provide an analysis of the level swell for AP1000 for this early period of a postulated DEDVI line break making correction for the NOTRUMP over-prediction of core CLL to ensure adequate core cooling prediction.

2. Westinghouse needs to address the following regarding OI 21.5-2P Addendum in the August 13, 2003, submittal:

    A. The third paragraph on Page 2 of DSER OI 21.5-2P Addendum 1 listed two OSU APEX tests, DBA-02 and DBA-03. The description of these two test conditions regarding the single failure of the ADS-4 is not correct. The DBA-02 test should have an ADS-4 failure on the non-pressurizer side, and the DBA-03 on the pressurizer side. Westinghouse needs to clarify the error.

    B. In the last paragraph on page 3, there is a discussion on break modeling. Prediction of the break flow is important in analysis of a small break LOCA such as a DVI line break. The response states that the break flow is "slightly over-predicted." The scale on the figure is deceiving. The NOTRUMP vessel-side break flow rate over the first 100 seconds of the transient is roughly 30% greater than the data break flow.

C. On page 7, the NOTRUMP prediction of ADS-4 integrated mass flow is claimed to be "predicted reasonably" in Figure 21.5-2.25. However, the test data shown in that figure may not be correct. In Figure 21.5-2.25, the test data for DBA-02, given by the solid line, reaches 1000 lbm at 1200 seconds. In Figure 4 (On DSER OI 21.5-2P page 7), the DBA-02 results, shown by the dash line, are that the integrated mass is well below 1000 lbm including the time period to 1500 seconds. Westinghouse needs to clarify the discrepancy between Figure 4 and Figure 21.5-2.25. If the Figure 4 data is correct, then the NOTRUMP results are not "predicted reasonably", but significantly over-predict the data.

D. Figures 21.5-2.19 and 21.5-2.23 should be re-submitted with the curves clearly identified. It appears that NOTRUMP over-predicts the levels, but the scale and the way the curves are plotted do not allow this to be viewed.

E. On page 5, the discussion on collapsed and two-phase levels, two-dimensional effects and subcooled core inlet temperatures are claimed to occur in the data. NOTRUMP, however, predicts the core inlet fluid to be at saturation between 400 and 100 seconds. Since a collapsed level will swell to a higher level if it is saturated, why does this not suggest a significant non-conservatism in NOTRUMP? Please explain.

F. The start of IRWST injection is significantly early in the NOTRUMP simulation of DBA-02. The claim on Page 7 that "NOTRUMP predicts IRWST flow reasonably well" is not justifiable. Early IRWST injection is non-conservative. Please explain why this discrepancy is considered acceptable in the NOTRUMP code's ability to predict the minimum vessel inventory.

<u>OI 21.5-2P Item 28</u>:

Original Issue:

The APEX test matrix contained a subset of tests expected to produce the lowest vessel collapsed liquid levels. The selection of these tests was in part based on results of NOTRUMP simulations. Westinghouse indicated that similar liquid levels were predicted for inadvertent ADS 1-3, 2 inch hot leg break and the DEDVI break. The DEDVI break is considered to be limiting small break LOCA and assessment has focused on this case. Please demonstrate the adequacy of the NOTRUMP simulation of inadvertent ADS 1-3 to ensure that the limiting break has been identified.

Remaining Issue:

In its December 31, 2003, submittal, Westinghouse provided the superficial gas velocity at the core exit as predicted by NOTRUMP for the AP1000 DEDVI break and inadvertent ADS actuation cases, and concluded shows that for a given backpressure, the DEDVI case is somewhat more limiting than the inadvertent ADS case.

The staff finds that showing comparisons for the superficial velocities are not sufficient to conclude which case is more limiting. Westinghouse should include comparison for certain dimensionless group that include superficial velocity and liquid height, etc., which are related to entrainment such as Froude number.

OI 15.2.7-1 Item 11:

Original Issue:

Post-LOCA Long-term cooling boron precipitation issues.

Remaining Issues:

1. As commented in the December 17, 2003 meeting, please provide the following information regarding long-term cooling boron precipitation: (1) the longest time for which the LTC is viable in terms of boron precipitation (i.e., identify the point in time entrainment no longer remove liquid through ADS-4 lines); and (2) the shortest reactor operating time which results in sufficient decay heat and steam generation to support boron removal by expulsion of liquid from the ADS-4 valves.

2. Please provide complete graphs of minimum containment pressure, minimum containment water level, and minimum and maximum containment water temperature as a function of time for 30 days, which the staff requested in the December 17, 2003, meeting.

3. Condensation on the outside of the ADS-4 pipe will collect boric acid on the inside of the ADS-4 piping. Westinghouse is requested to provide a technical justification why this will not happen including the time period the sump level is increased where the colder (minimum) temperature water is injected. Some boric acid precipitation tests have shown crystallization of boric acid on surfaces above the core exit exposed to steam flow.

JANUARY 12, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

JANUARY 15, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff comments (regarding Westinghouse's response to DSER Open Item 15.2.7-1, Item 7) were provided to Westinghouse via e-mail on January 14, 2004:

In Attachment B to the Westinghouse January 9, 2004, letter, on Open Item 15.2.7-1, Item 7, Westinghouse provided the containment sump water level as a function of time during the long-term cooling phase following a small break LOCA. The staff requests that Westinghouse justify the validity of the WGOTHIC calculation of the containment sump level during a LOCA and long-term cooling by responding to the following items.

1. Please describe how the WGOTHIC code (1) calculates containment water and vapor inventories during the transient, accounting for mass and energy inputs from the break and other sources, mass transfer from inter-compartment connections, leakage, condensation, and flashing; and (2) handles inter-compartment connections and cross-sectional areas at various elevations. To facilitate the staff's review of the code's capabilities in the sump water level calculation, please also provide a road map to the GOTHIC qualification study that address sump analyses, if they exist.

2. Please justify the validity of the geometric inputs to WGOTHIC in the AP1000 safety analyses, including inter-compartment connections and cross-sectional areas at various elevations that reflect the AP1000 plant containment layout, including the arrangement of various compartments, piping and equipment.

3. Please describe how the ITAAC reflects the AP1000 containment layout, piping and equipment that are the basis for its safety analysis inputs.

JANUARY 20, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff comments on Westinghouse's January 2, 2004 response to Open Item 21.5-2 Item 22 (regarding the possibility of a mass conservation error in NOTRUMP) were provided to Westinghouse via e-mail on January 15, 2004:

Another way to understand the excess mass in NOTRUMP in DBA-02, is to consider the system wide mass balance. At times in this transient, the mass variations are not consistent. For example, consider the period from 950 to 1150 seconds in the NOTRUMP DBA-02 simulation. This is reported in WCAP-15644, Rev 1. By way of reference, the RPV inital mass in APEX is approximately 800 lbm.

During this period ($950 < t < 1150$) flow from DVI-2 has stopped, so there is no injection to the vessel. The RPV mass remains constant during this period (Figure E-23). However, nearly 200 lbm is lost through the ADS-4 (Figure E-24). A small amount, maybe 25 lbm is lost out the break (Figure E-25). The pressurizer level is dropping during this period. Estimating the level decreases 2 feet (Figure E-5), and using the as-built APEX pressurizer ID of 11.91 in, this represents a mass addition of approximately 90 lbm. The flow in/out the ADS-123 is not shown, but should be 0.0. The SGs have long since drained, and as noted, there is no flow from the DVI.

This means the system lost 135 lbm, with no change to the vessel mass. Where did all this mass (roughly 16% of the total that the vessel can hold) come from? The cold legs should be empty. It may be from the hot legs, which are not shown, but it does not seem reasonable.

JANUARY 22, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

JANUARY 28, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following ACRS outstanding issues/commitments regarding thermal-hydraulics were provided to Westinghouse via e-mail on January 26, 2004:

1. Westinghouse agreed to provide the staff(and the Committee) with copies of its scaling evaluations that demonstrated that the AP600 CMTs were oversized, and that the AP1000 CMTs were not undersized.

2. The staff should carefully review the C-Y hand calculations of liquid entrainment that were used to validate the level swell behavior that Westinghouse has described.

3. Westinghouse will provide the staff with a comparison of the simplified model of IRWST injection flow, using the Yeh correlation, to one using the drift-flux model.

4. The staff should look at the scaling distortion between APEX1000 and the AP1000, considering the comments and observations made by Dr. Bannerjee about scaling the CMT flows, and the core flow resistance.

5. The staff will be reviewing Westinghouse's use of containment backpressure carefully.

6. Westinghouse should provide the staff with a clear, written description of the overall methodology presented during this meeting. It should include a clear statement of how the analyses fits into the other arguments, such as the C-Y correlation, the simplified model, and the NOTRUMP licensing calculations. The documentation should include more that one scenario.

7. Westinghouse should provide plots of the core mixture collapsed level for the USO DBA-02 test.

8. Westinghouse should keep looking at the 2-inch CLB and the inadvertent ADS cases, and should show that the collapsed level is sufficient to cool the core.

9. Westinghouse should use their C-Y method to predict the level swell data described by Dr. Bajorek, and demonstrate that they can predict that data.

10. The staff should continue to evaluate the results from RELAP5 to the calculations provided by Westinghouse, in order to better understand the differences.

11. Westinghouse agreed to determine what sort of containment sump screen blockage would be needed to cause the water level in the core to drop enough to cause high quality steam to flow out of the ADS4 line.

12. Westinghouse agreed to investigate whether there are any phase diagrams for water/steam/boric acid.

JANUARY 30, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff draft thermal-hydraulic review strategy was provided to Westinghouse via e-mail on January 29, 2004:

Review of AP1000 SBLOCA Thermal-Hydraulics

This draft review strategy is subject to change based on further staff review. Although it does not currently describe all Westinghouse analyses and NRC independent analyses, additional descriptions will be provided next week.

Westinghouse's licensing strategy involves the following steps:

1. Identify the phenomena that are high ranked in AP1000 that were not high ranked in AP600;
2. Identify the most limiting transient;
3. Prove NOTRUMP can model these phenomena for the most limiting transient (DEDVI);
4. Run the full scale AP1000 using NOTRUMP;
5. Perform bounding analyses for phenomena that are not well modeled by NOTRUMP;
6. Demonstrate WCOBRA/TRAC can model LTC.

The independent analyses performed by NRC to confirm Westinghouse's conclusions are described, below.

The AP1000 design is almost identical to that of the AP600, which was previously licensed. The design review of the AP1000 focused on the differences between the AP600 and the AP1000. Through the PIRT process, phenomena that were not high ranked and/or acceptably modeled by the Westinghouse analysis tools for the AP600 but are high ranked in the AP1000 were identified. These phenomena are level swell and upper plenum and hot leg entrainment.

The NRC and Westinghouse have focused the review on the double ended guillotine break of the Direct Vessel Injection Line (DEDVI). This is because the DEDVI is the most limiting transient for core cooling. Although the hot leg break and inadvertent ADS 1-3 also produce similar collapsed liquid levels (CLLs), qualitatively, the loss of ½ of the injection capability out of the DVI break make this the most limiting break. In addition, the DEDVI break is the most affected by entrainment, as demonstrated by the scaling group for entrainment compared to the values of the scaling group for the other transients. Therefore, it was necessary for Westinghouse to prove that NOTRUMP could appropriately model entrainment during the DEDVI, since entrainment was not as important during the AP600 transients. In addition, since the DEDVI was the limiting break with respect to core cooling, NRC independently verified that core cooling was ensured.

Since there is less margin to core uncovery in the AP1000 than in the AP600, level swell is more important in the AP1000. Therefore, the review also focused on the ability of Westinghouse's analysis tools to simulate this phenomena. Westinghouse proved both NOTRUMP's and WCOBRA-TRAC's simulation ability by comparing to separate effects test over prototypic ranges of conditions for both the short term and long term cooling (LTC) phases, respectively. NRC verified W's conclusions by using different data and interfacial drag models.

APEX-1000 at Oregon State University was evaluated by Westinghouse to be appropriately scaled to the AP1000 to simulate all the important phenomena affecting the overall system response. However, it is a 1/4 height facility with some non-conservative distortions (oversized downcomer) and cannot be used to demonstrate that the system is safe. However, since all the phenomena are represented, the data are used to benchmark the analysis tools. Westinghouse used other data to benchmark the phenomena not appropriately modeled in the facility. NRC independently confirmed these conclusions. NRC also performed a top-down scaling review and proved that both upper plenum and hot leg entrainment are appropriately scaled at the facility. The NRC used these data to verify that NOTRUMP and WCOBRA-TRAC were not grossly underestimating entrainment.

NRC also requested that Westinghouse benchmark NOTRUMP against an NRC-sponsored beyond design basis test that resulted in core heat up. Westinghouse did so and confirmed that NOTRUMP adequately simulated the transient and predicted heat up. This further confirmed Westinghouse's claim that NOTRUMP can effectively simulate the system response in a conservative manner.

To prove the adequacy of the design, Westinghouse developed a full scale AP1000 model using the same modeling approaches as were used to develop the APEX-1000 input model, and simulated the full scale system response for the most limiting transients. NRC audited Westinghouse's calculation notebooks to confirm that the input model is accurate. NRC developed its own AP1000 input model and ran independent calculations with RELAP5, which was proven to adequately simulate the AP600 system response. To verify RELAP5's ability to simulate AP1000, RELAP5 was benchmarked against APEX-1000 data. In general it was shown to conservatively predict the system response. The design was shown to meet the regulatory acceptance criteria of 10 CFR 50.46.

NOTRUMP overestimates CLL In the early stages of the DEDVI transient at APEX. This deficiency was attributed to underestimating the heat transfer from the downcomer walls to the liquid. Since there is plenty of water in the upper plenum and the system is below the CCFL limit, and there is violent flashing and a flow rate of about 0.7 in/s through the core, Westinghouse is not using the CLL as the figure of merit to indicate that the core is cooled. Instead, Westinghouse is comparing the AP1000 heat flux to the critical heat flux value based on boiling length approach at appropriate conditions. NRC confirmed these findings using data from RELAP5. Therefore, no heat up is expected during the ADS-4 blowdown period.

Westinghouse modified its input deck to more accurately predict the CLL by increasing the condensation heat transfer rate in the downcomer and using the EPRI drift flux model. The transient was repeated and the same long term CLL was produced, verifying Westinghouse's claim that the level in the hot leg would balance to the equilibrium point regardless of how much water was injected. The principle was further verified by Bill Brown's simplified model. Therefore, Westinghouse justified that the overestimation of CLL in the early stages of the transient does not contribute to overestimation of CLL in the later stages of the transient.

NRC used the CLL from RELAP, which is slightly less than that produced by NOTRUMP, and swelled the level to determine the location of the two phase mixture. NRC then performed a heat up calculation using steam cooling above the two phase level to verify that no heat up is expected during the IRWST injection period and LTC stages of the transient.

Instead of further benchmarking WCOBRA/TRAC (NRC-approved code for LBLOCA analysis), W has performed independent analyses (Terry Schultz model) to investigate the effect of two phase pressure drop and level swell on the quality out ADS4. All cases demonstrate adequate liquid flow to preclude boric acid concentration and core cooling. NRC has performed independent calculations using RELAP5 and simplified models. The system performance is highly dependent on backpressure and sump level.

Westinghouse ensured that there are no means available to reduce the containment pressure below that necessary for successful system operation. Westinghouse identified the time at which power is reduced to the point where liquid cannot be entrained. This time must be identified in procedures and action must be taken to prevent boric acid concentration and allow cleanup activities. Westinghouse committed to sample boric acid concentration before cooling down the system for post-accident cleanup.

Two phase pressure drop out of ADS-4 is critical during all stages of the transient. NRC requested Westinghouse to compare the pressure drop modeling to data. NRC independently verified this analysis.

FEBRUARY 3, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff comments (regarding upper plenum (UP) and hot leg (HL) entrainment in the NOTRUMP simulations of DBA-02 and NRC-05) were provided to Westinghouse via e-mail on February 3, 2004:

Comments on NOTRUMP Liquid Entrainment Capability Based on APEX-AP1000 Data

Note: Both tests are DEDVI breaks. In DBA-02 one ADS-4 of 2 valves on the non-pzr side is failed closed, while in NRC-05 both are failed closed. In both tests, both ADS-4 valves on the pressurizer side are open. In both tests the ADS-4 opens at approximately 250 seconds.

To estimate the how well NOTRUMP is calculating entrainment, prediction of the ADS4 flows in each case were examined. Flow quality is used because it frequently occurs explicitly in two-phase pressure drop calculations.

Description

NRC-05: NOTRUMP appears to have some type of compensating error over the period 250-400 seconds. In the data, there is a rapid expulsion of liquid just after ADS4 opening. The flow quality is low for a brief period and then increases to about 0.5. NOTRUMP does not recognize the two different periods. NOTRUMP expels liquid in a series of large, fairly regular slugs such that the integrated liquid flow is linear. The flow quality is 0.3 - 0.4, throughout the 250-400 second period. Thus, what NOTRUMP does is underpredict the liquid flow out the ADS4 for a while and then underpredict it so that by about 600 seconds the total liquid out the ADS4 in NOTRUMP and the data are in agreement.

Between 400 and about 800 seconds, the vessel mixture level is between the top and bottom of the hot leg. The hot leg levels show that the NOTRUMP water level is slightly lower than the middle of the pipe. During this period,. both the W and NRC data evaluations show a flow quality out the ADS4 of 0.5 - 0.7, while NOTRUMP is underpredicting the quality (and thus overpredicting the liquid flow).

After about 900 seconds, liquid flow to the ADS4 stops in the NOTRUMP calculation. In the data however, liquid flow clearly continues. In the data, the flow quality in the ADS4-4 remains nominally x = 0.6 while the two-phase level drops towards the upper core plate. Even when the level drops below the core plate and a temperature excursion occurs, liquid flow to the ADS4 continues. The flow quality increased to about 0.9 before the test was aborted. NOTRUMP fails to predict the upper plenum entrainment process. Rather than entraining liquid, NOTRUMP passes only steam.

DBA-02: As in the NRC-05 simulation, NOTRUMP appears to have some type of compensating error over the initial blowdown period (250-400 seconds). In the data, there is a rapid expulsion of liquid just after ADS4 opening. The flow quality is low for a brief period and then increases to about 0.5. NOTRUMP does not recognize the two different periods, and the quality to the ADS4 lines remains nearly constant.

During the period 400-800 seconds, the vessel level is in the hot legs. Flow quality to the ADS4 in the NOTRUMP calculation is higher than that found from the data in both vent paths.

During the period 1000 to 1200 seconds, the two-phase level in the vessel drops slightly below the bottom of the hot legs. (Figure E-21 shows the two-phase level never drops more than about 0.5 ft. below the bottom of the hot leg (5.38 ft.)). In both ADS4 vent paths, liquid flow continues. In ADS4-1 however, the NOTRUMP calculation shows steam only. In ADS4-2, liquid flow continues, but the flow quality is higher than in the data.

Conclusions

NOTRUMP appears to have a deficiency in predicting upper plenum entrainment. In both cases as the vessel mixture level drops below the bottom of the hot leg, the flow quickly changes to nearly all steam. This is in contradiction with the data, which shows significant liquid carryover even when the two-phase level drops to the upper core plate. NOTRUMP misses this process badly.

When the mixture level in the vessel is between the top and bottom of the hot legs, NOTRUMP appears to be reasonable but not necessarily conservative. In DBA-02, NOTRUMP slightly overpredicts the flow quality in the ADS4 lines, while in NRC-05 it underpredicts it. (It is difficult to tell from the figure supplied, but NOTRUMP may be overpredicting the mixture level in NRC-05. This would contribute to predicting a lower flow quality than the data.)

In addition, carryover to the ADS4 may be subject to a compensating error in the NOTRUMP calculation. Very early in ADS4 blowdown, the liquid flow is underpredicted (see for example comparison of integrated liquid flow to ADS4-2 in either DBA-02 or NRC-05). Later, the data shows a transition to higher flow quality in the ADS4. NOTRUMP appears oblivious to this transition.

FEBRUARY 6, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

FEBRUARY 9, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

FEBRUARY 12, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

FEBRUARY 25, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff comments (regarding the documentation of Westinghouse's small break loss of coolant accident evaluation model in Chapter 15 of the Design Control Document) were provided to Westinghouse via e-mail on February 24, 2004:

Long-Term Core Boron Concentration

Document how the analyses were done (equations, etc...), since it is part of the evaluation model that has to be repeated by a licensee if any change is made (or for a reload). The analysis is simplistic so its range of validity needs to be defined. In other words, there is uncertainty in the calculation but we accept it because there is plenty of margin to boron precipitation. What is the point at which the analysis is no longer reliable as the margin is reduced (boron concentration increases).

Provide applicable results of the analyses (boron concentration vs. time, whatever...).

In the text that says, "If the AP1000 operates in recirculation for an extended period of time...," the term "extended period of time" needs to be clarified. W is resolving the issue by relying on emergency response guidelines (ERGs) that the operator will take action within 1 hour. (The staff is comfortable with up to 3 hours). In this portion of the text, please reference the ERGs and also indicate the time in which the action must be taken. Show that this time has been determined by the analysis. This text needs to be shown to NRC before license approval, and has to be taken verbatim and put into the ERGs.

Small-break LOCA Analysis Methodology

15.6.5.4B.2.1 NOTRUMP Computer Code

The version of NOTRUMP used in the AP1000 ... test data (Reference 22). Add "The code was shown to be deficient in modeling upper plenum and hot leg entrainment and did not predict the collapsed liquid level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model and the critical heat flux assessment during the accumulator injection phase were relied upon to demonstrate the adequacy of the design."

"...the conservative failure of one o the ADS Stage 4 valves located off the PRHR inlet pipe, which adversely affects the depressurization necessary to achieve IRWST injection in small-break LOCAs, is assumed in all cases. Shouldn't it be the non-PRHR loop??

15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding

The use of FLOAD-4 and its validation for AP1000 ranges of conditions need to be described.

Keep the sentence that, "The ADS-4 resistance is computed for the NOTRUMP analyses in this section to be a 70% ADS-4 flow path resistance increase."

15.6.5.4B.2.2 NOTRUMP Homogeneous Sensitivity Model

You need to indicate that the DEDVI line is limiting for the effect of entrainment and that is why the sensitivity study need only be done to demonstrate that the lack of good modeling of entrainment by NOTRUMP does not affect the conclusion that the design is safe.

The technical evaluation ie, jg* needs to be discussed as demonstrating why you conclude DEDVI is limiting as compared to all other SBLOCA transients, including 10" CLB.

Also describe how you accounted for the lack of hydraulic head in the upper plenum by increasing the pressure drop by the particular factor – this has to be repeated for the evaluation model.

Indicate that you are running the homogeneous case to prove that entrainment does not matter and that the success criteria for concluding that entrainment does not matter is that the hot rod does not heat up to 2200 F and 17%. If it uncovers and starts to heat up, then a heat up calculation has to be done for the hot rod using an applicable heat up code that is approved by NRC for conditions prototypic of the range of AP1000.

15.6.5.4B.2.3 CHF Assessment During Accumulator Injection

Indicate that NOTRUMP has been shown to calculate G and P well and that is why we trust its output and can use it as input to the Chang CHF model.

Describe why Chang applies to AP1000 ranges of conditions shown in Table.

Indicate that even if heat flux exceeds CHF, it is not a safety issue because 2200 and 17% oxidation are the criteria. However, conservatively your acceptance criteria for demonstrating plant safety is that CHF is not exceeded during this period.

The results Table needs to be repeated in Section 15.6.5.4B.3.8, the results section. That is defining what information needs to be submitted by the licensee when using the evaluation model.

15.6.5.4B.3.1 Introduction

CHF calculation for accumulator injection period needs to be mentioned as part of the evaluation model.

FEBRUARY 26, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

MARCH 1, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

No comments provided.

MARCH 2, 2004
TELEPHONE CONFERENCE CALLS SUMMARY
COMMENTS

The following NRC staff proposed markups (regarding the documentation of Westinghouse's small break loss of coolant accident evaluation model in Chapter 15 of the Design Control Document) were provided to Westinghouse via e-mail on March 1, 2004:

The NRC staff proposes that the only portion of the evaluation model that needs to be Tier2* is under the critical heat flux assessment and under the homogeneous sensitivity study and are bolded below. The rest can be controlled via 10 CFR 50.46 because any change can be compared to a peack cladding temperature change of 50 °F.

DCD Section 15.6.5.4B.2.2, "NOTRUMP Homogeneous Sensitivity Model"

In order to address uncertainties associated with entrainment in the upper plenum and hot leg following ADS-4 operation, a sensitivity study is performed with the limiting break with respect to these phenomena, effectively maximizing the amount of entrainment downstream of the core. This methodology is described and the results are presented for the double-ended direct vessel injection (DEDVI) line break in detail in Reference 24.

[In order to maximize the entrainment downstream of the core for the limiting break with respect to entrainment, NOTRUMP is run with the regions of the upper plenum, hot leg and ADS-4 lines in a homogeneous fluid condition, with slip = 1, to demonstrate that even with maximum entrainment the 10 CFR 50.46 criteria are met]. All other features described in the previous section are retained.

DCD Section 15.6.5.4B.2.3, "Critical Heat Flux Assessment During Accumulator Injection**"**

[An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line. This time period corresponds to the accumulator injection phase of the transient. The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are used as input parameters to critical heat flux correlation as described in Reference 30. The requirements of 10 CFR 50.46 are met provided the maximum heat flux is less than the critical heat flux calculated by the correlation]. NOTRUMP has been shown (Reference 24) to adequately predict mass flux and pressure for integral systems tests.

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