March 16, 2004

APPLICANT: Westinghouse Electric Company

PROJECT: AP1000 Standard Plant Design

SUBJECT: SUMMARY OF DECEMBER 17, 2003, CATEGORY 1 MEETING WITH WESTINGHOUSE ELECTRIC COMPANY TO DISCUSS REMAINING OPEN ITEMS ASSOCIATED WITH THE AP1000 DESIGN CERTIFICATION REVIEW

On December 17, 2003, a public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of Westinghouse Electric Company (Westinghouse, the applicant), at NRC Headquarters in Rockville, MD. The purpose of this meeting was to discuss the remaining open items in the AP1000 design certification review. A list of meeting attendees is included as Attachment 1. Attachment 2 includes a summary of the open item issues as stated in Chapter 1 of the AP1000 draft safety evaluation report (DSER). The NRC staff provided additional comments to open items involving probabilistic risk assessment (PRA), materials, and fire suppression issues. These comments are included as Attachments 3, 4, and 5.

Below is a summary of each of the significant topics discussed at the public meeting.

Combined License (COL) Action Item Issues

The NRC staff stated that it was in the process of reviewing the COL action items in the AP1000 design control document. Attachment 6 contains a summary of issues that the staff has identified concerning COL action items. The staff further stated that the staff's review was not complete and anticipated that additional discussions would be necessary to discuss any remaining COL action item issues that were identified.

Summary of Structural/Seismic Audit

The NRC staff performed an audit of AP1000 design calculations from Monday, December 15, 2003, to Wednesday, December 17, 2003. During the public meeting NRC staff discussed with Westinghouse their findings during the audit.

A presentation given by Westinghouse at the start of the audit and has been added to the Agencywide Documents Access and Management System (ADAMS) ML040640564.

Reactor Systems Open Items

The NRC staff and Westinghouse discussed the following questions related to reactor system open items.

1. Provide justification of the acceptability of NOTRUMP small-break loss-of-coolant accident (SBLOCA) analysis in light of its over-prediction of core collapsed liquid level in the early

portion 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, (2) the NOTRUMP-RELAP5 comparison.

Westinghouse intends to provide a revised response that includes this period of analysis before January 1, 2004. The NRC staff discussed the effects of mass deficit and focus of coolability (real success path). Westinghouse agreed to rule out core heatup. The NRC staff also pointed out that they wanted a description on why NOTRUMP was that far off so that the staff can provide an evaluation on other analyses, which Westinghouse agreed to provide.

2. Provide complete graphs of minimum containment pressure, minimum containment water level and maximum water temperature as a function of time for 30 days.

   The staff calculations show that the ability to cool the core is highly dependant on containment pressure, containment water level, and sump temperature. Westinghouse has provided us with containment pressure only for the first 10000 seconds and then at 14 days and 28.5 days, and the containment water level only at 2.6 hours, 14 days and 28.5 days.

   Westinghouse stated that they would provide the requested graphs. They will perform long term cooling (LTC) analyses to provide a bigger picture of the behavior of fluid. The NRC staff stated that they wanted to see the dependence on containment pressure and sump water level in the analyses of LTC. Westinghouse did not believe this would be difficult to do.

3. Address inadvertent containment spray either from equipment failure or operator error during the 30 day period. Of particular concern is a sufficiently reduced containment pressure that increases the ADS-4 resistance and depresses the two-phase level into the vessel. If the vessel two-phase level is depressed below the top of the hot legs, liquid flow out the ADS-4 valves will decrease causing the boric acid to begin accumulating in the vessel.

   The NRC staff stated that they were interested in the dependence on back pressure. This was discussed at the last conference call. The NRC staff was concerned with the probability of no operator action, the sprays will initiate. Westinghouse stated that this is not possible but if it was not discussed in an earlier response, they will address it.

4. Respond to containment pressure issues.

   The NRC staff provided an assessment of uncertainty of modeling assumptions. The staff asked about areas to describe passive core cooling system. The staff stated that this needs to be included in the inspection, test, analyses, and acceptance criteria (ITAAC). The staff stated that in the AP 600 DCD, Westinghouse elected to take this out. Westinghouse came up with 12 recommendations, and are waiting for sensitivity runs before sending in response with revised containment back pressure analysis. Westinghouse will follow up with revised NOTRUMP and COBRATRAC analysis.

5. Once the containment pressure issue is resolved, the NRC staff requested Westinghouse redo long term cooling analyzes including those for the PRA using acceptable models.

   Westinghouse stated that this was related to question 4 and would redo the analyses.

6. 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; and (2) the shortest reactor operating time which results in sufficient decay heat and steam generation to support boron removal.

   Westinghouse stated that they would provide the response the first week of January.

7. Westinghouse response to 'Item 19' concluded that the APEX test facility is adequately scaled for downcomer inventory depletion relative to the AP1000 during a potential situation in an SBLOCA where only the liquid inventory in the downcomer is available for core cooling. This is inconsistent with OSU scaling report (OSU-APEX-03001 on page 6-7), which states that the APEX facility downcomer is oversized relative to the AP1000. Westinghouse needs to explain the inconsistency.

   Westinghouse stated that the response does not say that the APEX facility is oversized. The NRC staff stated that they thought that Westinghouses response was inconsistent with APEX scaling report. Westinghouse took another look at the scaling report and stated that they may need to have the OSU report revised. The NRC staff stated that Westinghouse needs to come up with conclusion for the ideal volume scaling since this did not match ideal scaling. Westinghouse stated that OSU did not do evaluation to assess implication of how fast does it drain. The NRC staff stated that they were worried about head of water. Westinghouse stated that the APEX facility area is distorted with respect to ideal facility. NRC staff stated that with the available head in the 20 percent region, the evaluation sees levels dropped in core. Westinghouse stated that they needs to focus on item 19 and will provide the appropriate argument to scaling and then will revise OSU report. Westinghouse will be revising open item response. NRC discussed the power to mass flow, and Westinghouse stated that this was discussed in the last workshop. Westinghouse stated that this was a long variation of the scaling group and that power was not in term. Westinghouse stated that in order to come up with power they used a simplified model. NRC staff will review the revised response, and Westinghouse will reconcile the two.

8. Westinghouse's response to the staff's comments ('Item 28') on the selection of the APEX test matrix and NOTRUMP simulations is insufficient. Westinghouse merely provided comparisons of the core collapsed liquid levels of the NOTRUMP simulations of the APEX-600 tests of double-ended direct vessel injection (DEDVI) and inadvertent ADS, and the NOTRUMP simulation of APEX-1000 DEDVI tests. Westinghouse concluded that this provides confidence that the NOTRUMP analysis of the AP1000 SBLOCA spectrum, including the inadvertent ADS case, are adequate.

   Westinghouse should provide comparisons of the relevant dimensionless groups for the inadvertent ADS case and DEDVI case to demonstrate the NOTRUMP simulation is adequate in identifying the limiting break for AP1000.

Westinghouse stated that this was discussed at the workshop. The NRC staff stated that the focus is on a DEDVI line break with inadvertent ADS 1, 2, and 3, which will produce the same inventory. Westinghouse stated that DEDVI break is the most limiting. The NRC staff stated the resolution would have to take a look at the scaling groups. Westinghouse stated that the phenomenon and DEDVI break governs. NRC staff stated that the response did not have scaling group. Westinghouse stated that they sent in revision 1 of item 28, and the NRC staff stated that they will review.

9. Westinghouse proposed passive residual heat removal (PRHR) acceptance criteria with two heat exchanger (HX) heat transfer rates and the HX elevation in ITAAC, Revision 8, are acceptable. However, Westinghouse should ensure that the numerical values of heat transfer rates in the acceptance criteria bound the calculated PRHR HX heat transfer rates in the safety analyses.

   Westinghouse stated that they don't need mass flow rate, and that revision 8 is acceptable. The NRC staff asked Westinghouse to ensure that the numerical values of heat transfer rates acceptable. Westinghouse stated that they performed tests and used test results to calculate what rates are in the design basis accident. Westinghouse stated that the in-containment refueling water storage tank (IRWST) temperature is not the design limit. The NRC staff asked if the number is consistent with safety analysis and Westinghouse stated that it was. The NRC staff wanted to see PRHR operated over the values. Westinghouse stated that they could not do this so they used the temperatures that they did. The NRC staff will evaluate the ITAAC.

10. Provide errata on Test Acceptance Reports OSU-AP1000-04 and -05 (Items 25 & 26):

    In OSU-AP1000-04, Table 4-1, assumed single failure of one ADS-4 valve on the non-pressurizer side.

    In OSU-AP1000-05, Table 4-1, ADS-1, 2, and 3 valves were open during the test.

    Westinghouse will reissue the document.

11. Revise WCAP-15644, Rev. 1, in the following areas (per Items 16, 17, and 29):

    Correct Figure F-12 (i.e., Figure 21.5-1.12) for ADS-4 Liquid Discharge Comparison.

    Correct text and Figure F-14 (i.e., Figure 21.5-1.14) to show Downcomer Pressure Comparison.

    Correct the equation for critical bubble radius R-bcr of the Cunningham-Yeh correlation on Pg. G-4.

    Westinghouse will reissue the document. The NRC staff also stated that there was an errata on page 21.5-2 in addendum 1, and in the first part of DVA-2 and DVA-3 the pages were reversed. Westinghouse will revise.

The NRC provided Westinghouse with questions related to APEX-AP1000 Scaling Report (ADAMS Accession No. ML040640981). Westinghouse stated that they will address these questions in a future submittal.

Dose analysis

The NRC staff stated the dose analysis is the only open issue and that the final information was just given to the contractor with the final factors. The review should be done by the end of January. Open Item (OI) 15.3-2 not affected since it is related to the atmospheric dispersion factors. The NRC staff will mark as confirmatory. OI 6.4-1 also is related to the atmospheric dispersion factors, so the NRC staff may need additional discussions.

Materials issues

There are no chemical engineering issues, however there are a number of issues related to material engineering. The NRC staff and Westinghouse discussed input to the FSER.

OIs 3.5.1.3-1 and -2, related to turbine missiles were submitted in some previous correspondence and Westinghouse stated that the design control document (DCD) has been revised and this issue has been resolved.

OI 4.5.1-1, Attachment 4 needs to be more specific with respect to pre-service inspection of the upper head penetration. Westinghouse will send in a revised response in a few days.

OI 4.5.1-2, Correspondence has resolved this issue, however, the NRC staff needs to see revision 8.

OI 5.2.3-1, DCD revision has been submitted and the response is acceptable, so the evaluation input will be written.

OI 5.3.3-1, vessel closure flange required pressure/temperature curves need to be revised. This item will be confirmatory until revision 8.

OI 5.4.2-1, steam generator (SG) elastic stability item was closed out in the July 3, 2003, correspondence. This letter is not proprietary.

OI 6.1.1-1, hydrogen generation, related to 50.44, item has been resolved and closed out.

OI 10.2.8-1, turbine rotor integrity, item has been resolved (also -2 and -3).

OIs 14.3.2-9, 10, and 11, resolved based on letter input, and did not require DCD changes.

Chapter 20 issues, revised some COL action items and closed out.

Bulletin 2001-01, which is related to OI 4.5.1-2, needs to be linked to this OI 20.7-1. OI 20.7-1, related to removal of insulation, needs to be related to an ITAAC. The OI has been resolved.

OI 4.5.1-2, Bulletin 2002-02, needs to have the DCD verified, to ensure that the writeup is acceptable.

With respect to Bulletin 88-05, on boric acid corrosion, Westinghouse needs to make a DCD revision instead of COL item. WCAP 15800 R1, for 88-05, says this is responsibility of COL applicant, however, NRC staff wanted to see in DCD a commitment that applicant will develop program. Westinghouse questioned where to put it in the DCD, possibly Chapter 5. The NRC staff thought that this was a reasonable location. This is an operational question, so it is not discussed in design space. Section 5.2.3 discusses materials in the reactor coolant system, so could indicate that materials are subject to degradation by boric acid corrosion. The NRC staff will discuss further with Westinghouse when necessary. The NRC staff will provide additional comments (document in summary) associated with COL action item 20.7-8. NRC may add to COL action item table.

Generic Letter (GL) 92-01 Revision 1, Supplement 1, item has been reviewed and rewritten, however, one more piece of material property data is needed related to the upper shelf energy.

GL 97-06, SG internals should have made an OI and have a committment to develop program for periodic monitoring. The NRC staff is looking for extra sentence in DCD. Westinghouse believes this will be answered in Revision 8.

OI 5.2.3-2, welding issues. There are 2 remaining items, and the December 12, 2003, response goes a long way to address these issues. The last paragraph of OI 5.2.3-2 discusses J-groove welds in the context of alloy 690. The NRC staff asked about the challenges of welding with 52/152. In the manufacturing process, they use post weld heat treatment. Westinghouse agreed, that to be technically correct they would revise the DCD to state that 52/152 welds need to be ultrasonically tested. Westinghouse suggested taking out the reference to alloy 690, which the NRC staff agreed with. Westinghouse will revise their response.

The second item dealt with the acceptance criteria for inspections specified by code and Westinghouse procedures. The acceptance criteria for lack of fusion in the code is zero tolerance (NB 5331 Section III). If while performing an ultrasonic test on these penetrations, and the licensee finds a lack of fusion, they will have to make a large number of repairs. Westinghouse participated in the Oconee inspection with Framatome using ultrasonic testing. The NRC staff stated that we were not sure if welding can be achieved without fusion. The NRC asked if Westinghouse would consider using 52 for first layer (buttering). The NRC staff was comfortable with Westinghouse using the code.

Leak before break open items being written.

OI 4.5.1-1, alloy 52/152, dealing with low temperature crack propagation. Westinghouse expects to submit their response this week, however, they are struggling with this response. Westinghouse has several references on issue and will provide a more detailed response on why there is no concern.

<u>Quality Assurance (QA)</u>

OIs 17.3.2-1 and 17.3.2-2, Westinghouse has sent in responses to the QA audit conducted in early November. In early December, a response was submitted on non-conformance violation (NCV) and contained a number of actions to be performed. The last action will be completed by the end of January. Westinghouse would like feedback on its response. The NRC staff stated

that they haven't received a formal response to findings. The December 6, 2003, NCV actions appear reasonable. Westinghouse stated that the NCV says that they did not follow procedures. Westinghouse stated that this was true, but that there are many ways to meet Appendix B. They stated that the basic requirements did comply with Appendix B, but because they did not follow their own procedures they were given a NCV. The NRC staff stated that they wanted to understand when actions were complete. Westinghouse will be revising OI 17.3.2-2 response when procedures are revised, which should be completed by the end of January.

For OI 17.3.2-1, Westinghouse plans on sending in letter on data which addresses OSU issues. This will demonstrate that the data is acceptable. The NRC staff requested that the DCD have a commitment to Appendix B, which Westinghouse stated that they will do.

Attachment 6 contains a summary of issues related to COL actions items. Attachment 7 contains a matrix of the status of all the DSER open items in the AP1000 design certification review as of the date of the public meeting.

Please direct any inquires concerning this meeting to Joseph Colaccino at 301-415-2753, or jxc1@nrc.gov.

/RA/

Steven D. Bloom, Project Manager
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 52-006

Attachments:
1. List of attendees
2. Remaining AP1000 Open Items
3. Comments on Certain AP1000 Open Items in DSER Chapter 19
4. Comments on AP1000 DSER Open Item 4.5.1-1
5. Comments on DSER AP1000 Open Items 9.5.1-1
6. Issues Related to COL Action Items
7. Summary of DSER Open Item Tracking List

cc w/ atts: See next page

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ADAMS ACCESSION NUMBER: ML040650220-Meeting Summary
ML040640564-Presentation Handouts
ML040640981-APEX Scaling Report
ML040650210-Pkg.

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JUhle
YHsii
LWard
NTrehan
LBrown
BHarvey
JWilson
TCheng

PUBLIC MEETING
REMAINING AP1000 OPEN ITEMS
ATTENDANCE LIST
DECEMBER 17, 2003

Name	Representing
J. Colaccino	U. S. Nuclear Regulatory Commission (NRC)
J. Starefos	NRC
J. Segala	NRC
B. Musico	NRC
N. Iqbal	NRC
P. Qualls	NRC
G. Cheruvenki	NRC
E. Sullivan	NRC
R. McIntyre	NRC
D. Thatcher	NRC
K. Coyne	NRC
E. Throm	NRC
L. Lois	NRC
W. Jensen	NRC
J. Uhle	NRC
Y. Hsii	NRC
L. Ward	NRC
N. Trehan	NRC
L. Brown	NRC
B. Harvey	NRC
J. Wilson	NRC
T. Cheng	NRC
T. Schulz	Westinghouse Electric Corporation (Westinghouse)
R. Vijuk	Westinghouse
M. Schoppman	Framatome ANP
M. King	Numark Associates
B. Smith	Energettics
L. Elwell	MPR
G. Quinn	Bechtel Power

Westinghouse Personnel Participating by Phone

T. Meneely	J. Grover
R. Orr	J. Winters
T. Andreychek	D. Wiseman
B. Brown	R. Kemper
A. Gagnon	R. Wright
C. Frepoli	K. Ihkawa
J. Scobel	R. Gold
R. Andersen	

Attachment 1

PUBLIC MEETING
REMAINING AP1000 OPEN ITEMS
DECEMBER 17, 2003

Draft Safety Evaluation Report (DSER) open items which remain open:

Reference: June 16, 2003, DSER Chapter 1 (ADAMS Accession Number ML031671486)

1.1-1	Unless otherwise noted, this report is based on DCD [design control document] Revision 3, dated February 6, 2003.
1.9-1	The staff has not yet identified all of the Tier 2* information pertaining to the AP1000 design. This effort will be completed to support the final safety evaluation report.
1.10-1	The staff has not yet completed the cross-reference of the COL [combined license] action items.
3.8.2.1-1	The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailed analyses.
3.8.4.5-2	During the review of the Wall 7.3 design calculation, the staff could not conclude that the corrected equation accurately calculates the necessary positive reinforcement.
4.5.1-1	The information on preservice examinations provided in Westinghouse's response to RAI [request for additional information] 252.001 should be addressed by a COL applicant, and should be reflected in DCD Tier 2 Section 5.3.6.
6.2.1.8.2-1	Westinghouse should justify the analysis used to determine the capability of the AP1000 in containment refueling water storage tank screens to accommodate anticipated debris loadings.
6.2.1.8.3-1	Westinghouse did not address the potential for other sources of debris, such as fibrous debris and floatble debris, to enter the reactor coolant system through a pipe break and block requisite core cooling flowpaths.
6.2.1.8.3-3	Westinghouse should justify the analysis associated with the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings.
6.4-1	Information is needed to complete the staff's review of the dose analysis for main control room personnel during design-basis accidents.

9.5.1-1	Westinghouse needs to demonstrate that the performance of the composite steel/concrete barrier provides an equivalent level of safety to that provided by using concrete and masonry.
13.6-1	The staff has not completed the review the applicant's change to the security plan.
14.2.10-1	The NRC staff has determined that the applicant should clarify whether this test should be performed for every AP1000 plant or justify that this test is a first plant-only test as described in DCD Tier 2 Section 14.2.5.
14.3.2-1	ITAAC [inspection, test, analysis, and acceptance criteria]: The staff finds that item No. 2 under the Design Description for the containment system states that the components identified in Table 2.2.1-1 and the piping identified in Table 2.2.1-2 are designed and constructed in accordance with ASME [American Society of Mechanical Engineers] Code Section III requirements. However, during the April 2-5, 2003, design audit, the staff found that the applicant did not complete the final analyses and design of the containment vessel, including attached components and piping systems.
14.3.2-8	ITAAC: The staff cannot complete its review of these ITAAC because the staff's review of the security program for AP1000 is not complete.
14.3.2-10	ITAAC: Please provide proposed ITAAC to verify that all Alloy 600/690 components and welds in the reactor coolant pressure boundary are identified and are readily accessible for bare metal visual inspection
14.3.2-12	ITAAC: Section 3.1, "Emergency Response Facilities," the staff finds this ITAAC unacceptable because it does not address the radiological habitability or the ventilation system for the technical support center; both of which should be the same as, or comparable to the main control room ITAAC.
14.3.4-1	ITAAC: Control room χ/Q values are not provided in Table 5.0-1, "Site Parameters." In addition, the staff has not completed its evaluation, but has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs.
15.2.7-1	The information provided regarding core void distribution was not sufficiently detailed to draw the conclusion that adiabatic heating would be avoided. The applicant should provide more detailed information of the axial void distribution during LTC [long term cooling] and show that the possibility of adiabatic fuel heating is excluded.
15.3-1	The staff has not completed its evaluation of the use of the AP600 aerosol removal coefficients. The staff is performing independent analyses to confirm information provided by the applicant.

15.3-2	The staff has not completed its evaluation of the hypothetical reference control room atmospheric dispersion factors, as discussed in Section 2.3.4 of this report. When this evaluation is completed, the staff will complete its review of the design basis accident control room habitability radiological consequences analyses.
15.3.6-1	The staff has not completed its evaluation of the LOCA [loss of coolant accident] radiological consequences analysis because of Open Items 15.3-1 and 15.3-2 discussed above. When these Open Items are resolved, the staff will complete its review of the design basis LOCA offsite and control room radiological consequences analyses.
16.2-2	Westinghouse should clarify the technical specification Bases for the passive core cooling system limiting condition for operation.
17.3.2-1	The NRC staff plans to perform a QA [quality assurance] test control implementation inspection to determine if additional testing activities performed at the test facility associated with the AP1000 design are accomplished, in accordance with the Westinghouse 10 CFR Part 50, Appendix B, QA program as described in Chapter 17 of the AP1000 [DCD].
17.3.2-2	The staff plans to conduct an inspection of the implementation of the project-specific quality plan to verify that design activities conducted for the AP1000 project complied with the Westinghouse QMS and the requirements of 10 CFR Part 50, Appendix B.
19.1.3.2-2	Westinghouse should include the failure mode of plugging of the drains near the floor of the annulus around the containment shell in the AP1000 PRA [probabilistic risk assessment].
19.1.10.1-2	The final list of “design certification requirements” related to the PRA should be in agreement with the resolution of the PRA open items identified in the AP1000 DSER.
19.1.10.1-3	Westinghouse should provide additional information related to the steps in the process for using PRA results to identify non-safety-related systems for regulatory oversight as well as the type and level of such oversight.
19.1.10.1-4	Westinghouse should address the impact of the uncertainties on the AP1000 PRA results and insights associated with (1) the assumed frequencies of large loss-of-coolant accidents and steam generator tube rupture accidents, and (2) the success criteria assumed for passive containment cooling by air flow.
19.1.10.2-6	Issues associated with the shutdown fire risk evaluation need to be resolved.
19.2.3.3-1	Westinghouse should address issues associated with the documentation of the ULPU Configuration V testing.

19.4-1 Issues associated with the severe accident mitigation design alternatives analysis need to be resolved.

19A.2-8 Westinghouse should address the necessity for a review of the core makeup tank HCLPF [high confidence low probability of failure] value if there is any increase in seismic response of the containment internal structure due to lift off of the internal structure or the nuclear island structure.

21.1-1 Chapter 21 currently contains references to NUREG-1512, which provides the basis for accepting the AP600 testing and computer codes. Prior to issuing the final safety evaluation report for the AP1000, the staff will remove these references and replace the references with the basis for its conclusion that the testing and computer codes are acceptable for the AP1000.

21.5-1 The applicant's submittals and responses to RAIs concerning hot leg phase separation were not sufficient to demonstrate that the codes used in AP1000 safety analysis model the hot leg phase separation process correctly. This issue is considered open until the applicant confirms the sensitivity studies performed by the staff using the code(s) the applicant intends to use to model SBLOCAs [small break LOCAs] in AP1000.

21.5-2 Given the lack of well scaled experimental data on upper plenum entrainment phenomena and the importance of predicting this process in an advanced plant SBLOCA transient, the staff has requested new experimental data to support the use of the upper plenum entrainment models in the AP1000.

21.5-3 Additional justification that the AP1000 core will remain covered as predicted by the codes should be provided since high void fractions were predicted.

<u>Post DSER open items which remain open</u>:

5.2.3-2 Alloy 52/152 materials are known to be difficult to weld. Address what examinations have been given to the adequacy of the quality assurance (QA) criteria for the Alloy 52/152 weldments that will be used to connect stainless steel piping to the ferritic pressure vessel? Address whether the QA criteria are commensurate with the risk associated with weldment failure?

Reference: NRC Letter dated September 3, 2003 (ADAMS Accession Number ML032330275)

5.2.3-3 The high-chromium nickel-base alloys (e.g. Alloys 690/52/152, as well as 82/182) may be susceptible to a significantly lowered fracture toughness if they have been exposed to high temperature hydrogenated water and then stressed at lower temperatures (e.g. <120C). This is a known phenomenon and may be of significance during a thermal shock event (i.e. during an accident scenario when there is ingress of large amounts of cold water into the primary system). Address whether this phenomenon could result in the failure of the nozzles between the pressure vessel and main recirculation or direct vessel injection (DVI) piping. If such a failure occurred, what are the consequences?

Reference: NRC Letter dated September 3, 2003 (ADAMS Accession Number ML032330275)

6.2.1.8.3-4 The NRC staff has been developing Revision 3 to Regulatory Guide 1.82, “Water Sources For Long-term Recirculation Cooling Following a Loss-of-coolant Accident.” During that review, the staff has identified concerns related to additional debris that can be caused by chemical reactions in the containment. The staff is requesting that the applicant address the following chemical effects as they relate to head loss calculations provided in the responses to Open Items 6.2.1.8.2-1, 6.2.1.8.3-1, and 6.2.1.8.3-3.

a. To minimize potential debris caused by chemical reaction of the pool water with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to containment cooling water through spray impingement or immersion should be minimized either by removal or by chemical-resistant protection (e.g., coatings or jackets).

b. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) should be considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

Reference: NRC Letter dated December 3, 2003 (ADAMS Accession Number ML033290096)

PUBLIC MEETING
COMMENTS ON CERTAIN AP1000 OPEN ITEMS IN DSER CHAPTER 19
DECEMBER 17, 2003

Note: These comments were transmitted from NRC staff to Westinghouse in an E-mail dated December 9, 2003.

19.1.10.3-1: The revised OI response (Rev 2) dated 9/18/2003 provided a markup of Chapter 45. One correction was missed. Specifically, in Section 45.2.4, the next to the last sentence, the words "hydrogen detonation" should be changed to "diffusion flames".

19.1.10.3-2: In the response to this item (dated 7/1/2003) Westinghouse identified the major causes of reactor cavity flooding failure and hydrogen igniter failure. During our phone call in September we indicated that the information provided was responsive to the request, but noted that the dominant contributors to reactor cavity flooding failure in AP600 (common cause failure of strainers and common cause failure of actuation software) contribute substantially less in AP1000. Westinghouse agreed to look into the reasons for this change. We haven't received this information yet.

PUBLIC MEETING
COMMENTS ON AP1000 DSER OPEN ITEM 4.5.1-1
DECEMBER 17, 2003

Note: This issue has been discussed previously with Westinghouse.

DCD Tier 2, Section 4.5.1, states, in part, the following:

> The recent experience with VHP nozzle cracking has identified the need for baseline inspection data to determine if an indication is service-induced cracking, or an artifact from fabrication. The staff requested information on what preservice examinations will be performed on the VHP nozzles. In a letter dated April 7, 2003, the applicant responded that preservice examinations for the closure head will include a baseline top-of-the head visual examination, ultrasonic examinations of the inside diameter surface of each vessel head penetration, eddy current examination of the surface of the head penetration welds and the inside diameter surface of the penetrations, and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations. Any indications exceeding the ASME Code Section III requirements would be removed. The information in the RAI response has been provided in DCD Tier 2 Section 5.3.4.7. The information on preservice examinations also needs to be addressed by a COL applicant, and should be reflected in DCD Tier 2 Section 5.3.6, "Combined License Information." This is identified as Open Item 4.5.1-1 and COL Action Item 4.5.1-1.

<u>Issue</u>: It is not clear if the entire volume of the nozzle is subject to preservice examination. Likewise, it is not clear how much of the outside and inside diameter surfaces will be subject to eddy current examination. Please clarify.

Reference: Westinghouse Revision 1 response to RAI 252.001 dated April 7, 2003

DCD Tier 2, Section 5.3.4.7

PUBLIC MEETING
COMMENTS ON DSER AP1000 OPEN ITEMS 9.5.1-1
DECEMBER 17, 2003

In the October 21, 2003, Revision 1 letter, (response to the Open Item 9.5.1-1 Part c) the applicant stated that,

> "AP 1000 Design Control Document, Appendix 9A, Table 9A-2 (Sheet 7 of 14) identifies the safe shutdown components in the PCS valve room (Room 12701, Fire Area 1000 AF 01, Fire Zone 1270 AF 12701). The components in this room consist of six (2 air operated and 4 motor operated valves) Passive Containment Cooling Water Storage Tank (PCCWST) isolation valves and five PCCWST flow/level instruments. In the unlikely event that a fire occurred in Room 12701 to the extent that these components were rendered inoperable, the ability to achieve safe shutdown would not be compromised.
>
> Normal shutdown operations may be required in the event that a fire were to significantly damage the safe shutdown components located in Room 12701. Normal shutdown operations do not require the actuation of the PCS valves located in this room.
>
> In the event of a design basis accident the three, normally closed, PCCWST isolation valves (two air operated valves and one motor operated valve) located in Room 12701 could receive an automatic signal to open. Access to the PCS valve room is not required for their operation.
>
> There is very little combustible material in the PCS valve room fire area. In the unlikely event of a fire, the fire brigade would approach the PCS valve room (Room 12701, Fire Zone 1270 AF, Fire Area 1000 AF 01) by using the Stairwell S03 of the adjacent elevator that is attached to the outside wall on the shield building. The fire bridge would egress from Stairwell S03 (Fire Area 120A AF 02) at the El. 264'-6" platform (Fire Area 1270 AF, Fire Area 1000 AF 01). They would proceed up the included stair S06 to the PCS valve room (Room 12701) located on El. 286'-6". From this position they could utilize fire extinguishers for manual fire fighting."

The staff review the applicant's response to Open Item 9.5.1-1 Part c and found it unacceptable. The staff believes that access is required to PCS valve room (Room 12701) since it contains safety-related equipment for safe shutdown. BTP CMEB 9.5.1, Regulatory Position C.6.c.4 require, "Interior manual hose should be able to reach any location that contains, or could present a fire exposure hazard to, safety-related equipment with at least one hose stream....." The staff believes that for manual fire fighting, fire extinguishers are not always effective and often water hose streams is required for effective fire fighting. Therefore, DSER Open Item 9.5.1-1 (Item c) is pending for resolution.

To resolve the DSER Open Item 9.5.1-1 (Item c) the staff requires additional information to evaluate fire extinguishers' effectiveness for suppressing cable and oil fire in the PCS valve room.

1. Provide quantities of in-situ combustibles materials, i.e., lubricants and cable insulation in the PCS valve room.

2. Provide the types of transient combustible materials introduced during maintenance work in the PCS valve room.

3. What types of fire extinguishers are provided and where are they located in relation to the PCS valve room? Describe the access path.

4. Provide type and quantity of additional fire extinguishers to be provided during the maintenance work in the PCS room.

5. What if the fire extinguishers in the PCS valve room cannot extinguish a cable or oil fire. Electrical cable insulation may develop into a deep-seated fire. What additional fire suppression capability is available?

6. Due to the access delays in fire fighting in the PCS valve room could potentially compromise the second element of the defense-in-depth (DID) in this safety-related fire area result in reduction of the DID required by the regulation. Section II of Appendix R to 10 CFR Part 50, "General Requirements," states that the fire protection program shall extend the concept of defense-in-depth to fire protection in fires areas that are important to safety, with the following objectives:

   - Prevent fires from starting.

   - Rapidly detect, control, and extinguish those fires that do occur.

   - Protect structures, systems, and components that are important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

   Provide a valid technical justification that how manual fire fighting with fire extinguishers in the PCS valve room will rapidly control and extinguished a flammable liquid and cable fire.

   - Are fire water drains installed in the PCS valve room?

PUBLIC MEETING ISSUES RELATED TO COL ACTION ITEMS DECEMBER 17, 2003					
Item #	**Comment Description**	**Open**	**Confirm**	**Resolved**	**Status Details**
7.2.3-2	DSER addresses two issues. (1) Response time testing, and (2) Setpoint methodology. The Westinghouse wording only mention "Setpoint methodology". The DCD should be modified to include "Response time testing". Then our FSER can use their wording.	1			
7.2.6-1	DSER addresses two issues. (1) Completion Time, and (2) FMEA. The Westinghouse wording only mention FMEA. The DCD should be modified to include "Completion Time". Then our FSER can use their wording.	1			
14.4-4	DSER wording is significantly different from the writeup in the DCD, Revision 7. The FSER should be revised to state: "The COL applicant ~~and~~or holder is responsible for review and evaluation of individual test results. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retest, as required, will be performed. ~~In as much as test results will not be available until a facility is built, the NRC staff determined that it is appropriate and acceptable to defer the review and evaluation of individual test results to the COL applicant or COL holder, as appropriate.~~ This is a COL Action Item 14.4-4"	1			DCD Rev. 7, Section 14.4.4, "Review and Evaluation of Test Results," states: "The Combined License applicant and holder is responsible for review and evaluation of individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed."

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	The DCD should be revised to state: "The COL applicant or holder is responsible for review and evaluation of individual test results. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retest, as required, will be performed."				
14.4-6	The FSER should be revised to state: "The COL applicant or ~~licensee~~holder for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in subsection 14.2.5, or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. This is COL Action Item14.4-6." The DCD should be revised to state: "The COL applicant or holder for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant will either perform the tests listed in subsection 14.2.5, or will provide a justification that the results of the first-plant-only tests or the first-three-plant tests are applicable to the subsequent plant."	1			This also a Tier 2* item? DCD Rev 7 Section 14.4.6 states:: [The COL applicant or licensee for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in subsection 14.2.5, or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant.]*

Item #	Comment Description	Open	Confirm	Resolved	Status Details
17.1-1	DSER wording is significantly different from the writeup in the DCD, Revision 7 The FSER should be revised to state: The ~~Combined License (~~COL~~)~~ applicant or holder will address its quality assurance (QA) program for the design phase, as well as its QA program for procurement, fabrication, installation, construction and testing of structures, systems, and components (SSCs) in the facility. The quality assurance program will include provisions for seismic Category II structures, systems and components. ~~When completing the detailed design during the COL design phase, the COL applicant is required to submit its design phase QA program for staff review. This will be in addition to the staff review of the COL applicant 's QA program for construction of the facility. This is described as a COL information item in DCD Tier 2 Section 17.5. The NRC staff agrees that this part of the QA program can be the COL applicant's responsibility and that making this a COL item in DCD Tier 2 Section 17.5 is acceptable.~~ This is COL Action Item 17.1-1. The DCD should be revised to state: The COL applicant or holder will address its design phase Quality Assurance program, as well as its Quality Assurance Program for procurement, fabrication, installation, construction, and testing of	1			DCD Rev. 7 Section 17.5 states: "The Combined License applicant will address its design phase Quality Assurance program, as well as its Quality Assurance program for procurement, fabrication, installation, construction and testing of structures, systems and components in the facility. The quality assurance program will include provisions for seismic Category II structures, systems, and components."

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	structures, systems and components in the facility. The quality assurance program will include provisions for seismic Category II structures, systems and components.				
17.2-1	The FSER should be revised to state: The COL applicant or holder will address its QA program for operations. This is described as a COL information item in DCD Tier 2 Section 17.5. ~~The NRC staff agrees that this part of the QA program can be the COL applicant's responsibility and that making this a COL item in DCD Tier 2 Section 17.5 is acceptable.~~ This is COL Action Item17.2-1. The COL applicant or holder will address its QA program for operations.	1			
20.3-1	DCD 1.9.4.2.2 Issue 142 AP1000 Response (Page 1.9-65) has error in their statement. This statement is same as AP600 response to Issue 142. However, the AP1000 DCD Chapter 7 has been revised such that there is no Section 7.1.4.2.7, "Conformance to the Requirements Concerning Control and Protection System Interaction," and "Isolation Devices" is under Section 7.1.2.10, not under "7.1.2.11" as stated in AP1000 response to Issue 142. Westinghouse should revise DCD 1.9.4.2.2 Issue 142 AP1000 Response to correct all the discrepancies between chapter 7 and chapter 1.9 statements. and provide proper justification why "COL Action" is not required. Also, AP1000 should	1			

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	address compliance with IEEE-603 standard, not IEEE-279.				
20.7-1	This item is included in WCAP-15800, "Operational Assessment for AP1000." WCAP-15800 identifies that it is part of COL Verification/Procedural issue DCD Section 13.5. However, DCD Section 13.5 does not specifically address this issue.	1			BL 80-06, "Engineered Safety Feature Reset Controls"
20.7-8	GL-88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary in PWR Plant Components," requested assurance that licensees had implemented a program to ensure that boric acid corrosion does not degrade the RCPB. In WCAP-15800, Revision 1, “Operational Assessment for AP1000,” the applicant indicated that this GL is not applicable to the AP1000 design and because it is the responsibility of the COL applicant. The staff agrees that this is an inspection issue and within the scope of the COL applicant. However, the DCD Tier 2 material does not provide a COL commitment that the COL applicant will be developing a boric acid corrosion program to provide reasonable assurance of compliance with the applicable regulatory requirements. The DCD needs to be revised, for example in Section 5.2, to indicate this COL commitment. Westinghouse is requested to address this issue.	1			
20.7-13	COL Action Item 20.7-13 is related to Generic letter 93-04, Rod Control System Failure and Withdrawal of Rod Cluster Assemblies. GL 93-04 was closed	1			

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	based on NRC approval of WCAP-13864, Rev. 1, which requires licensees to perform additional rod control system surveillance tests at the beginning of each cycle. Westinghouse indicated that this is COL's responsibility. To respond to COL Action Item 20.7-13, AP1000 DCD should include a statement that COL applicants referencing AP1000 certified design will establish procedures to perform rod control system surveillance tests specified in WCAP-13864, Revision 1, at the beginning of each fuel cycle.				
20.7-14	This item is included in WCAP-15800, "Operational Assessment for AP1000." WCAP-15800 identifies that it is part of DCD Section 7.1.2. However, DCD Section 7.1.2 does not specifically address this issue. Westinghouse's August 21,2003 letter identifies that this issue is under DCD Section 13.5. However, DCD Section 13.5 does not specifically address this issue.	1			GL 96-01, "Testing of Safety-Related Logic Circuits"
20.7-15	DCD Section 9.1.6 should be revised to state the following: "The Combined License applicant is responsible for a confirmatory criticality analysis for the new fuel rack, as described in subsection 9.1.1.3. **This analysis should address the degradation of Boraflex in the spent fuel pool storage racks as identified in GL-96-04, and assess the Boraflex capability to maintain a 5% subcriticality margin.**	1			

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	The Combined License applicant is responsible for a confirmatory criticality analysis for the spent fuel racks, as described in subsection 9.1.2.3. **This analysis should address the degradation of Boraflex in the spent fuel pool storage racks as identified in GL-96-04, and assess the Boraflex capability to maintain a 5% subcriticality margin.**"				
20.7-16	GL-97-06, "Degradation of Steam Generator Internals," requested, in part, that licensees discuss any programs in place to detect degradation of steam generator internals including a description of the plans, scope, frequency, methods, and equipment used. In WCAP-15800, Revision 1, "Operational Assessment for AP1000," the applicant indicated that this GL is not applicable to the AP1000 design since it is a procedural issue and the tube supports are fabricated from stainless steel. The staff agrees that this is a procedural issue that will have to be addressed by the COL applicant and that the likelihood of degradation of the SG internals will be less given the AP1000 SG design; however, the design does not eliminate the potential for degradation of the steam generator internals to occur. As a result, the staff concludes that the COL applicant will need to develop a program for periodic monitoring for potential degradation of steam generator internals and that the DCD needs to be revised, for example in Section 5.4, to indicate this	1			

Item #	Comment Description	Open	Confirm	Resolved	Status Details
	COL commitment. Westinghouse is requested to address this issue.				
Total Comments:		**13**	**0**	**0**	

OI No.	Open	Confirm	Resolve
1.1-1	1		
1.9-1	1		
1.10-1	1		
2.3.4-1		1	
2.5.1-1		1	
2.5.2-1			1
2.5.4-1			1
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2.5.4-3			1
3.3.1-1		1	
3.3.1-2		1	
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3.3.2-2		1	
3.3.2-3			1
3.5.1.3-1			1
3.5.1.3-2			1
3.6.3.4-1			1
3.6.3.4-2			1
3.7.1.5-1		1	
3.7.2.1-1			1
3.7.2.3-1		1	
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3.7.2.3-3		1	
3.7.2.9-1		1	
3.7.2.16-1		1	
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3.8.4.5-1		1	
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3.8.5.1-1		1	
3.8.5.4-1			1
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4.4-1			1
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4.5.2-1			x
5.2.3-1			1
5.2.3-2	x		
5.2.3-3	x		
5.3.3-1			1
5.4.2-1			1
6.1-1			x
6.1.1-1			1
6.2.1.8.1-1			1
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6.2.1.8.3-1	1		
6.2.1.8.3-2			1
6.2.1.8.3-3	1		
6.2.1.8.3-4	x		
6.2.5-1			1
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6.4-1	1		
8.2.3.1-1			1
9.4-1			1
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9.5.2-4			1
10.2.8-1			1
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13.6-1	1		
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14.2-1.c			x
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14.2-1.e			x
14.2-1.f		x	
14.2-1.g			x
14.2-1.h			x
14.2-1.i			x
14.2-1.j			x
14.2-1.k			x
14.2-1.l			x
14.2-1.m			x
14.2-1.n			x
14.2-1.o			x
14.2-1.p			x
14.2-1.q			x
14.2-1.r			x
14.2-1.s			x
14.2-1.t		x	
14.2-1.u		x	
14.2-1.v		x	
14.2-1.w			x
14.2-1.x			x
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14.3.3-9		1	
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14.3.3-11			1
14.3.3-12			1
14.3.3-13			1

OI No.	Open	Confirm	Resolve
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14.3.3-17			1
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18.11.3.4-1			1
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20.7-2			1
21.1-1	1		
21.5-1	1		
21.5-2	1		
21.5-3	1		
	34	**47**	**93**

AP 1000

cc:

Mr. W. Edward Cummins
AP600 and AP1000 Projects
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. H. A. Sepp
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230

Lynn Connor
Doc-Search Associates
2211 SW 1ST Ave - #1502
Portland, OR 97201

Barton Z. Cowan, Esq.
Eckert Seamans Cherin & Mellott, LLC
600 Grant Street 44th Floor
Pittsburgh, PA 15219

Charles Brinkman, Director
Washington Operations
Westinghouse Electric Company
12300 Twinbrook Parkway, Suite 330
Rockville, MD 20852

Mr. R. Simard
Nuclear Energy Institute
1776 I Street NW
Suite 400
Washington, DC 20006

Mr. Thomas P. Miller
U.S. Department of Energy
Headquarters - Germantown
19901 Germantown Road
Germantown, MD 20874-1290

Mr. David Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1707 H Street NW, Suite 600
Washington, DC 20006-3919

Mr. Paul Gunter
Nuclear Information & Resource Service
1424 16th Street, NW., Suite 404
Washington, DC 20036

Mr. Tom Clements
6703 Guide Avenue
Takoma Park, MD 20912

Mr. James Riccio
Greenpeace
702 H Street, NW, Suite 300
Washington, DC 20001

Mr. James F. Mallay, Director
Regulatory Affairs
FRAMATOME, ANP
3315 Old Forest Road
Lynchburg, VA 24501

Mr. Ed Wallace, General Manager
Projects
PBMR Pty LTD
PO Box 9396
Centurion 0046
Republic of South Africa

Mr. Vince Langman
Licensing Manager
Atomic Energy of Canada Limited
2251 Speakman Drive
Mississauga, Ontario
Canada L5K 1B2

Mr. Gary Wright, Manager
Office of Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive
Springfield, IL 62704

Dr. Gail H. Marcus
U.S. Department of Energy
Room 5A-143
1000 Independence Ave., SW
Washington, DC 20585

Mr. Paul Leventhal
Nuclear Control Institute
1000 Connecticut Avenue, NW
Suite 410
Washington, DC 20036

Mr. Jack W. Roe
SCIENTECH, INC.
910 Clopper Road
Gaithersburg, MD 20878

Patricia Campbell
Winston & Strawn
1400 L Street, NW
Washington, DC 20005

Mr. David Ritter
Research Associate on Nuclear Energy
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. Ronald P. Vijuk
Manager of Passive Plant Engineering
AP1000 Project
Westinghouse Electric Company
P. O. Box 355
Pittsburgh, PA 15230-0355

Mr. Russell Bell
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708