June 9, 2003

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

Dear Mr. Cummins:

As you are aware, the U.S. Nuclear Regulatory Commission (NRC) staff is preparing the draft safety evaluation report (DSER) for the AP1000 design certification application submitted by Westinghouse Electric Company (Westinghouse) on March 28, 2002. The staff expects to issue the DSER in June 2003. As of this date, the staff has identified 43 potential open items for DSER Chapter 14, "Verification Systems," which are enclosed for your information. Please note that the staff's review of the application will continue during preparation of the DSER, which may result in changes to the potential open items identified in the enclosure, or the addition of other open items.

The potential open items in the enclosure have the original request for additional information (RAI) number included for reference, if applicable. If the staff cannot resolve the potential open items before the issuance of the DSER, these items will be issued as DSER open items and be tracked with a corresponding open item number.

Previously, Westinghouse committed to provide responses to all identified open items within 9 weeks after the issuance of the DSER. The staff will be prepared to review your responses to the open items and have conference calls and meetings with your staff, as appropriate, after the DSER is issued. If Westinghouse chooses to address some or all of these open items before the issuance of the DSER, the staff may not have sufficient time to evaluate every response to the potential open items that Westinghouse submits to the NRC and make changes to the DSER before the scheduled DSER issuance in June 2003.

Please contact one of the following members of the AP1000 project management team if you have any questions or comments concerning this matter: Mr. John Segala (Lead Project Manager) at (301) 415-1858 or jps1@nrc.gov, Mr. Joseph Colaccino at (301) 415-2752 or jxc1@nrc.gov, or Ms. Joelle Starefos at (301) 415-8488 or jls1@nrc.gov.

Sincerely,

/RA/

James E. Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

Docket No. 52-006

Enclosure: As stated

cc: See next page

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Sincerely,

/RA M. Gamberoni for:/

James E. Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

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Enclosure: As stated
cc: See next page

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DATE	6/9/2003	6/9/2003	6/9/2003

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Westinghouse AP1000
Draft Safety Evaluation Report
Potential Open Items
Chapter 14
Verification Programs

Open Item Number: 14.2-1

Original RAI(s): N/A

Summary of Issue: The review documented in Section 14.2 of this chapter reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria. Pending completion of the review, this is Open Item 14.2-1.

Open Item Number: 14.2.7-1

Original RAI(s): 261.014

Summary of Issue: RG 1.41, “Preoperational Testing of Redundant On-Site Electric Power Systems to verify Proper Load Group Assignments.” Revision 0. In DCD Appendix 1A, the applicant provided the following information related to RG 1.41:

> The guidelines are followed for Class 1E dc power systems during the preoperational testing of the AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional testing, an evaluation is used to supplement the testing.

The NRC staff lacked sufficient information to determine if this exception to RG 1.41 was acceptable. Specifically, the staff was unable to identify which regulatory position in RG 1.41 the exception applied to, if the exception applied to both alternating current and direct current systems, and in what cases was it not practical to perform functional testing. Therefore, in Request for Additional Information (RAI) 261.014, the NRC staff requested the applicant to provide additional specific information regarding this exception. This is Open Item 14.2.7-1.

Open Item Number: 14.2.7-2

Original RAI(s): 261.015

Summary of Issue: RG 1.68, “Initial Test Program for Water-Cooled Nuclear Power Plants,” Revision 2, Appendix A, Item 4.t. In DCD Appendix 1A, the applicant provided the following information related to RG 1.68, Appendix A, test 4.t:

> For the AP1000, natural circulation heat removal to cold conditions using the steam generators is not safety-related, as in current plants. This safety function is performed by the PRHR [Passive Residual Heat Removal]. Natural circulation heat removal via the PRHR is tested for every plant during hot functional testing.

Because the PRHR heat exchanger is the safety related heat sink for the AP1000 design, the NRC staff determined that natural circulation testing of the PRHR system, rather than the steam generators, met the intent of RG 1.68 and was therefore acceptable. However, the NRC found that the exception to RG 1.68, Appendix A, Item 4.t, contradicts with the low power test abstracts in DCD Tier 2 Section 14.2.10.3.6, “Natural Circulation (First Plant Only)” and DCD Tier 2 Section 14.2.10.3.7, “Passive Residual Heat Removal Heat Exchanger (First Plant Only).” Specifically, the exception to RG 1.68 states, in part, that “the PRHR is tested for every plant during hot functional testing.” However, the lower power natural circulation test abstracts state that this test is a “first-plant-only” test. Additionally DCD Tier 2 Section 14.2.10.3.7, states, in part, PRHR natural circulation testing is not required to be performed if a large scale test of the AP600 or AP1000 type passive residual heat removal heat exchanger has been conducted, and has provided data confirming adequate heat removal capability. Because of the conflicting information contained in the DCD, the staff was unable to complete the review of this regulatory position exception. Therefore, in RAI 261.015, the NRC staff requested the applicant to clarify and justify the inconsistent natural circulation testing requirements in the exception to RG 1.68 and in Test Abstracts 14.2.10.3.6 and 14.2.10.3.7 and clarify under what circumstances natural circulating testing would be performed. This is Open Item 14.2.7-2.

Open Item Number: 14.2.7-3

Original RAI(s): N/A

Summary of Issue: The NRC staff determined that the applicant should delete this exception to RG 1.68 in WCAP-15799 and state that the test abstract in DCD Tier 2 Section 14.2.10.4.28 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, test d.d, or provide additional information to clarify this exception. This is DSER open item 14.7.2-3.

Open Item Number: 14.2.10-1

Original RAI(s): 261.009, 261.016

Summary of Issue: RG 1.68, Appendix A, Item 4.c recommends performance of pseudo-rod ejection testing to verify calculation models and accident analysis assumptions during low power testing. The NRC staff could not locate an AP1000 low power test abstract that performed this testing. In RAI 261.009, the NRC staff requested that the applicant provide additional information regarding the performance of pseudo-rod ejection testing for the AP1000 design. In their November 13, 2002, RAI response, the applicant stated that sufficient test data has been obtained from previous plant startups, which allows new plants to require only confirmation of the calculational models. The applicant also provided several licensing precedents associated with this position.

The NRC staff lacked sufficient information to conclude that this testing was only required to confirm calculational models. As described in the staff evaluation of RAI 261.007b, Item 2, below, the NRC staff requested that the applicant provide additional information relating to the conduct of pseudo rod ejection testing. This request for additional information is identified as RAI 261.016. Pending resolution of RAI 261.016 and RAI 261.009, this is Open Item 14.2.10-1.

Open Item Number: 14.2.10-2

Original RAI(s): 261.007b

Summary of Issue: RG 1.68, Appendix A, Item 5.e. recommends performance of pseudo-rod ejection testing during the power ascension test phase to validate the rod ejection accident analysis. RG 1.68 further states that this test need not be repeated for facilities using calculation models and design identical to prototype facilities. The NRC staff could not locate a power ascension test abstract that addressed this testing. In RAI 261.007b, the NRC staff requested that the applicant provide additional information regarding the performance of this testing. In their November 13, 2002, RAI response, the applicant stated, in part, that this test was part of the rod cluster control assembly out of bank measurements in DCD Tier 2 Section 14.2.10.4.6, “Rod Cluster Control Assembly Test.” The applicant notes that this test is only performed on the first plant to validate the analysis.

The NRC staff determined that the pseudo rod or Rod Cluster Control Assembly (RCCA) ejection test is performed in Test Abstract 14.2.10.4.6; therefore, RAI 261.007b, Item 2 is partially resolved. However, the applicant states that this test is performed on the first plant only. The NRC staff 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. The NRC staff also notes that DCD Tier 2 Section 14.4.6 requires the COL applicant or licensee to either perform the tests listed in DCD Tier 2 Subsection 14.2.5 or provide justification that the results of the first plant only tests are applicable to subsequent plants. This is Open Item 14.2.10-2.

Open Item Number: 14.2.10-3

Original RAI(s): 261.007b

Summary of Issue: RG 1.68, Appendix A, Section 5.i recommends that the capability and/or sensitivity, as appropriate for the facility design of incore and excore neutron flux instrumentation, to detect a control rod misalignment equal to or less than the technical specification limits be demonstrated during power ascension testing at 50 percent and 100 percent of rated reactor power. However, as described in DCD Tier 2 Section 14.2.10.4.6, "Rod Cluster Control Assembly Out of Band Measurements," this test is performed at 30 percent and 50 percent of rated thermal power. In RAI 261.007b, the NRC staff requested that the applicant provide additional information justifying not performing this testing at 100 percent of rated thermal power. In the applicant's response dated November 13, 2002, the applicant stated that the rod cluster control assembly out of bank measurements test is not performed at full power as it would cause the plant to exceed peak power limits. The NRC staff agrees that this test should not be performed at a power level that could cause the plant to exceed thermal limits. However, the applicant should either perform the test at a higher power level consistent with RG 1.68 or provide additional information to justify performing this test at a maximum of 50 percent power. This is Open Item 14.2.10-3.

Open Item Number: 14.2.10-4

Original RAI(s): 261.018

Summary of Issue: RG 1.68, Appendix A, Section 5.m.m recommends that the power ascension test program include demonstrations that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves (MSIVs). In reviewing the power ascension test program test abstracts, the NRC staff noted that no MSIV closure testing is performed during power ascension testing. In RAI 261.007b, the staff requested that the applicant provide additional information regarding performance of MSIV closure testing. In their November 13, 2002, RAI response, the applicant stated that the dynamic response of the plant to closure of all MSIVs is bounded by a plant trip from 100 percent power, which is performed in Test Abstract 14.2.10.4.24.

The NRC staff lacks sufficient information to conclude that the plant trip from 100 percent bounds the MSIV closure transient. In RAI 261.018, the NRC staff requested the applicant to provide additional information regarding the basis for the statement that the MSIV closure transient is bounded by a plant trip from 100 percent power. This is Open Item 14.2.10-4.

Open Item Number: 14.3.2-1

Original RAI(s): N/A

Summary of Issue: Section 2.2.1, “Containment System.” The staff found 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 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 (Section 3.8.2.1 of this report). The issue related to the containment design is designated as Open Item 14.3.2-1.

Open Item Number: 14.3.2-2

Original RAI(s): N/A

Summary of Issue: Section 2.2.1. The phrase “structural integrity and” should be added in two places: (1) Design Description No. 5 for the containment system, and (2) Subitem No. 5.ii under the Acceptance Criteria of ITAAC Table 2.2.1-3. The sentence should read “. . . the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of <u>structural integrity and</u> safety function.” This is Open Item 14.3.2-2.

Open Item Number: 14.3.2-3

Original RAI(s): N/A

Summary of Issue: Section 2.2.1. The thickness of the steel containment vessel should be designated as Tier 1 information and specified in Section 2.2.1 or listed in Table 3.3-1. This is Open Item 14.3.2-3.

Open Item Number: 14.3.2-4

Original RAI(s): N/A

Summary of Issue: Section 2.3.2, “Chemical and Volume Control System.” The staff found that incomplete design commitments related to controls and displays exist in the current system-based ITAAC. For example, one current description states that, “Controls exist in the MCR [main control room] to cause the pumps identified in Table 2.3.2-3, to perform the listed function.” The staff recommends revising this design commitment to indicate that, not only should the controls exist in the MCR and perform their intended functions but, the controls should be designed so that they are usable by operators. A suggested revision to accommodate this change is, "Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3, to perform the listed function and are designed in accordance with state-of-the-art human factors principles as required by 10 CFR 50.34(f)(2)(iii)." The same concern applies to the current design commitment statements related to displays. As an example, the current design commitment of, "Safety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR," should be changed to, "Safety-related displays identified in Table 2.3.2-1 can be retrieved from the MCR, perform their intended function, and are designed in accordance with state-of-the-art human factors principles as required by 10CFR50.34(f)(2)(iii)." These recommended changes to the above-cited examples apply to other current design commitments for system-based ITAAC. This is Open Item 14.3.2-4.

Open Item Number: 14.3.2-5

Original RAI(s): N/A

Summary of Issue: Section 2.3.5, "Mechanical Handling System," the design description (items 3.b and 3.c) for the equipment hatch hoist and the maintenance hatch hoist are not identified as single failure proof as they are in Tier 2. In addition to not being identified as single failure proof, Table 2.3.5.2 does not require a test, inspection, or analysis to demonstrate whether these items of equipment will meet their design criteria. As such, the design description in Tier 2 is inconsistent with that of the ITAAC. This is Open Item 14.3.2-5.

Open Item Number: 14.3.2-6

Original RAI(s): N/A

Summary of Issue: Section 2.3.9, "Containment Hydrogen Control System," must remain open because hydrogen control is an open item in this report (See DSER Section 6.2.5 and Open Item 6.1.1-1 of this report for details). Briefly, this is because the AP1000 Tier 2 information is written in anticipation of a rule change to 10 CFR 50.44 that will relax requirements, but has not been finalized. This is Open Item 14.3.2-6.

Open Item Number: 14.3.2-7

Original RAI(s): N/A

Summary of Issue: Section 2.3.19, "Communication Systems." ITAAC's have not been identified for the Communication system (EFS) as discussed in Tier 2 Section 9.5.2 beyond those given in Tables 2.3.19-2, and 3.1-1 (Emergency Response Facilities). There is no assurance that the appropriate tests and confirmatory criteria will be accomplished to meet requirements of the CFR's especially 10 CFR 73.55(e)(f)(g) and noise level considerations for worse case postulated noise levels. Provide appropriate ITAAC for all the communication systems. This is Open Item 14.3.2-7.

Open Item Number: 14.3.2-8

Original RAI(s): N/A

Summary of Issue: Sections 2.6.9 and 2.6.10. The staff cannot complete its review of these ITAAC because the staff's review of the security program for AP1000 is not complete (see Section 13.6 of this report). This is Open Item 14.3.2-8.

Open Item Number: 14.3.2-9

Original RAI(s): N/A

Summary of Issue: New ITAAC. In RAI 252.001, the staff requested information related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience, NRC Bulletins 2001-01, 2002-01 and 2002-02. Since the RAI was issued, the staff has issued Orders, EA-03-009, to operating license holders related to inspection for cracks in these penetrations and attachment welds. The staff subsequently issued followup questions to the applicant related to changes in design and fabrication to reduce residual stresses, ability to visually inspect 360 degrees around each nozzle, preservice volumetric inspection, and determination of the operating head temperature. The applicant responded to the followup questions in a letter dated April 7, 2003. Please provide proposed ITAAC related to the issues noted above which were discussed in your RAI responses. This is Open Item 14.3.2-9.

Open Item Number: 14.3.2-10

Original RAI(s): N/A

Summary of Issue: New ITAAC. Operating experience continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack, based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with reactor coolant is a substantial improvement over the use of materials currently in wide use in the industry. However, data is not presently available to demonstrate that cracking in these welds and components will not occur over the projected 60-year design lifetime of an AP1000 plant. The staff also believes that bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs. Please provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions. 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. This is Open Item 14.3.2-10.

Open Item Number: 14.3.2-11

Original RAI(s): N/A

Summary of Issue: New ITAAC. The staff reviewed Tier 2 Section 5.3.4 as it applies to pressurized thermal shock in accordance with SRP 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.” Section 50.61 of 10 CFR Part 50, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” defines the fracture toughness requirements for protection against pressurized thermal shock (PTS) events. The requirements in 10 CFR 50.61 establish the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (RT_{PTS}). These criteria are 148.0°C (300°F) for circumferential welds and 132.2°C (270°F) for plates, forgings, and axial welds. To verify that the design will be in accordance with the regulatory requirements associated with PTS, the applicant needs to provide an appropriate ITAAC. The following is a suggested design commitment for this ITAAC: The amount of copper and nickel in the reactor vessel materials and the projected neutron fluences for the 40 year period of the COL will result in RT_{PTS} values lower than the screening criteria contained in 10 CFR 50.61. This is Open Item 14.3.2-11.

Open Item Number: 14.3.2-12

Original RAI(s): N/A

Summary of Issue: 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 with, the main control room ITAAC. This is Open Item 14.3.2-12.

Open Item Number: 14.3.2-13

Original RAI(s): N/A

Summary of Issue: Section 3.3, “Buildings.” Item 2.a of the Design Description and Table 3.3-6 states that the nuclear island structures, including the critical section listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design-basis loads (including seismic loads), as specified in the design description, without loss of structural integrity and the safety related functions. However, as identified in Open Items 3.7.2.3-1, 3.7.2.3-3 and 3.8.5.4-1, the applicant did not demonstrate that the foundation mat will not lift up, and/or the shear walls will not crack during a postulated seismic event. The phenomenon of the foundation mat uplifting and shear wall cracking will directly affect the design adequacy of the nuclear island structures, systems and components, including the thickness of structural elements listed in Table 3.3-1 and safety-related piping systems. Consequently, this is open item 14.3.2-13.

Open Item Number: 14.3.2-14

Original RAI(s): N/A

Summary of Issue: Section 3.3, ITAAC Table 3.3-6, Acceptance Criteria 2.g states that the tolerance on the height of the containment vessel is +12", -6" and the tolerance on the inside diameter is also +12", -6". The information included in Tier 2 related to the containment design does not address the +12" tolerance on the inside diameter. All of the applicant's analyses, calculations, and responses to the RAIs related to the containment vessel are based on the nominal inside diameter of 130 feet. From its review, it is the staff's understanding that the vessel wall thickness, currently specified for 130'(-0"), marginally meets ASME Code allowable. Adding 1 foot to the vessel diameter will reduce the design margin. The applicant should justify the use of the proposed tolerances. This is Open Item 14.3.2-14.

Open Item Number: 14.3.2-15

Original RAI(s): N/A

Summary of Issue: Section 3.7, "Design Reliability Assurance Program" (D-RAP). The staff found that the list of risk significant components in Table 3.7-1 was not updated to include all risk- significant structures, systems, and components (SSCs) from the list of risk significant SSCs identified in Tier 2 Section 17.4, Table 17.4-1, "Risk Significant SSCs within the Scope of D-RAP." Specifically, the list of risk significant components should include:

- Compressed and Instrument Air System Air Compressor Transmitter
- Passive Containment Cooling System Diverse (3rd) Motor Operated Drain Isolation Valve function
- In-containment Refueling Water Storage Tank Vents
- Normal Residual Heat Removal Valve V055 function
- Feedwater Isolation Valves

As discussed in Section 17.4 of this report, the staff determined that Table 17.4-1 contained an acceptable list of risk significant SSCs under the scope of D-RAP. In Table 17.4-1, the applicant also removed the safety related passive core cooling condensate sump re-circulation valves' automatic open function from the D-RAP for the AP1000 design and this should be reflected in ITAAC Table 3.7-1. This is Open Item 14.3.2-15.

Open Item Number: 14.3.3-1

Original RAI(s): N/A

Summary of Issue: Section 2.5.1, "Diverse Actuation System," Table 2.5.1-1, "Functions Automatically Actuated by the DAS" should be modified to include "actuate core makeup tanks, and trip the reactor coolant pumps on low wide-range steam generator water level." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-1.

Open Item Number: 14.3.3-2

Original RAI(s): N/A

Summary of Issue: Section 2.5.1 design description item 2(c) should be modified to include "the DAS manual control bypasses the protection and safety monitoring system cabinets." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-2.

Open Item Number: 14.3.3-3

Original RAI(s): N/A

Summary of Issue: Section 2.5.1 design description item 3(e) should be modified to include "The DAS uses sensors that are separate from those being used by the PMS and the plant control system." This comment is based on the review of the Tier 2 Section 7.7.1.11, "Diverse Actuation System." This is Open Item 14.3.3-3.

Open Item Number: 14.3.3-4

Original RAI(s): N/A

Summary of Issue: Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-1 and Figure 2.5.2-1 should be modified to include "two divisions of safety-related postaccident parameter displays" to be consistent with the Tier 1 Section 2.5.2 design description. This is Open Item 14.3.3-4.

Open Item Number: 14.3.3-5

Original RAI(s): N/A

Summary of Issue: Section 2.5.2, Table 2.5.2-4, “PMS Manually Actuated Functions,” is not consistent with the information provided in Tier 2 Table 7.2-4, “System-Level Manual Inputs to the Reactor Trip Functions,” and Table 7.3-3, “System-Level Manual Inputs to the ESFAS.” Tier 1 design description Item 6(c) should be modified to clarify that the functions listed on Table 2.5.2-4 is based on minimum inventory requirements. This is Open Item 14.3.3-5.

Open Item Number: 14.3.3-6

Original RAI(s): N/A

Summary of Issue: Section 2.5.2 design description Item 8(b) should be modified to clarify that the control transfer function is implemented by multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single non-safety-related group. The ITAAC table should reflect this feature. This is Open Item 14.3.3-6.

Open Item Number: 14.3.3-7

Original RAI(s): N/A

Summary of Issue: Section 2.5.2, Table 2.5.2-7, “PMS Interlocks,” should be modified to include “Interlocks for the Accumulator Isolation Valves and IRWST Discharge Valve” to be consistent with Tier 2 information provided in Section 7.6.2.3. This is Open Item 14.3.3-7.

Open Item Number: 14.3.3-8

Original RAI(s): N/A

Summary of Issue: Tier 1 Section 2.5.2, Table 2.5.2-6, “PMS Blocks,” should be modified to include (1) block automatic rod withdrawal (P-17) and (2) block automatic safeguards (P-4). This comment is based on the review of the Tier 2 Table 7.2-3, “Reactor Trip Permissives and Interlocks,” and Table 7.3-2, “Interlocks for Engineered Safety Features Actuation System.” This is open item 14.3.3-8.

Open Item Number: 14.3.3-9

Original RAI(s): N/A

Summary of Issue: Section 2.5.2, Table 2.5.2-8, ITAAC 7(c) columns do not have sufficient criteria to verify that the design commitment is met. Removal of power of non-safety components and review of gateway filtering is not enough. Language should be consistent with SRP 7.9, data communications, especially “control of access.” Language should also be consistent with acceptance criteria for other ITAAC’s in this section such as 7(a) and 7(b). This is Open Item 14.3.3-9.

Open Item Number: 14.3.3-10

Original RAI(s): N/A

Summary of Issue: Section 2.5.2, Table 2.5.2-8, ITAAC 7(d) columns may not be sufficient to verify the design commitment, especially the terminology “non-class 1E controls” in the performance of the operational tests. Language should be similar to other ITAAC’s in this section such as 7(a) and 7(b). This is Open Item 14.3.3-10.

Open Item Number: 14.3.3-11

Original RAI(s): N/A

Summary of Issue: Section 3.2, Table 3.2-1, “Acceptance Criteria for Design Commitment 3," should include the following as a last criterion: "Man-in-the loop engineering test reports." This is Open Item 14.3.3-11.

Open Item Number: 14.3.3-12

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, “Acceptance Criteria for Design Commitment 4,” should be changed to indicate that the verification and validation implementation plan includes the following activities (terminology to be consistent with NUREG-0711, Revision 1):

- Operational Conditions Sampling
- Design Verification
  (HSI Task Support Verification)
  (HFE Design Verification)
- Integrated System Validation
- Human Engineering Discrepancy Resolution
- Plant HFE/HSI (as designed at the time of plant start-up) verification.

This is Open Item 14.3.3-12.

Open Item Number: 14.3.3-13

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, "Design Commitment" statement No. 5, should be changed to indicate that the verification and validation implementation plan includes the following activities (terminology to be consistent with NUREG-0711, Revision 1):

- Operational Conditions Sampling
- Design Verification
  (HSI Task Support Verification)
  (HFE Design Verification)

- Integrated System Validation
- Human Engineering Discrepancy Resolution
- Plant HFE/HSI (as designed at the time of plant start-up) verification.

This is Open Item 14.3.3-13.

Open Item Number: 14.3.3-14

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, "Acceptance Criteria," for "Design Commitment" statement No. 5, should be changed to include a new, "a)" to indicate that, "a) Operational Conditions Sampling was conducted in accordance with the implementation plan." The remaining criteria should be re-lettered. This is Open Item 14.3.3-14.

Open Item Number: 14.3.3-15

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, "Inspections, Tests, Analyses," item "d)", should be changed to replace "design issues resolution" with "human engineering discrepancy resolution." This is Open Item 14.3.3-15.

Open Item Number: 14.3.3-16

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, "Acceptance Criteria," item "d)," should be changed to, "human engineering discrepancy resolution verification was conducted in accordance with the implementation plan and includes verification that human factors issues that were documented in the design issues tracking system and human engineering discrepancies that were identified in the design process have been addressed in the final design." This is Open Item 14.3.3-16.

Open Item Number: 14.3.3-17

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, “Acceptance Criteria,” Items 7.iii and 7.iv: These acceptance criteria do not relate to providing a suitable work space environment for MCR operators. There is nothing in Tier 1, Subsection 2.6.3, that evaluates the adequacy/effectiveness/suitability of illumination levels for the facility or the workstations in the facilities. As part of evaluating a suitable work space environment for the MCR and RSR, there should be an assessment of auditory levels (noise) as well. This comment also applies to Table 3.2-1, “Acceptance Criterion,” Item 10.ii. This is Open Item 14.3.3-17.

Open Item Number: 14.3.3-18

Original RAI(s): N/A

Summary of Issue: Table 3.2-1, item 10.i: Subsection 2.7.1 does not have an ITAAC related to RSR - there is nothing in the ITAAC that requires inspection, test, and analyses for the RSR and ventilation. Please clarify. This is Open Item 14.3.3-18.

Open Item Number: 14.3.3-19

Original RAI(s): N/A

Summary of Issue: In SECY-02-0059, the staff identified an issue to the Commission regarding the applicant's proposed use of piping DAC, which is different than the approach used in previous design certification applications. Westinghouse proposed to provide, as part of a COL application that references the AP1000 design, its analyses for piping design in which a leak-before-break (LBB) approach is used. In previous design certification reviews, the applicants provided as a part of design certification their bounding piping analyses in which an LBB approach was used. Accordingly, Westinghouse's approach for the AP1000 is to establish bounding curves at the design certification phase and to provide an evaluation of LBB piping at the COL phase. This is not consistent with Commission policy. Postponing the completion of analyses for LBB piping until the COL phase would leave open the question of whether there is sufficient margin in the piping to demonstrate that the probability of pipe rupture is extremely low. Thus, the finality of design might not be assured during the design certification review. The staff discussed this issue in Section 3.6.3.4 of this report. This is Open Item 14.3.3-19.

Open Item Number: 14.3.4-1

Original RAI(s): N/A

Summary of Issue: Control room χ/Q values are not provided in Table 5.0-1, "Site Parameters." In the staff's judgement these values should also be provided in the table as were the Exclusion Area Boundary and Low Population Zone χ/Q values. However, even when provided in Table 5.0-1, the control room χ/Q values remain an open item for the following reason. As part of its review of Table 15A-5, "Atmospheric Dispersion Factors (χ/Q) for Accident Analysis," in Tier 2, the staff initially asked the applicant if the methodology and all inputs and assumptions related to the control room χ/Q values would be evaluated as part of the COL review. The applicant provided a detailed response stating that the methodology, inputs and assumptions would be provided as part of the COL and noting additional information about the analysis. NRC staff issued a second RAI to inquire if the applicant was seeking certification of any of the AP1000 design values used as inputs to the χ/Q calculations. The applicant subsequently provided certain design-specific information that was used as input to the assessment and for which the applicant was seeking certification. The staff has not completed its evaluation of this response, 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. Pending completion of the review, this is open item 14.3.4-1.

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