June 6, 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 30 potential open items for DSER Chapter 19, “Severe Accidents,” 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.

Eighteen of the potential open items in the enclosure are new issues. The 12 other potential open items in the enclosure has its original request for additional information (RAI) number included for reference. 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 will 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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Westinghouse AP1000
Draft Safety Evaluation Report
Potential Open Items
Chapter 19
Severe Accidents

Open Item Number: 19.1.10.1-1

Original RAI(s): 720.035

Summary of Issue: Digital Instrumentation and Control

In Chapter 26 of the AP1000 PRA, a design option for the PMS in addition to the one modeled in the PRA is proposed. The option to use the Common Qualified Platform (Common Q) is proposed because of the rapid changes that are taking place in the digital computer and graphic display technologies employed in the modern human systems interface. The applicant assumes that the use of the Common Q option, in the place of the PRA PMS model, does not have any impact on the design certification process because such a process focuses upon the process used to design and implement instrumentation and control systems for the AP1000 rather than on the specific implementation. The staff requested the applicant (see RAI 720.035) to explain the process that will be used to verify that a PMS designed with the “Common Q” option will have equivalent or better reliability than the system modeled in the PRA and how the introduction of the “Common Q” option will affect important PRA-based insights about the PMS.

In its response to RAI 720.035, the applicant asserted that the PRA results are not sensitive to small changes in PMS failure probabilities and that the general architecture of the Common Q PMS is similar to that modeled in the AP1000 PRA. In addition, the applicant stated that the AP600 I&C functional requirements, which have received design certification, will be retained to the maximum extent compatible with the Common Q hardware and software. Also, it is stated that although the details of the AP1000 PRA model follow the AP600 design, the Common Q hardware and software provide a degree of redundancy that is equivalent to the redundancy modeled in the AP1000 PRA. However, the staff believes that further clarification of this issue is needed to ensure that the “Common Q” option will have the same or better reliability than the PMS design modeled in the PRA. A comparison of important features between the “Common Q” option and the PMS modeled in the PRA could help clarify this issue. This comparison should include features found by the PRA to be important contributors to the assumed high reliability of the PMS. Such a comparison, may identify the need for additional or different “design certification requirements” for the “Common Q” option” of PMS. This is Open Item 19.1.10.1-1.

Enclosure

Open Item Number: 19.1.10.1-2

Original RAI(s): 720.038

Summary of Issue: PRA Input to Design Certification Process

An important objective of the AP1000 design certification PRA is to identify important PRA insights and assumptions and make sure that they are addressed in the design certification through "design certification requirements," such as requirements for ITAAC, the requirement for a D-RAP and COL action items. These requirements will be incorporated in the DCD to ensure that any future plant which references the design will be built and operated in a manner that is consistent with important assumptions made in the design certification PRA.

In its response to RAI 720.038, the applicant provided a preliminary and, recently, a revised list of "design certification requirements." The staff expects the final list of "design certification requirements" to be in agreement with the resolution of all open items identified in the AP1000 DSER. The staff is still reviewing the list of "design certification requirements" proposed by the applicant, especially in light of assumptions and insights related to differences in PRA models between the AP600 and AP1000 designs (e.g., differences in assumptions made in the fire risk analysis). The staff expects the applicant to continue providing requested information to ensure that all important assumptions made in the design certification PRA are appropriately included in the final list of design certification requirements. This is Open Item 19.1.10.1-2

Open Item Number: 19.1.10.1-3

Original RAI(s): 720.039, 720.027, 720.030

Summary of Issue: PRA Input to RTNSS Process

An important objective of the AP1000 PRA is to provide risk-based input to design certification regarding the need for regulatory oversight of certain non-safety-related systems (RTNSS). The same process used in the AP600 design certification is also used in the AP1000 design certification. The staff asked the applicant (see RAI 720.039) to use the results of the AP1000 PRA to provide input to the RTNSS process. Although the applicant has identified RTNSS systems (documented in WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Revision 1, April 2003), the NRC staff do not have enough information to make a determination regarding the proper use of PRA results and insights in the RTNSS process.

The staff requested the applicant provide all steps in the process of using PRA results to identify non-safety-related systems for regulatory oversight as well as the type and level of such oversight. These steps are needed for the following reasons:

- show the link between the plant risk when only safety-related systems are credited in the PRA and the plant risk when the selected systems for regulatory oversight (including the type and level of such oversight) are credited in the PRA

- compare to the probabilistic criteria (safety goals, including the containment performance goal) documented in Section 19.1.7 of this report and SECY-94-084

- account for important uncertainties in the AP1000 PRA models (as stated in SECY-94-084). Examples of such uncertainties are discussed in the sensitivity studies documented in Section 19.1. 3.1.5 of this report. In addition, the uncertainty in the large LOCA initiating event frequency assumed in the PRA (see RAI 720.027) and the uncertainty in the success criteria used for passive containment cooling by air flow (see RAI 720.030) should be addressed.

This information has not been provided by the applicant. This is Open Item 19.1.10.1-3.

Open Item Number: 19.1.10.1-4

Original RAI(s): 720.027

Summary of Issue: Impact of Uncertainties on PRA Results and Conclusions

The staff review has identified two AP1000-specific areas of uncertainty which individually, or collectively with other areas of uncertainty (e.g., uncertainty associated with failure probabilities of squib valves), have the potential to affect the PRA results and conclusions regarding the need for “certification requirements,” such as ITAACs, RTNSS and COL action items. These uncertainties have also the potential to increase the number of “low margin risk significant” sequences which should be analyzed conservatively to bound thermal-hydraulic (T-H) uncertainty and determine success criteria for systems and operator actions. These two areas of uncertainty are discussed below.

One area of uncertainty is related to initiating event frequencies assumed in the PRA. The staff requested additional information (see RAI 720.027) about differences in initiating event category frequencies used in the AP600 and the AP1000 PRAs for large LOCAs and SGTR accidents. The applicant’s response to RAI 720.027 did not address adequately the basis for the decrease of such initiating event frequencies which have a significant impact on the PRA results. For the large LOCA category, the applicant states that in the AP1000 PRA “operating experience” data reported in NUREG/CR-5750 for pipe breaks were used. However, the NUREG/CR-5750 data rely on expert opinion and include significant uncertainty. In addition, since NUREG/CR-5750 was published additional information (e.g., Davis Besse finding) is available. For SGTR events, the frequency used in the AP1000 PRA is based on a more recent calculation that was performed in conjunction with a replacement steam generator project which is proprietary to the applicant. The staff believes that the impact of uncertainties on PRA results and insights, associated with the frequencies of large LOCAs and SGTR accidents assumed in the AP1000 PRA, needs to be investigated and addressed appropriately by the design certification process.

A second area of uncertainty is related to the success criteria assumed in the AP1000 PRA for passive containment cooling by air flow. The AP1000 PRA event trees include a top event for containment cooling (event CHR). It is stated that *"For success paths that result in steam release to the containment, the success of containment cooling (PCS or RNS) is modeled. If containment cooling is successful, then the path ends in an OK state. If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made."* This "special OK" end state is labeled "late containment failure (LCF)" end state and defined as an end state *"...where the containment heat removal by either passive containment cooling system (PCS) or component cooling water (CCS) heat exchangers via normal residual heat removal (RHR) fails."* The staff requested clarification (see RAI 720.030) about the meaning of the "special Ok" status. The applicant responded that a sensitivity study shows that even if the LCF state is considered to be a core damage, the plant CDF would increase by only 29 percent. The staff needs further information regarding the impact of this assumption on the focused PRA, where no credit is taken for the non-safety-related systems, and on the RTNSS process.

The impact of these two areas of uncertainty on the results of the PRA, including the PRA results used in the RTNSS process, should be addressed in the design certification process. This is Open Item 19.1.10.1-4.

Open Item Number: 19.1.10.1-5

Original RAI(s): 720.009, 720.012, 720.013, 720.014, 720.017, 720.021, 720.024, and 720.025

Summary of Issue: The applicant has utilized a systematic approach in categorizing success paths for the PRA for the purpose of minimizing the number of analyses need to justify success. In many cases the MAAP4 code was used to identify the limiting sequences. To justify that the limiting sequences provide for adequate core cooling, the applicant has performed bounding analyses using conservative computer codes that the NRC staff has reviewed for design basis accidents. In some instances the NRC staff has identified limiting sequences that have not been bounded. Other sequences have not been analyzed for AP1000 but success has been inferred by the applicant from analyses performed for AP600. These deficiencies are an open item in the AP1000 DSER.

The deficiencies are listed below with reference to the NRC staff RAI where the issue was first raised:

(a) Additional justification in needed for long-term cooling analyses for which the initial and boundary conditions were obtained from analyses using MAAP4 for input into WCOBRA/TRAC (RAI 720.013)

(b) Additional justification should be provided that a large break LOCA can be mitigated if one of the two CMTs fail (RAI 720.012-2)

(c) Additional justification should be provided that adequate water can be maintained within the containment to provide for long term core cooling if containment isolation fails (RAIs 720.021 and 720.024)

(d) Additional justification should be provided that one of the two startup feedwater pumps can deliver adequate water to the two steam generators following an ATWS event (RAI 720.024)

(e) Additional justification should be provided that evaluations made for AP600 are appropriate to be used in the AP1000 PRA Table 6-1 and in the response to RAI 720.025 where the applicant assumes that 30 minutes of core cooling is available following a small break LOCA, steam generator tube rupture or transient with no accumulator injection (RAIs 720.024 and 720.025)

(f) Additional justification should be provided that sequences which assume failure of one of the four ADS stage 4 valves and also assume failure of containment isolation, will end in successful core cooling (RAIs 720.012-014, 720.009 and 720.017).

This is Open Item 19.1.10.1-5.

Open Item Number: 19.1.10.1-6

Original RAI(s): n/a

Summary of Issue: Fire-Specific Operator Actions

The fire PRA identified the following two fire-specific operator actions:

(1) Operator action to switch off the electrical power for each division in case of fire to avoid spurious action of valves

(2) Operator action to manually actuate a valve to allow fire water to reach the automatic fire suppression system in containment maintenance floor (fire area 1100 AF 11300B)

The COL applicant will develop procedures for implementing these fire-specific operator actions. This is COL Action Item 19.1.10.1-1 and Open Item 19.1.10.1-6

Open Item Number: 19.1.10.2-1

Original RAI(s): 720.099

Summary of Issue: Shutdown Risk due to Vacuum Refill Operations

The applicant stated that the shutdown risk due to vacuum refill operations are included in the calculation of shutdown risk during vented drained conditions. The staff is reviewing the applicant's response to RAI 720.099 (dated 3/28/03 and 4/12/03) to determine if the shutdown risk due to vacuum refill operations is included in the calculation of shutdown risk during vented drained conditions. The staff noted during their review of the applicant's response to RAI 720.099 that investment protection short term availability controls do not include RNS and its support systems such as component cooling water system, service water system, and ac power supplies during vacuum refill operations. Assuming an extended loss of RNS during vacuum refill operations, the staff questions using the RNS suction relief valve to relief RCS pressure should the operators not open the ADS valves. The operators may instead isolate the RNS suction relief valve to isolate RCS leakage. This is Open Item 19.1.10.2-1.

Open Item Number: 19.1.10.2-2

Original RAI(s): n/a

Summary of Issue: Dominant Shutdown Accident Sequences

The applicant did not report the dominant shutdown accident sequences in the AP1000 shutdown PRA. The staff requests Westinghouse to report the dominant shutdown accident sequences in the AP1000 shutdown PRA. This is Open Item 19.1.10.2-2.

Open Item Number: 19.1.10.2-3

Original RAI(s): 720.038

Summary of Issue: Shutdown Risk Importance Analysis

As requested by the staff in the follow up to RAI 720.038, the applicant did not provide any importance analyses (such as risk achievement and risk worth) in their response to RAI 720.038 (dated 03/28/03 and 4/12/03). This is Open Item 19.1.10.2-3.

Open Item Number: 19.1.10.2-4

Original RAI(s): 720.038

Summary of Issue: Documentation of Shutdown Focused PRA Results

The "focused" PRA shutdown CDF was estimated to be 1.23E-6. Over eighty five percent of the risk resulted from a loss of offsite power during drained conditions and during non-drained conditions. Some of the dominant cutsets have the basic event IWX-MV-GO1. In the early versions of the AP600 PRA, the basic event, IWX-MV-GO1, was used to model common cause failure of the 4 out of 4 IRWST injection motor operated valves (MOVs) to open. The later versions of the AP600 design and the AP1000 design changed the 4 MOVs to squib valves. In the AP 1000 design, the low pressure squib valves (120 A/B) in the recirculation lines were changed to high pressure squib valves. In preparing the AP600 cutset file for use as the starting point in creating the AP1000 shutdown mode, the basic event IWX-MV-GO1 was changed to, IWX-EV-SA, common cause failure of IRWST squib valves. This basic event has a failure probability of 2.6E-5. As a follow-up to RAI 720.38, the staff needs to understand why basic event IWX-MV-GO1 appears in the "Focused PRA" cutsets for the AP1000 design. The staff also needs a list of basic events and their description for the AP1000 shutdown model. This is Open Item 19.1.10.2-4.

Open Item Number: 19.1.10.2-5

Original RAI(s): 720.038

Summary of Issue: Shutdown Fire Risk Evaluation

The applicant submitted the AP1000 shutdown fire risk evaluation on 3/28/03. The AP1000 fire risk analysis has a different grouping of fire areas and different combustible loadings than the AP600 shutdown fire risk evaluation. Therefore, the adequacy of the AP1000 shutdown fire risk evaluation is still being reviewed by the staff. This is Open Item 19.1.10.2-5.

Open Item Number: 19.1.10.2-6

Original RAI(s): n/a

Summary of Issue: Shutdown PRA Sensitivity Studies

The staff will confirm that the results of the sensitivity studies (including cutsets) are documented into the AP1000 Shutdown PRA. This is Open Item 19.1.10.2-6.

Open Item Number: 19.1.10.3-1

Original RAI(s): n/a

Summary of Issue: Representative Sequences for Assigning Source Terms

The accident sequences used to represent the various release categories are identified and briefly described in PRA Chapter 45. Additional sequence information is provided in PRA Chapter 34. The basis for selecting the representative sequence for each release category is not provided. Such information is necessary in order to confirm that the sequence selected to represent each release category is reasonably representative of the collection of sequences assigned that category, in terms of the magnitude, timing, energy, and elevation of release. Based on the limited information that was provided, the staff noted a number of inconsistencies. Specifically, for release category CFE releases from the ADS Stage 4 valves enter directly into containment rather than into the IRWST, and given the location of the valves relative to the containment shell, would not result in containment failure from diffusion flames as assumed in the PRA. For release category CFL containment failure is assumed at 3 hours, which is inconsistent with the time frame for late containment failure. Also, important details impacting the release characteristics need to be documented, such as whether an additional decontamination factor is credited in determining the source term for SGTR events (as it was in AP600), and the containment isolation failure location and size assumed for containment isolation sequences. This is Open Item 19.1.10.3-1.

Open Item Number: 19.1.10.3-2

Original RAI(s): n/a

Summary of Issue: Major Contributors to System Failures

The major causes of reactor cavity flooding failure and hydrogen igniter failure in AP1000 have not been provided. Such information is useful for identifying major contributors to system failure and confirming that reasonable measures have been taken to reduce risk. The staff will request that the applicant provide this information for AP1000. This is Open Item 19.1.10.3-2

Open Item Number: 19.3.3-1

Original RAI(s): 440.110

Summary of Issue: The applicant provided the following information for the design features that reduce risks associated with temporary RCS boundaries for the AP1000:

Reduced reliance on freeze seals

Freeze seals are used for repairing and replacing components such as valves, pipe fittings, pipe stops and pipe connections when it is impossible to isolate the area of repair any other way. Industrial experience indicates that some freeze seals have failed in nuclear power plants and resulted in significant events. In addressing the issue of freeze seals failure, the AP1000 design reduces the potential applications of freeze seals by reducing the number of lines that connect to the RCS and by providing the ability to perform inservice tests (ISTs) on many valves that connect to the RCS pressure boundary. The IST program reduces the requirements for disassembling RCS pressure boundary valves to perform operability tests. The use of freeze seals during a forced outage typically occurs in cold shutdown (Mode 5). During Mode 5, the PXS is required by the TSs (DCD Tier 2 Chapter 16, Table B 3.0-1) to be available, and therefore, the PXS can respond to a loss of coolant through a failed freeze seal.

The staff finds that the reduction of RCS penetrations, the ability to perform ISTs, the use of fixed incore system, and higher nozzle dam design pressure will reduce the risks associated with the loss of temporary RCS boundaries. Therefore, the staff concludes that the design relative to temporary RCS boundaries are acceptable.

DCD Tier 2 Section 13.5, "Plant Produces," contains COL information items requiring plant procedures. However, the COL applicant should develop plant specific guidelines that would reduce the potential for loss of RCS boundary and inventory when using freeze seals. This COL information is not specified in the DCD. Therefore, this is Open Item 19.3.3-1 and COL Action Item 19.3.3-1.

Open Item Number: 19.3.7-1

Original RAI(s): n/a

Summary of Issue: Outage Planning and Control

The technical findings of NUREG-1449 supports the determination that a comprehensive program for planning and controlling outage activities would reduce risk during shutdown, by reducing the frequency of precursor events. The staff realizes that the ultimate responsibility for outage planning and control is within the scope of the plant owners and considers this a COL action item.

DCD Tier 2 Section 13.5.1 requires the COL applicants to develop plant procedures for normal and abnormal operations; emergency operation; refueling and outage planning; alarm response; maintenance; inspection; test and surveillance, as well as administrative controls. This is COL Action Item 19.3.7-1.

The staff will review the COL applicants' outage planning and control program, and the COL applicants will have to appropriately address the factors that improve low-power and shutdown operations. As a minimum, these factors will include the following important elements:

- an outage philosophy which includes safety as a primary consideration in outage planning and implementation,

- separate organizations responsible for scheduling and overseeing the outage; provisions for an independent safety review team that would be assigned to perform final review and grant approval for outage activities,

- control procedures which address both the initial outage plan and all safety-significant changes to schedule,

- provisions to ensure that all activities receive adequate resources,

- provisions to ensure defense-in-depth during shutdown and ensure that margins are not reduced; an alternate or backup system must be available if a safety system or a defense-in-depth system is removed from service, and

- provisions to ensure that all personnel involved in outage activities are adequately trained; this should include operator simulator training to the extent practicable; other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing.

This COL information is not specified in the DCD. Therefor, this is Open Item 19.3.7-1 and COL Action Item 19.3.7-2.

Open Item Number: 19.3.10-1

Original RAI(s): 720.038

Summary of Issue: In a letter dated March 28, 2003, the applicant responded to RAI 720.038 by providing an evaluation of plant risk associated with internal floods at shutdown. The objective of this study was to confirm that the design incorporates adequate capability to achieve safe shutdown following these events, by showing that the associated plant risk is sufficiently small. Deterministic criteria were used to screen out any areas in which the risk from flooding is clearly insignificant, on the basis of the lack of flood initiation sources or absence of equipment important to safe shutdown, as modeled in the internal events PRA. Because the plant is already in shutdown, an initiating event for the shutdown analysis was considered an event leading to a threat to equipment needed for the normal decay heat removal function.

Based on the staff's preliminary review of this letter, it appears to have errors in the calculated CDF for two of the eight sequences. The applicant needs to address these errors and the staff needs to complete its review. This is Open Item 19.3.10-1.

Open Item Number: 19.4-1

Original RAI(s): 720.060

Summary of Issue: In a revised RAI response dated March 31, 2003, the applicant provided an updated evaluation addressing these concerns. The staff has not completed of its evaluation of SAMDAs for AP1000. Therefore, this is Open Item 19.4-1.

Open Item Number: 19A.2-1

Original RAI(s): n/a

Summary of Issue: The applicant included the following basic factors for the seismic margin calculation:

- Deterministic strength factor
- Variable strength factors
- Material
- Damping
- Inelastic energy absorption/ductility
- Analysis or modeling error
- Soil-structure interaction

Deterministic Strength Factor

The deterministic design process involves the use of: (1) actual stress that is less than the allowable value specified in the design code, and (2) the margin used in the code allowable values by the code or standard developing body. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-1.

Open Item Number: 19A.2-2

Original RAI(s): n/a

Summary of Issue: Variable Strength Factors

Variability exists between the design capacity and the test capacity. This phenomenon is inherent in the manner in which an actual structure redistributes loads based on redundancy, excess capacity provided by design, end constraints and other factors. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-2.

Open Item Number: 19A.2-3

Original RAI(s): n/a

Summary of Issue: Material

The allowable stress values provided in codes and standards are based on minimum specified yield strength in tension or compressive strength in crushing. Consequently, actual material properties that are derived from the yield strength or crushing strength have variability. The applicant has not explained how this factor was used in its probabilistic fragility analysis. This is Open Item 19A.2-3.

Open Item Number: 19A.2-4

Original RAI(s): n/a

Summary of Issue: Analysis or Modeling Error

Modeling error stems from a number of sources that include stiffness parameters, modeling of masses due to live load, connectivity between structural members, support conditions, and others. The applicant did not explain how this factor was used in its probabilistic fragility analysis of various structures and equipment. For the modal frequency variation, the applicant used a composite logarithmic standard deviation, $ß_c$, of 0.3. The use of a $ß_c$ value of 0.3 means that modal frequency values can vary by a factor of 1.8. The applicant needs to justify the use of such a high variability factor for the natural frequency calculations when using detailed finite element models. This is Open Item 19A.2-4.

Open Item Number: 19A.2-5

Original RAI(s): n/a

Summary of Issue: Soil-Structure Interaction

How the structure behaves with the foundation material in which the structure is embedded when subjected to seismic excitation, is analytically determined by the soil-structure interaction (SSI) analysis. For design purposes, the soil parameters are varied by a factor of 2 higher and lower, then the results are enveloped. Consequently, the SSI effect can introduce a considerable variation in the calculated margin. However, the AP1000 design is to be located on hard rock sites and no SSI analysis is involved in its design. Therefore, the discussion about the SSI related variability in Chapter 55 of the PRA report for AP1000 is inappropriate, since the use of the variability factor $(\beta_c)_{SSI}$ is not justified. This issue is Open Item 19A.2-5.

Open Item Number: 19A.2-6

Original RAI(s): n/a

Summary of Issue: Conservative Deterministic Failure Margin Method (CDFM)

The applicant used the CDFM method to calculate the HCLPF value of the shield building using strength, inelastic energy absorption and damping as areas where the shield building capacity is increased over the design capacity to determine the cumulative effect of those factors. The applicant has increased the shear capacity of a concrete section by increasing the shear modulus to account for the shear strength of reinforcement bars where the shear load exceeds the shear strength of concrete alone. The ACI 349 Code, the applicable concrete design code, allows the addition of reinforcement strength, but not by increasing the shear modulus of the concrete section. The shield building tension ring has a HCLPF value of 0.51 g (Table 55-1, Sheet 1 of 4). Therefore, a validation of the capacity of the shield building shear walls is important. With respect to inelastic energy absorption and damping factors, it is not clear as to whether or not the applicant has double counted damping values through the use of hysteretic damping for inelastic energy absorption and a damping value of 10 percent. The applicant needs to justify the details of the CDFM approach for calculating HCLPF values for important structures and equipment. It should be noted that the containment internal structure and the nuclear island basemat are predicted to lift up under the SSE loading. As noted in Section 3.7 of this report, the effect of uplift due to design basis seismic excitation is an open area. Consequently, at 0.5 g review level earthquake, the capacity of the tension ring could potentially be lower. Therefore, the validation of HCLPF values calculated by the CDFM approach is Open Item 19A.2-6.

Open Item Number: 19A.2-7

Original RAI(s): n/a

Summary of Issue: The applicant determined the HCLPF values on the basis of the estimated lower bound of qualification test results. When natural frequencies were not known, it was assumed that the equipment natural frequency coincides with the response spectra peak. When equipment frequencies are known and used for comparing the required response spectra (RRS) to the test response spectra (TRS), this information is to be included in the design specification. The applicant has not identified any equipment for which such design specification will be included. Although the applicant appears to have used a conservative approach to obtain the equipment HCLPF value from test results, it not clear how the use of known natural frequency values for equipment within the standard design scope will be implemented. Since there are many electrical components with HCLPF values at 0.54 g and one at 0.53 g, electrical components may become critical in determining the plant HCLPF value. This is Open Item 19A.2-7.

Open Item Number: 19A.2-8

Original RAI(s): n/a

Summary of Issue: Deterministic Approach

The applicant used the deterministic approach to estimate the HCLPF values of primary system component supports. The components included in this approach are: polar crane, baffle plate supports, heat exchanger for the passive residual heat removal system, core makeup tank and valves. The applicant used lower bound values, and it appears that there was no need for invoking factors of conservatism to arrive at the HCLPF values. It is noted that the core makeup tank has a HCLPF value of 0.54 g; therefore, any increase in seismic response of the containment internal structure due to lift off of the internal structure or the nuclear island structure would necessitate a review of this HCLPF value. This is Open Item 19A.2-8.

Open Item Number: 19A.2-9

Original RAI(s): n/a

Summary of Issue: Generic Fragility Data

When HCLPF values could not be determined by using one of the methods described above, Westinghouse used generic fragility data. The cases where this approach was used are the following:

- Reactor internals and core assembly that includes fuel
- Control rod drive mechanism (CRDM) and hydraulic drive units
- Reactor coolant pump
- Accumulator tank
- Piping
- Cable trays
- Valves
- Main control room operation and switch stations
- Ceramic insulators
- Battery racks

The generic fragility data came from the Utility Requirements Document which was reviewed by the NRC. Therefore, the use of generic fragility data developed by a joint industry group in the Utility Requirements Document is acceptable. However, the applicant has not indicated what amplification factor, if any, was used to adjust the generic fragility data for the AP1000 configuration. The PCS water flow transmitter, located at Elevation 261' with a HCLPF value of 0.53 g, is likely to have an amplified seismic response. The applicant needs to justify the HCLPF values in the range of 0.53 g and 0.73 g that were obtained from the generic data as shown in the AP1000 PRA Table 55-1, Sheet 3 of 4. This is Open Item 19A.2-9.

Open Item Number: 19A.3-1.

Original RAI(s): n/a

Summary of Issue: Major SMA Model Assumptions

The applicant has used a PRA based seismic margin analysis method similar to the AP600 plant. In conducting its SMA, the applicant made the following assumptions:

- Seismic events occur at full power
- The review level earthquake (RLE) is 0.5 g
- The loss of offsite power occurs at the RLE. No credit is taken for non-safety related diesel generators for on-site AC power
- No credit is taken for non-safety related systems

- Initiating seismic event categories are derived from the AP600 model and the min-max method was used to calculate the plant HCLPF value

The staff notes that the seismic response of the AP1000 structures and some primary system components could be higher than those in AP600, because the height of the containment and the overall mass of AP1000 plant have increased. As indicated in the previous section of this report, it will be necessary to resolve the open items prior to the acceptance of the validity of plant seismic event trees derived from the AP600 model. This is Open Item 19A.3-1.

Open Item Number: 19A.3-2.

Original RAI(s): n/a

Summary of Issue: Initiating Event Category HCLPFs

For all seismic event categories, except for the EQ-LOSP category, the HCLPF values of various seismic initiating event groups exceed 0.5 g. Each category of HCLPF group is discussed further below:

- EQ-STRUC Group: The lowest HCLPF value of the Nuclear Island (NI) structure that can influence the plant HCLPF value is .05 g, based on the values shown in Table 55-1 of the AP1000 PRA. The HCLPF values shown in Table 55-1 need to be validated through the resolution of Open Item 19A2-8 discussed in the previous section. The applicant has assumed that there is no detrimental effect from any seismic interaction between the NI and the adjacent turbine, annex, diesel generator and radwaste building structures. The applicant has stated, “this assumption needs to be verified by a plant walkdown when an AP1000 plant is built.” However, there is no entry on the COL interface requirement about the plant walkdown in Table 1.8-2 of the DCD. There is an entry in Table 1.8-2 19.59.10-1, “As-Built SSC HCLPF Comparison to Seismic Margin Evaluation.” The applicant needs to justify why a specific item on plant walkdown verification of seismic interaction between the NI and adjacent structures is not included in the COL interface requirement. This is Open Item 19A.3-2.

Open Item Number: 19A.3-3.

Original RAI(s): n/a

Summary of Issue:

- EQ-SLOCA: The applicant included a number of elements of seismic fragility in this group. These elements include, simultaneous failure of all small diameter instrument lines, steam generator tube rupture, and large steam line breaks. Steam generator tube rupture event considers up to 5 simultaneous tube ruptures. The EQ-SLOCA grouping appears reasonable. However it is not clear if the applicant considered degradation of steam generator tubes under the full service life of steam generators for developing the seismic fragility. The applicant should explain how service related degradation of steam generator tubes was considered in the development of the HCLPF value of this group. This is Open Item 19A.3-3.

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