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April 11, 2003

SUBJECT: Transmittal of Westinghouse Responses to US NRC Requests for Additional Information on the AP1000 Application for Design Certification

This letter transmits the Westinghouse responses to NRC Requests for Additional Information (RAI) regarding our application for Design Certification of the AP1000 Standard Plant. A list of the RAI responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the RAI responses.

Please contact me if you have questions regarding this submittal.

Very truly yours,

M. M. Corletti
Passive Plant Projects & Development
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/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1571"

2 Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated April 2003

D063

DCP/NRC1571

April 11, 2003

Attachment 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1571"

Attachment 1

Table 1 "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1571"
100.003, Rev. 1 470.011, Rev. 1 471.011, Rev. 1 472.003, Rev. 1 480.010, Rev. 0 480.011, Rev. 0

Attachment 2

Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission
Requests for Additional Information dated April 2003

RAI Number: 100.003 (Response Revision 1)

Question:

With respect to the regulatory treatment of non-safety systems (RTNSS), is the process you utilized to determine which AP1000 systems are designated as RTNSS systems (and subsequently covered by your investment protection short-term availability controls stated in Section 16.3 of the DCD) consistent with that utilized for the AP600 and the process discussed in NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," Chapter 22? If any differences exist, please describe. Has an AP1000-specific review (based on the AP1000 design and probabilistic risk assessment (PRA)) of which systems should have regulatory treatment been performed? If not, please justify.

Westinghouse Response:

WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process" provides the basis for the implementation of the RTNSS policy statement. The AP1000 implementation of the RTNSS policy statement is consistent with the approach that was taken for the AP600. Note that PRA sensitivity studies with nonsafety systems failed were used to evaluate the importance of nonsafety features in PRA accident mitigation, instead of the focused PRA sensitivity study. The AP1000 PRA, revision 0, includes the CMF sensitivity study; the LRF sensitivity study was performed recently based on the response to RAI 720.057 and will be added to the PRA in revision 1.

Use of these sensitivity studies increases the need for passive safety feature actuation signals since non-safety AC power is assumed to be available after each accident except for loss of offsite power. As a result, some non-safety manual Diverse Actuation System (DAS) controls are required to meet the licensing PRA safety goals. These manual DAS controls are captured as RTNSS important and additional regulatory oversight is provided. Since these manual DAS controls meet the technical specification screening criteria for PRA importance, a Technical Specification is added on these manual DAS controls. The additional Technical Specification is included in the revised AP1000 Technical Specifications that are submitted as part of the response to RAI 630.001.

These DAS manual controls were the only additional non-safety featured captured in the AP1000 RTNSS evaluation. The list of non-safety features that have short-term availability controls is the same for AP1000 and AP600.

Design Control Document (DCD) Revision:

See the attached Technical Specification for DAS that is included in the revised AP1000 Technical Specifications.

PRA Revision:

None

NRC Additional Comments:

a. Need list of technical specification task force (TSTF) items that were incorporated upon adopting technical aspects of NUREG-1431, Rev. 2

b. Technical Specification (TS) 3.3.5 Required Action C.1 references wrong surveillance requirement (SR) - according to proposed Bases, it should be SR 3.3.2.2

c. TS 3.3.5 Required Actions B.1 and C.1 Completion Times of "Once per 31 days on a STAGGERED TEST BASIS" should be discussed and justified in the associated Bases. Also, these actions should specify a Completion Time for the initial performance of the test, followed by the periodic testing thereafter (see TS 1.3 Example 1.3-7). Otherwise, please explain how the proposed testing schedule would work and provide an example; TS 1.4 does not appear to explain such a Frequency.

Westinghouse Additional Response:

a. As discussed in the response to RAI 630.001, the AP1000 Technical Specifications (TS) update to NUREG-1431, Rev. 2 does NOT include incorporation of any TSTF improvements for differences or improvements beyond Rev. 2 of the STS. Outstanding TSTFs are periodically incorporated into the current revision of NUREG-1431. Therefore, the AP1000 Design Certification approach is to update the AP1000 TS to a specific revision of NUREG-1431. This approach simplifies AP1000 TS development and precludes the need to separately track incorporated and/or outstanding TSTFs as part of the design certification process. The AP1000 DCD 16.1 includes a COL action to update TS. The AP1000 TS will be updated to the current revision of the STS by the COL at the time of plant licensing, and any appropriate outstanding TSTFs as required by the NRC at that time.

b. The reference to SR 3.3.2.3 was a typographical error in preparing TS 3.3.5. Technical Specification (TS) 3.3.5 Required Action C.1 will be revised to correctly reference SR 3.3.2.2, which is consistent with the Bases discussion.

c. The Completion Times for Required Actions B.1 and C.1 were identified based on the predominant equipment failures that established the need for the associated DAS manual controls, as discussed in the Bases for Required Actions B.1 and C.1. The periodic testing interval specified in the Completion Time increases the test frequency for surveillance of the associated protection system equipment in each action from once every 92 days to once every 31 days.

Therefore, these Completion Times increase the testing frequency of the reactor trip breakers and safeguards actuation logic by a factor of 3, which helps to compensate for the unavailability of the DAS manual reactor trip or component actuation controls. The significantly increased testing frequency makes it more likely to detect any potential equipment failures while the DAS manual control is inoperable.

The original justifications for the reducing the Completion Time for Actions B.1 and C.1, provided in the second paragraphs for the Bases for Actions B.1 and C.1, have been revised to provide additional information, as shown below.

The periodic testing for proposed Required Actions B.1 and C.1 is expected to be consistent with the discussion for Condition A of Example 1.3-6 in the AP1000 TS. Since the periodic testing for the SRs referenced in Conditions B and C is performed on a staggered test basis, it's expected that the first division tested for Required Actions B.1 and C.1 would occur within 31 days of entering Conditions B or C. Subsequent divisions for each SR would be tested within 62 days, 93 days, and so on, of entering Condition B or C, until the respective condition was exited.

Since there is no note excluding the requirements of SR 3.0.2, it is applicable for Required Actions B and C. Therefore, there is no time extension for performance of the first division tested within 31 days of entering Condition B or C, but the 25 percent extension is available for subsequent divisions tested. It is expected that the normal sequence for testing divisions for these SRs would be continued while performing Required Actions B.1 and C.1. However, it is also possible that the sequence could be changed where this may be beneficial. For example, if corrective maintenance were performed on a component in a division and the associated SR is required as post-maintenance testing to restore the equipment to operable status, the re-test would satisfy Required Actions B.1 or C.1. The normal periodic 92-day test frequency requirements are still required to be satisfied for all divisions while in Conditions B or C, since the normal test frequency is still applicable.

In the process of preparing this RAI response, an editorial error for reactor trip instead of actuation was found in the Bases for Required Action C.1, and will be corrected as shown below. In addition, the following change will be made to the first paragraph in the Bases Background.

DCD Revision 3 incorporates the changes identified in the original response to this RAI. DCD Revision 3 will be updated to address the following change.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

<u>Required Action C.1 of TS 3.3.5 will be revised as follows:</u>

C.1 Perform SR 3.3.2.2~~3~~.

<u>The first paragraph in the Background to the Bases of TS 3.3 5 will be revised as follows:</u>

The Diverse Actuation System (DAS) manual controls provide **non-Class 1E** backup controls in case of common mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the AP1000 PRA. These **DAS manual** controls are not credited for mitigating accidents in the DCD Chapter 15 analyses.

<u>The Bases for Required Action B.1 of TS 3.3.5 will be revised as follows:</u>

Required Action B.1 requires SR 3.3.1.5, "Perform TADOT" for the reactor trip breakers, ~~is~~ to be performed once per 31 days, instead of~~n~~ once every 92 days. **Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action B.1. The initial performance of SR 3.3.1.5 on the first division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition B. The normal surveillance test frequency requirements for SR 3.3.1.5 must still be satisfied while performing SR 3.3.1.5 for Required Action B.1.** The predominant failure requiring the DAS manual reactor trip control is common mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry. **Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the reactor trip breaker surveillance.frequency from once every 92 days to once every 31 days.**

The Bases for Required Action C.1 of TS 3.3.5 will be revised as follows:

Required Action C.1 requires SR 3.3.2.2, "Perform ACTUATION LOGIC TEST," to be performed once per 31 days, instead of~~n~~ once every 92 days. **Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action C.1. The initial performance of SR 3.3.2.2 on the first division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition C. The normal surveillance test frequency requirements for SR 3.3.2.2 must still be satisfied while performing SR 3.3.2.2 for Required Action C.1.** The predominant failure requiring the DAS manual **actuation** ~~reactor trip~~ control is common mode failure of the PMS actuation logic software or hardware. This change in surveillance frequency for actuation logic testing increases the likelihood that a common mode failure of the PMS actuation logic from either cause would be detected while any DAS manual actuation control is inoperable. This reduces the likelihood that a diverse component actuation is required. It is not required to perform a TADOT for the manual actuation control device since the manual actuation control devices are very simple and highly reliable. **Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the automatic actuation surveillance.frequency from once every 92 days to once every 31 days.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.011 (Response Revision 1)

Question:

(Appendix 15B, paragraph 15B.2.6) The paragraph presents a qualitative discussion of the differences between AP600 and AP1000 designs concluding that the use of the AP600 removal coefficients is conservative. Please, provide either a sample calculation, or an analytical justification for this conclusion. Also, one potentially important difference is omitted, i.e., the increased height of the AP1000 containment. It is known that the increased height decreases the rate of aerosol removal, which would be a non-conservative effect. Please, discuss the significance of this issue.

Westinghouse Response Revision 1:

The analytical justification stated in DCD paragraph 15B.2.6 is:

- An increase in the power level results in an increase in the amount of decay heat that will be removed by condensation onto the containment shell and heat transfer to the containment shell thus enhancing the removal of particulates by diffusiophoresis and thermophoresis respectively.
- An increase in the projected containment pressure results in an increase in the rate of heat transfer by condensation onto the containment shell and heat transfer to the containment shell thus enhancing the removal of particulates by diffusiophoresis and thermophoresis respectively.

This discussion does not address the impact of the increase in containment height which will have an adverse impact on the sedimentation removal. Additionally, the increase in containment volume associated with the increased height adversely affects the removal of aerosols through diffusiophoresis and thermophoresis because of the reduction in aerosol concentration. An evaluation has been performed to determine the impact of the increase in containment height on the overall aerosol removal coefficients (without taking into account the impact of the increase in decay heat). The impact of the height increase is to reduce the calculated removal coefficients by ~20%. However, the aerosol removal coefficients reported in DCD Table 15B-1 include an arbitrary reduction of 0.1 hr^{-1} (the value of this reduction ranges from 12% to 19%). The negative impact of the increase in containment height can be eliminated in large part by simply removing this arbitrary conservatism.

A preliminary analysis has been performed to calculate aerosol removal coefficients that are specific to the AP1000. This analysis includes both the change in containment height and the thermal-hydraulic data for the AP1000 in place of the AP600 thermal-hydraulic data. The analysis confirms that the aerosol removal coefficients based on the AP1000-specific input exceed those that were calculated using AP600 parameters. Thus, the AP1000 LOCA dose analysis reported in the DCD, which used the aerosol removal coefficients based on AP600 parameters, is conservative. Once finalized, these aerosol removal coefficients could be used to support a change in the AP1000 reference site definition to relax the value for the Site Boundary atmospheric dispersion factor from the current value.

Design Control Document (DCD) Revision:

The following changes will be made in DCD Paragraph 15B.2.4.1:

> The increase in the AP1000 containment height **adversely affects the removal of aerosols by sedimentation. The greater containment height** also results in an increase in containment volume of 3.3E5 ft^3. The larger volume results in a reduction in airborne concentration which, when taken alone, ~~would~~ results in a reduction in the aerosol removal coefficient.

The following changes will be made in DCD Paragraph 15B.2.6:

> As discussed in 15B.2.4, the greater containment **height and** volume for the AP1000 ~~results in a reduction in airborne concentration which, all things being equal, would~~ results in a reduction in the aerosol removal coefficients. However, this increase in volume is offset by a number of other differences identified in 15B.2.4:

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.011 (Response Revision 1)

Question:

Regulatory Guide (RG) 8.12 (regarding criticality monitors) has been withdrawn. References to this RG were deleted in Section 1.9.1.4 of the DCD but not in Section 11.5.6. Section 11.5.6 also references 10 CFR 70.24. Since issuance of the AP600 SE, there has been new guidance on criticality monitors (namely 10 CFR 50.68).

Please provide the applicable guidance that the AP1000 design is using for criticality monitors (i.e. are you still complying with 70.24 or 50.68?).

Westinghouse Response:

The references to RG 8.12 will be deleted as shown in the attached DCD pages.

The AP1000 is designed for compliance with 10 CFR 70.24. Meeting 10 CFR 70.24 is cited in 10 CFR 50.68 paragraph (a) as an acceptable method for compliance with 10 CFR 50.68.

Design Control Document (DCD) Revision:

Delete references to Regulatory Guide 8.12 from Section 11.5 as shown on the attached pages.

PRA Revision:

None

NRC Additional Comments:

Design control document (DCD) Section 12.1.3, references RG 8.3, which has been withdrawn.

RG 8.13 (Instruction Concerning Prenatal Radiation Exposure) is referenced in the 1981 version of the SRP and should have been referenced in the COL Action Item in Section 12.1.3 of the DCD along with the other RGs.

Westinghouse Additional Response:

The reference to RG 8.3 will be deleted from DCD Section 12.1.3 as shown in the attached DCD page. RG 8.13 has been added to the list on the same page.

11.5.4 Process and Airborne Monitoring and Sampling

Radiation monitors are used to initiate automatic closure of isolation valves and dampers in liquid and gaseous process systems as described in subsection 11.5.2.3. These radiation monitors address the requirement of General Design Criterion 60 to suitably control the release of radioactive materials in gaseous and liquid effluents.

Radiation monitors are used in the radioactive waste processing systems as described in subsection 11.5.2.3. These radiation monitors address the requirement of General Design Criterion 63 to monitor radiation levels in radioactive waste systems.

Radiation monitors are used in the ventilation systems as described in subsection 11.5.2.3 to ensure that airborne concentrations within the plant are within the limits of 10 CFR 20.

11.5.5 Post-Accident Radiation Monitoring

The radiation monitors listed below meet the guidelines of Regulatory Guide 1.97 and are described in subsections 11.5.2.3 and 11.5.6.2. For further Regulatory Guide 1.97 information refer to Appendix 7A and Section 7.5.

- Main steam line radiation monitors
- Steam generator blowdown radiation monitor
- Main control room supply air duct radiation monitors
- Plant vent radiation monitor
- Turbine island vent discharge radiation monitor
- Containment high range radiation monitors
- Primary sampling room area monitor
- Technical support center area monitor

The post-accident sampling system is described in subsection 9.3.3 and is used to obtain samples for onsite laboratory analysis, including radioisotopic analysis, after a postulated accident.

11.5.6 Area Radiation Monitors

The area radiation monitors are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, ~~8.8, and 8.12.~~ and 8.8.

During refueling operations in containment and the fuel handling area, criticality monitoring functions, as stated in 10 CFR 70.24 ~~and Regulatory Guide 8.12~~, are performed by the area radiation monitors in combination with portable bridge monitors.

11.5.6.1 Design Objectives

The design objectives of the area radiation monitors during normal operating plant conditions and anticipated operational occurrences are to:

- Measure the radiation intensities in specific areas of AP1000
- Warn of uncontrolled or inadvertent movement of radioactive material in AP1000
- Provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where substantial changes in radiation flux might be of immediate importance to personnel
- Annunciate and warn of possible equipment malfunctions and leaks in specific areas of AP1000
- Furnish information for radiation surveys
- Minimize the time, effort, and radiation received by operating personnel during routine maintenance and calibration
- Incorporate modular design concepts throughout, to provide easy maintenance

By meeting the above objectives, the radiation monitoring system aids health physics personnel in keeping radiation exposures as-low-as-reasonably-achievable (ALARA).

Locations of area monitor detectors are based on the following criteria:

- Area monitors are located in areas that are normally accessible and where changes in normal plant operating conditions can cause significant increases in exposure rates above those expected for the areas.
- Area monitors are located in areas that are normally or occasionally accessible where significant increases in exposure rates might occur because of operational transients or maintenance activities.
- Area monitors are located to best measure the increase in exposure rates within a specific area and to avoid shielding of the detector by equipment or structural materials.
- In the selection of area monitors, consideration is given to the environmental conditions under which the monitor operates.

- Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.

12.1.3 Combined License Information

Operational considerations of ALARA, as well as operational policies and continued compliance with 10 CFR 20 and Regulatory Guides 1.8, 8.8, and 8.10, will be addressed by the Combined Operating License applicant. In addition, the Combined Operating License applicant will address operational considerations of the Standard Review Plan to the level of detail provided in Regulatory Guide 1.70. Regulatory Guides that will be addressed include: 8.2, ~~8.3,~~ 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 472.003 (Revision 1)

Original Question:

Section 9.4.1.2.1.1 indicates that radiation monitors are located inside the main control room upstream of the supply air isolation valves and that these monitors isolate the main control room form the nuclear island non-radioactive ventilation system on high-high particulate or iodine radioactivity concentrations. Does this include isolating the technical support center as well?

Original Westinghouse Response:

No, only the main control room is isolated on a high-high signal. At that time, themain control room emergency habitability system is placed into operation to protect the main control room operators. Please refer to the "Abnormal Plant Operation" portion of DCD subsection 9.4.1.2.3.1, which provides details as to the operation of the main control room and technical support center HVAC subsystem during abnormal events involving high and high-high signals.

Also see DCD subsection 18.8.3.5 "Technical Support Center Mission and Major Tasks"for discussions of the technical support center (TSC) including habitability and evacuation during emergencies.

NRC Additional Comments:

The staff has reviewed your response to RAI 472.003 dealing with technical support center (TSC) ventilation (i.e., habitability). The response referred to the Design Control Document (DCD) sections that covered TSC ventilation and habitability. While this answered the specific RAI question, it did not address apparent incorrect statements and inconsistencies in the system design, or the justification for relocation of TSC function to the emergency offsite facility (EOF) rather than to the main control room (MCR). Below are two questions pertaining to DCD Section 18.8.3.5, and an additional question pertaining to use of the EOF when the TSC becomes uninhabitable:

1. DCD Section 18.8.3.5 states that "Consistent with NUREG 0737 . . . the technical support center has no emergency habitability requirements." In accordance with NUREG-0737, the TSC is a "vital area" and should comply with radiological habitability requirements of General Design Criteria (GDC) 19 for the duration of an accident. Please provide justification for why the TSC has no emergency habitability requirements.

2. DCD Section 18.8.3.5 states (in italics) that "The TSC complies with the habitability requirements of Reference 27 [i.e., Supplement 1 to NUREG-0737] when electrical

power is available." First, Supplement 1 requires the same radiological habitability requirements as GDC 19, and thus, this statement contradicts (1), above; and second, the reference to "when electrical power is available" is but one, of two, triggering events that would automatically isolate the Main Control Room from the TSC. The second trigger is "High-high" particulate or iodine radioactivity in MCR air supply" (see DCD Section 6.4.4, page 6.4-9). Please provide justification for the inconsistencies.

3. In the event a relocation of the TSC to the EOF is allowed, rather than to the MCR (as required by NUREG-0696 guidance), how will the physical location of the EOF be addressed, as it relates to TSC support functions? There is currently a trend of utilities attempting to consolidate their EOFs for multiple plants. The physical location aspect is not addressed in the DCD, including whether the NRC would allow it. The implication is that the EOF could be anywhere, and as such, the transferred TSC functions could be anywhere.

Westinghouse Additional Response:

1& 2 The nuclear island nonradioactive ventilation system (VBS) maintains habitability in the TSC to the requirements of GDC 19 for normal and accident scenarios as long as electrical power is available and radiation levels do not exceed a predetermined ,"high-high" threshold. The VBS has two safety-related functions. The first is to monitor the air coming into the MCR and the second is to isolate the MCR envelope during a loss of electrical power of more than 10 minutes or upon a "high-high" radiation signal. As this system has no safety-related AC electrical system, it is not credited as meeting GDC 19 for the protection of the MCR operators. The safety-relatedmain control room emergency habitability system (VES) is credited as meeting GDC 19 for the protection of the MCR operators. Thus, Westinghouse agrees that the statement, "Consistent with NUREG-0737...the technical support center has no emergency habitability requirements.", is confusing. The statement will be removed from DCD section 18.8.3.5 in the next revision of the DCD. See the **Design Control Document (DCD) Revision:** section below for detail changes.

In the event of high radiation, the VBS operates in a recirculation mode filtering the air in the MCR and the TSC. In this mode, the VBS is designed to provide a capability similar to that of the engineered safety features (ESF) systems in operating plants with respect to air filtration and adsorption. Should a "high-high" radiation signal or if a station blackout of more than 10 minutes occur, the VBS stops, isolates the MCR envelop and the VES begins operation to protect the MCR operators. If the system has power and is operating, it will prevent a "high-high" radiation signal. This is the reason DCD 18.8.3.5 states, "The TSC complies with the habitability requirements of Reference 27 [i.e., Supplement 1 to NUREG-0737] when electrical power is available."

In practical terms, the TSC does have emergency habitability capabilities comparable to those of operating plants as long as electrical power is available either from offsite power or from the onsite diesel generators. See the response to item 3 below, for a discussion on the probability of losing both offsite power and the onsite diesel generators.

3. The AP1000 design philosophy for the MCR and TSC habitability is the same as for the AP600. Discussions of this design were provided in **AP600** RAIs 100.10 and 100.33. In a very limited number of instances, the TSC may become uninhabitable. As stated in the DCD 18.8.3.5, even in the low probability case of a station blackout, the TSC will still most likely remain habitable. The doors to the TSC can be opened to aid with ventilation and control of room temperature for the two hours that the workstations continue to operate. The TSC workstations are powered from the non-Class 1E uninterruptable power supplies, therefore plant monitoring capability from the TSC exists for two hours following a station blackout. (The probability of a station blackout is discussed in the AP1000 Probability Risk Assessment. The probability of a station blackout occurring is 8.57×10^{-4}. The probability of non-recovery within 2 hours is specified in the EPRI ALWR Utility Document as 0.37.)

To assure that the functions of the TSC are notimpeded, Westinghouse states in DCD 13.3 that staffing of the EOF for the AP1000 will occur consistent with current operating practice and revision 1 of NUREG-0654/FEMA-REP-1. In the unlikely event of a loss of offsite power and loss of all onsite AC power, the Combined License applicant shall immediately activate the EOF rather bringing it to standby status. As stated in DCD 18.8.3.5 a communicator is assigned to the MCR as part of the emergency staffing. The communicator is responsible for providing direct interface between the TSC and the MCR operators. If the TSC function has been transferred to the EOF, then the communicator provides the direct interface between the EOF and the MCR operators. The Combined License applicant is responsible for the EOF design, including the specification of its location(s) (DCD subsection 18.2.6), emergency planning, and associated communication interfaces among the MCR, the TSC, and the EOF (DCD subsection 13.3). Westinghouse has committed to providing a TSC communicator in the MCR for the unlikely event that the TSC becomes uninhabitable. When the Combined License applicant establishes the emergency plan, and associated communication interfaces among the MCR, the TSC, and the EOF; the NRC will have an opportunity to review the plan including the total number of TSC support personnel that will be sent to the MCR in the event that the TSC becomes uninhabitable as well as the location of the EOF.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision: (Revision 1)

Revise the fourth paragraph of DCD 18.8.3 as indicated below:

Consistent with NUREG 0737, the ~~following criteria are established for the technical support center:~~
~~The~~ technical support center is nonsafety-related and is not required to be available after a safe shutdown earthquake.

- ~~The technical support center has no emergency habitability requirements.~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 480.010

Question:

DCD Tier 2, Chapter 16, contains proposed TS for AP1000 plants. In section 5.5, "Programs and Manuals," is TS 5.5.8, "Containment Leakage Rate Testing Program." It includes this passage:

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is less than the design pressure of containment.

In contrast, the WOG Standard TS states:

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig].

Option B of Appendix J to 10 CFR Part 50 requires the numerical value of P_a to be specified in the TS.* The AP600 proposed TS did so, correctly, but this has been changed for AP1000, without explanation. Please Provide justification for why the AP1000 DCD does not comply with the regulations.

* App. J allows any plant to choose to conform to either Option A of App. J (Prescriptive Requirements), Option B (Performance-Based Requirements), or a specific combination of Options A and B. Their TS must specify which choice they have made. The WOG STS contains three versions of this TS, to account for these possibilities. Two of the versions (Option B, and Options A and B combined) specify the value of P_a, but the Option A version does not. This is because Option A does not require it; Option B does. The AP1000 DCD allows COL applicants to choose which option of App. J they want, but it is highly unlikely that an applicant will choose Option A alone. All operating plants today have chosen either Option B or a combination of Options A and B, because of the millions of dollars saved by using Option B. Also, the AP1000 proposed TS follow the Option B model.

Westinghouse Response:

TS 5.5.8.b is being revised to reference Bases 6.2.4, which identifies the containment design pressure. Bases 6.2.4 will be revised to add the value for the calculated peak containment internal pressure for the limiting design basis accident. Since there are no Bases for TS Section 5.0, it is appropriate to reference the value in Bases 6.2.4.

As discussed in the response to RAI 630.042, the AP1000 TS Bases 3.6.4, 3.6.5, and 3.6.6 had previously been written so as to include the calculated peak containment internal pressure analyses result value via reference. This method provided the benefit of TS simplification during plant life as no update to the TS Bases is required solely for the purpose of maintaining the Bases in the event that re-analyses are performed, but the TS remained bounded by the limits identified in the Bases. Thus, the AP1000 DCD 16.1 TS Bases 3.6.4, 3.6.5, and 3.6.6 appeared somewhat different than their counterparts in AP600, but contained the same information and reduced future changes to the AP1000 DCD.

The limiting calculated peak containment internal pressure, Pa, from the design basis safety analyses results for SLB and LOCA and the containment design pressure referenced by these TS Bases can be found in AP1000 DCD Subsection 6.2.1. Table 6.2.1.1-1 identifies the calculated peak containment internal pressure for the four design basis events analyzed, and the table also specifies the containment design pressure. The containment design pressure is also specified in the discussion on Applicable Safety Analyses in the Bases for TS 3.6.4 since the design containment pressure does not change with re-analysis.

The intent of referencing the design basis analyses in Section 6.2.1 in the Bases of these TSs, rather than identifying the specific safety analysis results, is consistent with the overall TS improvement strategy to minimize the need for a plant license amendment or Bases update for parameters that are expected to change due to re-analysis. For example, the plant COLR and PTLR were implemented to specifically identify plant parameters that are routinely expected to vary from fuel cycle to fuel cycle, such as shutdown margin, MTC, rod insertion limits, core peaking factors, and RCS pressure and temperature limits. The values for these parameters are specifically not included in Technical Specifications so that license amendments and Bases updates are not required every fuel cycle.

Therefore, Technical Specifications 3.6.4, 3.6.5, 3.6.6, and 5.5.8.b were written following this same strategy, to eliminate the need for revising the Technical Specification in the future where there may be a change to the calculated peak containment internal pressure if containment re-analysis is performed. This precludes the need to require a license amendment or a Bases revision simply to update to the limiting calculated peak containment internal pressure value resulting from the containment re-analysis.

It is not clear that Appendix J to 10 CFR 50 specifically requires that the numerical value for Pa be included in the Technical Specifications or Bases. The only discussions in Appendix J related to Pa and Technical Specifications are in two sentences in Appendix J that appear to be somewhat inconsistent between each other. The section of Appendix J that is applicable to AP1000, TS 5.5.8.b, needs to be revised to consider the limiting AP1000 safety analyses event.

In the discussion of Option A in Appendix J (which is not applicable to AP1000), the only reference to Pa is found not in a requirement for the containment leakage testing program, but in the Explanation of Terms, Section II.I of Appendix J. The intent appears to be to explain the definition of the term Pa and to indicate where the specification(s) for this Pa can be found, which is stated as "...either in the technical specification or the associated bases." It is not clear from the statement in Option A that a numerical value is required.

In the discussion for Option B (which is applicable to AP1000), the only reference to Pa is again found not in a requirement for the containment leakage testing program, but in the Definitions, Section II of Appendix J. The intent again appears to be to provide the definition of the term Pa and to indicate where the specification(s) for this parameter can be found. The location of the Pa is stated differently, as "...in Technical Specifications." This is assumed by Westinghouse to reference to the entire Technical Specifications document, which includes the individual technical specifications and the associated bases. So although the Option B discussion is slightly different from the discussion in Option A, it is assumed to reference both sections of the TS document.

However, the definition of Pa for this section is incorrect for AP1000. In Section II of Option B, it states that "Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design loss-of-coolant accident..." For the AP1000, this definition is not correct since the limiting calculated peak containment internal pressure in DCD 6.2 occurs for a steamline break accident. However, the steamline break pressure is only slightly larger than the resulting pressure following a loss-of-coolant accident, so it is possible that the limiting event could change for future containment analyses. It is also not clear from the statement in Option B that a numerical value is required.

Based on Option B, TS 5.5.8.b currently specifies that Pa is less than the containment design pressure, but it does not identify the numerical value to be consistent with the overall TS improvement strategy. A sentence will be added to reference the Bases for TS 3.6.4. TS 3.6.4 Bases currently reference the analyses that calculate Pa in DCD Section 6.2.1 where the value of Pa is identified in Table 6.2.1.1-1. This was originally considered sufficient to satisfy the intent of Appendix J, both Option A, Section II.I, Explanation of Terms and Option B, Section II, Definitions.

However, to prevent lengthy negotiations over the interpretation of statements in Appendix J concerning Pa, the Bases for TS 3.6.4 will be revised to include the actual value for the limiting calculated peak containment internal pressure.

This change is considered to be inconsistent with the overall intent of the TS improvement program efforts to minimize the need for unnecessary license amendments or Bases updates for information that is more appropriately contained in the DCD and is currently appropriately referenced from the AP1000 TS Bases.

Design Control Document (DCD) Revision:

DCD TS 5.5.8.b will be revised as follows:

b. The ~~peak~~ calculated **peak** containment internal pressure for the design basis ~~loss-of-coolant~~ accident, Pa, is less than the design pressure of containment. **The Applicable Safety Analyses section of the Bases for TS 3.6.4 identifies the peak containment internal pressure for the design basis event and the design containment pressure.**

The second paragraph of DCD TS 3.6.4 Applicable Safety Analyses will be revised as follows:

The initial pressure containment used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a DBA as indicated in Reference 1. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting DBA. The maximum containment pressure resulting from the worst case DBA **of [57.3] psig** does not exceed the containment design pressure, 59 psig.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 480.011

Question:

DCD Tier 2 Section 6.2.5, "Containment Leak Rate Test System," has a bad reference. The section cites Appendix J to 10 CFR Part 50, as it should, but refers the reader to Reference 14, which reads as follows:

14.10 CFR 50, Appendix J (Draft Proposed Revision), "Containment Leak Rate Testing," January 10, 1992.

The Draft Proposed Revision should not be referenced in the DCD for the following reasons:

a. The cited document is not generally available to the public. It was not published in the Federal Register. It was not finalized. Even the NRC would have difficulty finding this document in its files.

b. This document predates the effort to revise App. J that culminated in the issuance of Option B of the regulation in 1995. Thus, the cited document would have none of the Option B information which was clearly used by Westinghouse to prepare the DCD.

c. App. J has not changed since 1995, so there is no reason to cite anything other than the current regulation.

It seems clear by reading the DCD that the AP1000 design uses the current regulation. This appears to simply be a left-over, out-of-date reference. Please remove the reference to the "Draft Proposed Revision" or provide the basis for using the draft proposed revision.

Westinghouse Response:

DCD 6.2.7 will be updated to reference the current regulation.

Design Control Document (DCD) Revision:

Item 14 of DCD Section 6.2.7 will be revised as follows:

14. 10 CFR 50, Appendix J ~~(Draft Proposed Revision)~~, "Containment Leak Rate Testing," **September 26, 1995**~~January 10, 1992~~.

PRA Revision:

None