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Your ref: Docket No. 52-006
Our ref: DCP/NRC1715

August 11, 2004

SUBJECT: Transmittal of Proposed Revisions to AP1000 DCD Sections 13.6.10, 5.3.6.3, 3.9.1.1.1.18, 5.3.2.6, and 5.3.3.1

This letter transmits proposed revisions to the Westinghouse AP1000 Design Control Document (DCD) to resolve the comments communicated by the U.S. Nuclear Regulatory Commission during the August 9, 2004 telecon. Specifically, the revisions pertain to AP1000 DCD Sections 13.6.10, 5.3.6.3, 3.9.1.1.1.18, 5.3.2.6, and 5.3.3.1. The revision mark-up is non-proprietary and is transmitted as Attachment 1.

Westinghouse will include these changes in the AP1000 DCD, Revision 14.

Please contact me at 412-374-4728 if you have any questions concerning this submittal.

Very truly yours,

for R. P. Vijuk, Manager
Passive Plant Engineering
AP600 & AP1000 Projects

/Attachments

1. "NRC Comment Item Response," Sections 13.6.10, 5.3.6.3, 3.9.1.1.1.18, 5.3.2.6, and 5.3.3.1, dated 8/11/04.

D063

DCP/NRC1715
Docket No. 52-006

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Attachment 1

AP1000 Design Certification Review
NRC Comment Item Response

DCD Section 13.6.10

DCD Section 5.3.6.3

DCD Section 3.9.1.1.1.18

DCD Section 5.3.2.6

DCD Section 5.3.3.1

AP1000 DESIGN CERTIFICATION REVIEW

NRC Comment Item Response

NRC Comment: **from August 9, 2004 telecon**

Original RAI Number(s): **NA**

Summary of Issue:

In this telecom, the NRC identified two items:

1. In DCD section 13.6.10 on security communications, change the word "fed" to "powered".
2. In DCD section 5.3.6.3 on reactor vessel material property verification, change phrase "60-year period of COL".

Westinghouse Response:

1. Westinghouse agrees to make this change. See attached DCD markup.
2. Westinghouse proposed to change this phrase to "plant design objective of 60-years". We would also like to change phrases in DCD sections 3.9.1.1.1.18, 5.3.2.6, and 5.3.3.1 to use similar phrases ("design objective" instead of "design life"). The proposed DCD wording changes are also attached.

Design Control Document (DCD) Revision:

Revise DCD 13.6.10 as follows:

13.6.9 Security Power Supply System

Security equipment that supports critical monitoring functions, such as intrusion detection, alarm assessment, and the security communication system, can receive power from the security-dedicated uninterruptible power supply (UPS) system. Switchover to the uninterruptible power supply system is automatic and does not cause false alarms on annunciation modules. The uninterruptible power supply system is capable of sustaining operation for a minimum of 24 hours. The location of the security power supply system is specified in Reference 6. The final design of the security power supply system is the responsibility of the Combined License applicant.

13.6.10 Communications

The final design of the security communication system will be addressed by the Combined License applicant.

Two two-way communications paths are provided between the control room and the alarm stations within the AP1000. A single act of sabotage cannot sever both communication paths. Security force members with responsibilities to respond to acts of sabotage have the capability for continuous two-way communication with the alarm stations, and with each other. The centralized communication equipment is located in a vital area so that it will remain operable during a radiological sabotage event.

Non-portable security communications equipment can be powered ~~fed~~ from the security power supply system so that it remains operable in the event of the loss of normal power.

13.6.11 Testing and Maintenance

The Combined License applicant will address testing and maintenance aspects of the plant security system.

Revise DCD 5.6.3.4 as follows:

5.3.6.3 Surveillance Capsule Lead Factor and Azimuthal Location Confirmation

The Combined License Applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

5.3.6.4 Reactor Vessel Materials Properties Verification

The Combined License applicant will address verification of plant-specific belt line material properties consistent with the requirements in subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification will include a pressurized thermal shock evaluation based on as-procured reactor vessel material data and the projected neutron fluences for the plant design objective of 60-years ~~period of the COL~~. This evaluation report will be submitted for NRC staff review.

The verification will include structural analysis of the AP1000 reactor vessel insulation and support structure.

5.3.6.5 Reactor Vessel Insulation

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention. The ULPU Configuration V test data is suitable to be used to develop the design loads for the AP1000 reactor vessel insulation design.

Revise DCD 3.9.1.1.1.18 as follows:

3.9.1.1.1.17 Passive Residual Heat Removal Test

During hot functional testing with the reactor coolant system in hot standby condition, the passive residual heat removal flow and heat transfer rates are tested. Passive residual heat removal flow is initiated by opening the passive residual heat removal isolation valves. The passive residual heat removal cools the reactor coolant system for up to 30 minutes.

3.9.1.1.1.18 Reactor Coolant System Makeup

The chemical and volume control system makeup subsystem is used to accommodate normal minor leakage from the reactor coolant system. On a low programmed pressurizer level signal one of the chemical and volume control system makeup pumps starts automatically in order to provide makeup. The pump automatically stops when the pressurizer level increases to the high programmed setpoint. The addition of the makeup water to the reactor coolant system via the chemical and volume control system purification loop and attendant changes in reactor coolant system parameters constitute the reactor coolant system makeup design transient. The total number of occurrences of the makeup transient is 2820, which corresponds to once per week during the ~~60-year~~ plant design ~~life~~ objective of 60 years assuming a 90 percent availability factor for the plant.

3.9.1.1.2 Level B Service Conditions (Upset Conditions)

The following paragraphs describes the reactor coolant system upset condition transients, which are considered to be plant condition PC-2 and PC-3 per ANS N51.1. From the standpoint of the use of design transient in the evaluation of cyclic fatigue, there is no difference between PC-2 and PC-3. These transients are analyzed using Level B service limits and are as follows:

Revise DCD 5.3.2.6 as follows:

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in subsection 5.3.2.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on capsules withdrawn. The recommended program schedule for removal of the capsules for post-irradiation testing includes five capsules to be withdrawn instead of four as specified in ASTM E-185 (Reference 1) and Appendix H of 10 CFR 50. The following is the recommended withdrawal schedule of capsules for AP1000.

Capsule	Withdrawal Time
1st	When the accumulated neutron fluence of the capsule is 5 x 10^{18} n/cm^2.
2nd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel 1/4T location.
3rd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel inner wall location.
4th	When the accumulated neutron fluence of the capsule corresponds to a fluence not less than once or greater than twice the peak end of vessel life fluence.
5th	End of plant design objective~~life~~ of ~~(~~60 years~~)~~
6th	Standby
7th	Standby
8th	Standby

5.3.2.6.1 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate estimate of the average neutron flux level, and hence, time integrated exposure (fluence) experienced by the sensors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

Revise DCD 5.3.3.1 as follows:

Predicted ΔRT_{NDT} values are derived considering the effect of fluence and copper and nickel content for the reactor vessel steels exposed to 550°F temperature. U.S. NRC Regulatory Guide 1.99 is used in calculating adjusted reference temperature. Since the AP1000 cold leg temperature exceeds 525°F (minimum steady-state temperature is 535°F at 100% power, thermal design flow, and 10% tube plugging), the procedures of Regulatory Guide 1.99 for nominal embrittlement apply. The heatup and cooldown curves are developed considering a sufficient magnitude of radiation embrittlement so that no unirradiated ferritic materials in other components of the reactor coolant system will be limiting in the analysis.

The pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (EFPY) consistent with the ~~an expected~~ plant design objective~~life~~ of 60 years with 90 percent availability. Copper, nickel contents and initial RT_{NDT} for materials in the reactor vessel beltline region and the reactor vessel flange and the closure head flange region are shown in Tables 5.3-1 and 5.3-3. The operating curves are developed with the methodology given in Reference 6, which is in accordance with 10 CFR 50, Appendix G with the following exceptions:

1. The fluence values used are calculated fluence values (i.e., comply with Regulatory Guide 1.190), not the best-estimate fluence values.