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Your ref: Docket No. 52-006
Our ref: DCP/NRC1563

April 2, 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
AP600 & AP1000 Projects

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1560"

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

D063

			Attachment
bcc:	C. B. Brinkman	- Westinghouse, Rockville, MD	1
	M. M. Corletti	- Westinghouse, Pittsburgh, PA	2
	Doc Control	- US NRC, Rockville, MD	Original
	W. E. Cummins*	- Westinghouse, Pittsburgh, PA, EC E3	
	J. Segala	- US NRC, Rockville	10
	H. A. Sepp*	- Westinghouse, Pittsburgh, PA, EC E4-07A	
	R. P. Vijuk*	- Westinghouse, Pittsburgh, PA, EC E3-05	
	J. W. Winters*	- Westinghouse, Pittsburgh, PA, EC E3-08	

* Letter only

DCP/NRC1563

April 1, 2003

Attachment 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1563"

Attachment 1

Table 1 "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1563"
210.050, Rev. 1 440.119, Rev. 1 440.188, Rev. 0

Attachment 2

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

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.050 (Revision 1 Response)

Question:

Section 3.9.3.1.2: The discussion on the identification and evaluation of the pressurizer surge line susceptible to thermal stratification (Bulletin 88-11) is identical to the AP600. Did Westinghouse consider the differences between the AP1000 and the AP600 with regard to the potential for stratification between the pressurizer and the hot leg? Specifically, it is not well known that the pressurizer could be stratified and **the heat-up and cool-down rate** could exceed the defined limit with large surge flow rate. Please describe in the DCD the control of the heat-up and cool-down procedure such that the ΔT between the pressurizer and the reactor coolant system (RCS) hot leg will be less than acceptable value(s) and pressurizer stratification will not be a concern from the stress and fatigue points of view.

Westinghouse Revision 0 Response:

The design of the AP1000 surge line is identical to the AP600 surge line. Therefore, the discussion and evaluation presented in Section 3.9.3.1.2 is the same for the AP600 and AP1000. Section 3.9.3.1.2 provides a detailed conformance assessment of the AP1000 design to NRC Bulletin 88-11.

The AP1000 design will have a slightly lower susceptibility to surge line stratification during normal operation than the AP600 design due to the increased AP1000 operating temperature. Specifically, surge line stratification can develop due to the temperature difference between the pressurizer and the hot leg. In the AP600, the hot leg temperature ranges between 545 and 600 F, while the AP1000 hot leg temperature ranges between 557 and 610 F. The pressurizer operating temperature is 653 F for both plants. Therefore the normal operating ΔT for the AP600 approximately 53 F, while the normal operating ΔT for the AP1000 surge line is 43 F. The surge line is designed to accommodate a temperature difference of 320 F, which can occur during shutdown operations. This limitation is identified in Appendix E of Chapter 19 in the AP1000 DCD (subsection 19E.3.1.3.4.).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Follow-On Comments:

Address the specific staff concern that stresses in the pressurizer may exceed fatigue stress or heatup/cooldown rate limits. Provide additional information regarding the method of heat-up and cool-down and the procedure to control the compliance to limits in order to ensure that pressurizer stratification and heatup/cooldown will not be issues of concern.

Westinghouse Response to NRC Follow-On Comments:

A Westinghouse Owners Group program evaluated the presence of insurge/outsurge transients in operating plants that could result in stratification of the pressurizer. This program recommended modified plant operational procedures to protect the pressurizer lower head from these transients during heatup and cooldown operations when the temperature differences between the pressurizer and hot leg are at their maximum values.

The discussion of system operation provided in the AP1000 DCD section 5.4.5.2.3 is consistent with the recommended operational procedures from the Westinghouse Owners Group program. The following provides a discussion of these procedures in more detail. Note that operational procedures are the responsibility of the Combined License applicant as discussed in DCD section 13.5.1.

Continuous pressurizer outsurge flow is maintained during plant heatup and cooldown by energizing all of the backup pressurizer heater groups. Operation of the heaters results in continuous spray flow and consequently, continuous outsurge flow. A reactor coolant pump in one of the two cold legs providing pressurizer spray flow is operated to provide the required spray flow. This strategy of maintaining continuous pressurizer outsurge flow has been implemented at several operating plants and has been shown to reduce the number of insurge transients on the pressurizer lower head.

The maximum temperature difference between the pressurizer and hot leg in the AP1000 is reduced compared to operating plants because of the use of the canned motor reactor coolant pumps. The minimum pressure in the reactor coolant system in current plants during heatup and cooldown is dependent upon the #1 seal differential pressure requirement for the shaft seal reactor coolant pump. Since there are no shaft seals in the AP1000 canned motor reactor coolant pumps, this requirement is eliminated and the pumps can operate at a lower reactor coolant system pressure. This reduces the maximum system temperature difference, which occurs during the pressurizer pressure/saturation temperature plateau during heatup and cooldown.

Although pressurizer insurge and outsurge transients can be largely avoided by implementing a continuous pressurizer outflow operating mode, the Owner's Group program also recommended that provisions be made in the design for inadvertent or potentially unavoidable transient events that may occur. These transients are included in the fatigue evaluation and cover insurges into the pressurizer that result in pressurizer temperature changes exceeding the overall saturation temperature change transients of 100 F/ hour heatup and 200 F / hour cooldown. Design transients specifically for the lower pressurizer head and shell region derived from current plant monitoring programs are included in the AP1000 pressurizer equipment specification. The transients are defined to envelope a fast moving stratified interface of hot/cold water on the surge line nozzle, lower pressurizer head, and lower portion of the shell region.

Pressurizer stratification and heatup/cooldown limits are addressed in the AP1000 through the following design and operating provisions:

- A continuous pressurizer outsurge is maintained during plant heatup and cooldown.
- The temperature difference between the pressurizer and hot leg is minimized to the extent possible.
- Specific design transients for the lower pressurizer head and lower shell region are included in the design analyses to show design acceptability for insurge/outsurge events that result in stratification in the pressurizer.

A description of AP1000 operations to minimize pressurizer stratification during plant heatup and cooldown will be included in DCD section 5.4.5.2.3.

Design Control Document (DCD) Revision:

From DCD Revision 3 page 5.4-30:

5.4.5.2.3 Operation

During steady-state operation at 100 percent power, approximately 50 percent of the pressurizer volume is water and 50 percent is steam. Electric immersion heaters in the bottom of the vessel keep the water at saturation temperature. The heaters also maintain a constant operating pressure.

A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the pressurizer liquid and the reactor coolant. This continuous flow also prevents excessive cooling of the spray piping. Proportional heaters in the control group are continuously on during normal operation to compensate for the continuous introduction of cooler spray water and for losses to ambient.

These conditions result in a continuous out-surge in most cases during normal operation and anticipated transients. The out-surge minimizes the potential for thermal stratification in the surge line.

During an out-surge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the low-pressure engineered safety features actuation setpoint. During an in-surge from the reactor coolant system, the spray system (which is fed from two cold legs) condenses steam in the pressurizer. This prevents the pressurizer pressure from reaching the high-pressure reactor trip setpoint. The heaters are energized on high water level during insurge to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

During heatup and cooldown of the plant, **when the potential for thermal stratification in the pressurizer is the greatest,** the pressurizer may be operated with a continuous ~~spray and a portion of the heaters energized to result in an~~ out surge of water from the pressurizer. **This is achieved by continuous maximum spray flow and energizing all of the backup pressurizer heater groups. The temperature difference between the pressurizer and hot leg is minimized by maintaining the lowest reactor coolant system pressure possible consistent with operation of a canned motor reactor coolant pump.** This mode of operation minimizes the **frequency and magnitude of** thermal shock to the surge **line** nozzle **and lower pressurizer head,** and the potential for stratification in the **pressurizer and** surge line. **The design analyses of the pressurizer include consideration of transients on the lower head and shell regions to account for these possible insurge/outsurge events.**

The pressurizer is the initial source of water to keep the reactor coolant system full of water in the event of a small loss of coolant. Pressurizer level and pressure measurements indicate if other sources of water, including the chemical volume and control system and passive safety systems, must be used to supply additional reactor coolant.

Power to the pressurizer heaters is blocked when the core makeup tanks are actuated. This action reduces the potential for steam generator overfill for a steam generator tube rupture accident.

PRA Revision:

None

RAI Number: 440.119 (Response Revision 1)

Question:

It indicates on pages 19E-34 and-35 that Reference 10 ("AP600 Shutdown Evaluation Report) of Apendix19E documents analyses of LOCA events and loss of RNR events at lower modes for AP600. It also indicates that Reference 10 for AP600 is applicable to AP1000 because (1) accident analyses presented in Chapter 15 demonstrated that the AP1000 plant response to accidents is similar to the AP600 plant response, and (2) availability of the passive core cooling system components in lower modes is the same for both the AP600 and AP1000.

Discuss a comparison of applicable Chapter 15 analyses to demonstrate that the AP1000 plant response to accidents is similar to the AP600 plant response. In lower modes, the AP1000 plant response to accidents may be different from the AP600 plant response. Explain why the use of Chapter 15 accident analyses (which are performed for Modes 1 and 2 conditions) is acceptable for justifying the applicability of the cited reference to the AP1000 plant at lower modes.

Westinghouse Original Response

The original response provided an update to DCD Section 19E to include the AP1000-specific analysis of a loss of RNS cooling in shutdown modes. The analyses were included in DCD Chapter 19 Appendix E Revision 3.

NRC Additional Comment:

The analysis of a loss of RNS in Mode 4 provided in DCD Appendix 19E is performed assuming both CMT are available. The statement is made in the DCD that the plant response is similar if one CMT is available. Please justify this statement.

Westinghouse Response Revision 1

In the analysis of the loss of RNS in Mode 4 with the RCS intact, both CMTs were assumed to operate, and the analysis results provided in the DCD Appendix 19E demonstrate the adequacy of the AP1000 passive safety systems to protect the plant in shutdown modes. To confirm the statement made in the DCD Appendix 19E that the plant response is similar for the loss of RNS in Mode 4 considering only one CMT available, an additional analysis has been performed. The attached figures show a comparison of the plant response to a loss of RNS cooling in Mode 4, with only one CMT available. Results show similar plant response for either case, and further demonstrate the adequacy of the passive safety systems to protect the plant in shutdown modes.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

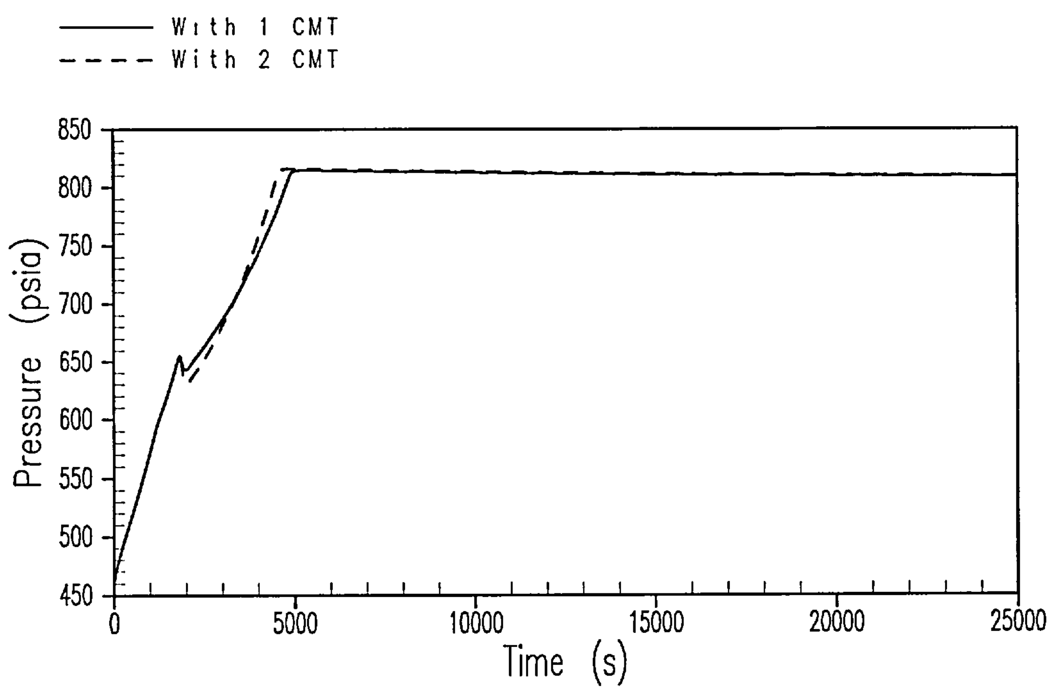

Figure 440.119-1 Pressurizer Pressure, Loss of RNS in Mode 4 RCS Intact
Comparison of 1 CMT vs 2 CMT Operating

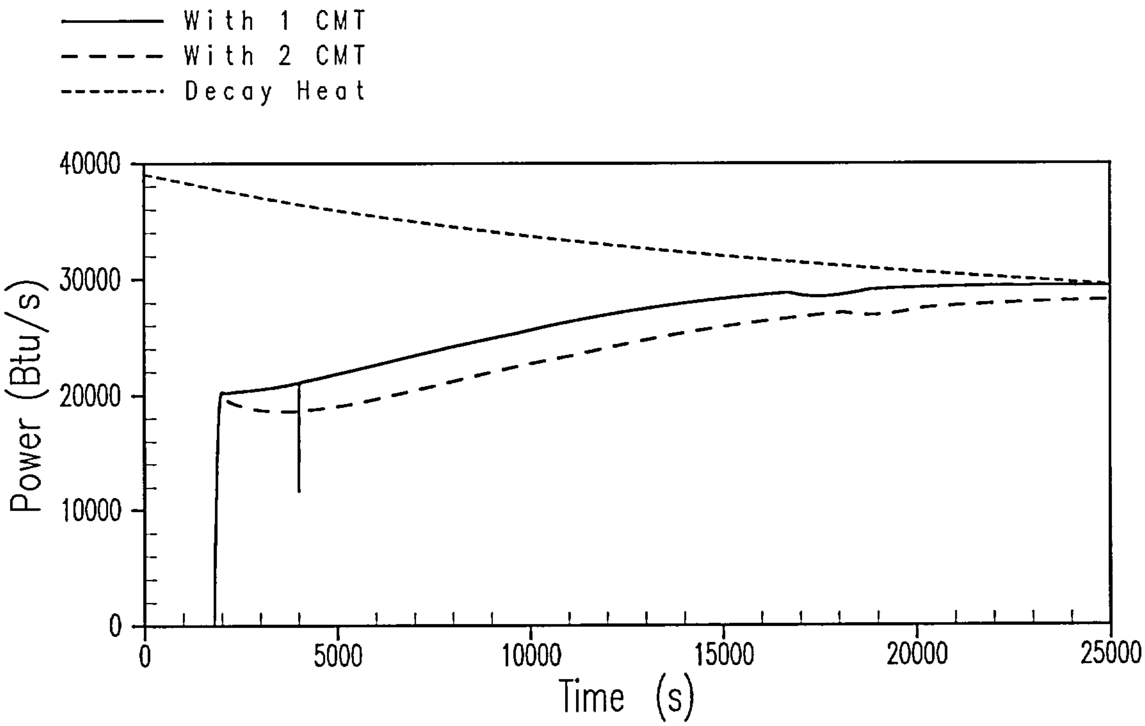

Figure 440.119-2 Decay Heat and PRHR Heat Removal, Loss of RNS in Mode 4 RCS Intact Comparison of 1 CMT vs 2 CMT Operating

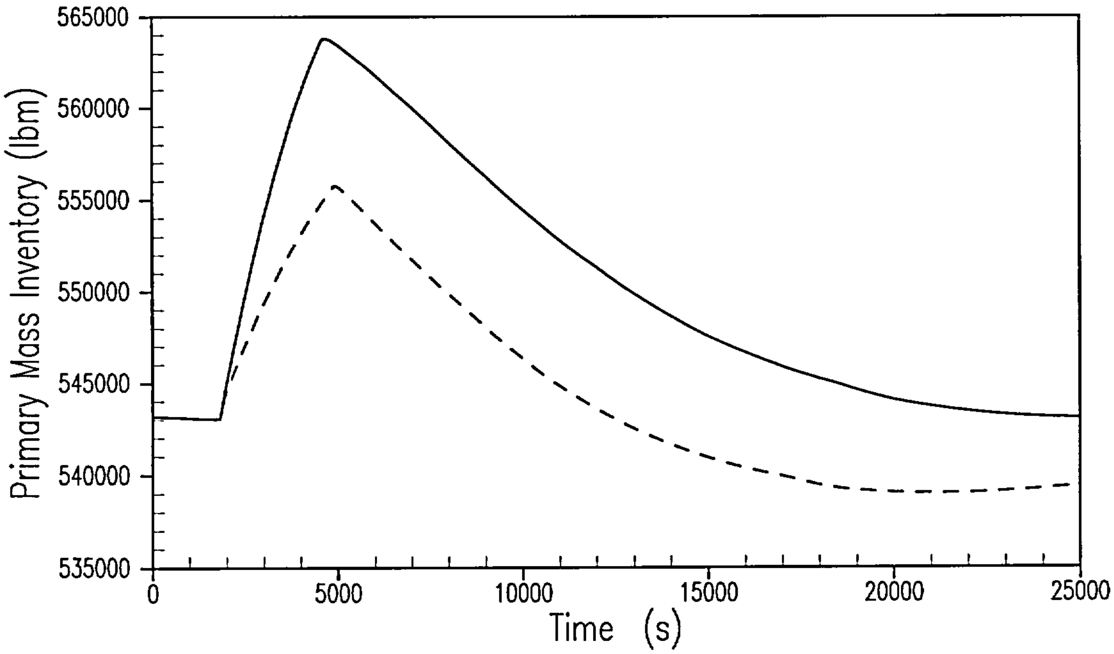

Figure 440.119-3 Primary System Inventory, Loss of RNS in Mode 4 RCS Intact Comparison of 1 CMT vs 2 CMT Operating

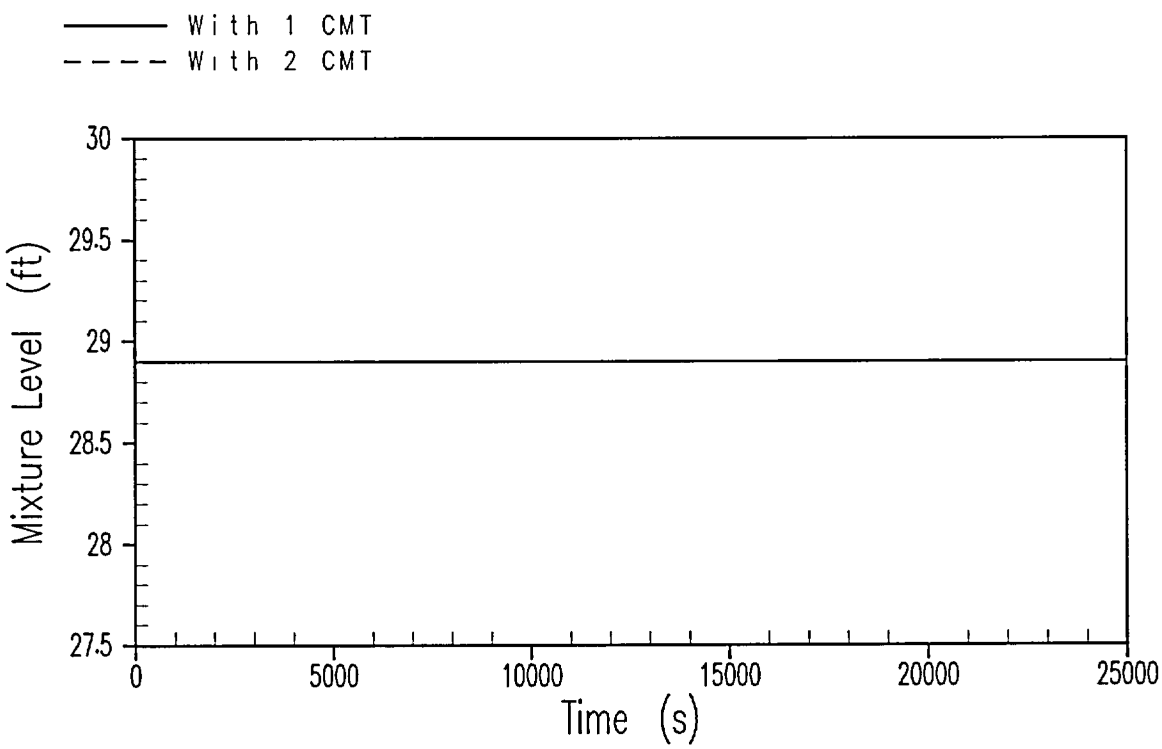

Figure 440.119-4 Core Mixture Level, Loss of RNS in Mode 4 RCS Intact
Comparison of 1 CMT vs 2 CMT Operating
(Top of core is 20.5 ft)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.188

Question:

a. Generic Letter 93-04 identified a potential problem with rod control system failure and inadvertent withdrawal of a single rod control cluster assembly. WCAP-13854, Revision 1, "Ron Control System Evaluation Program," provided Westinghouse Owners Group's response to the issues raised in GL 93-04. WCAP-15800, Revision 1, "Operational Assessment for AP1000," indicates that the resolution of GL 93-04 for the AP1000 design is addressed in the AP1000 DCD Section 3.9.4. The staff reviewed Section 3.9.4 and could not find the description of the resolution of GL 93-04.

b. Please identify where in DCD Section 3.9.4 that the GL 93-04 issue is addressed, and/or describe how the issue is resolved for the AP1000 design, including the design and surveillance tests of the AP1000 rod control system which are consistent with WCAP-13854 and acceptable for the resolution of GL 93-04.

Westinghouse Response:

The AP1000 control rod drive system is the same as the AP600 control rod drive system and is based on a proven Westinghouse design that is used in many operating nuclear power plants. The control rod drive system is described in DCD section 3.9.4. The rod control system is described in DCD section 7.7.1.2.

The AP1000 rod control system incorporates design improvements developed by Westinghouse in response to operating experience, including the current order timing modification described in WCAP-13864, Revision 1-A, "Rod Control System Evaluation Program." The current order timing modification ensures that, if failures similar to those that occurred at Salem are present, the control rods insert symmetrically.

Based on its review of WCAP-13864, Revision 1, the NRC concluded that the current order timing modification and additional surveillance tests at the beginning of each cycle is an acceptable resolution to the concerns raised in Generic Letter 93-04.

The rod control system is extensively tested during preoperational and startup testing. See DCD Sections 14.2.9.1.8 and 14.2.10.1.11 for additional details. Additional testing to be performed during the operational phase of the plant is the responsibility of the Combined License applicant, as stated in DCD Section 13.5.

Design Control Document (DCD) Revision:

None

PRA Revision:

None