52-003

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 22, 1996

Mr. Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities
Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

SUBJECT: FOLLOWON QUESTIONS CONCERNING THE AP600 PROBABLISTIC RISK ASSESSMENT (PRA)

Dear Mr. Liparulo:

As a result of its review of the June 1992, application for design certification of the AP600, the staff has determined that it needs additional information in order to complete its review. Specifically, the enclosed questions are related to Westinghouse's responses documented in Letter NSD-NRC-96-4662 (March 8, 1996), related to several Draft Safety Evaluation Report Open Items.

You have requested that portions of the information submitted in the June 1992, application for design certification be exempt from mandatory public disclosure. While the staff has not completed its review of your request in accordance with the requirements of 10 CFR 2.790, that portion of the submitted information is being withheld from public disclosure pending the staff's final determination. The staff concludes that these followon questions do not contain those portions of the information for which exemption is sought. However, the staff will withhold this letter from public disclosure for 30 calendar days from the date of this letter to allow Westinghouse the opportunity to verify the staff's conclusions. If, after that time, you do not request that all or portions of the information in the enclosures be withheld from public disclosure in accordance with 10 CFR 2.790, this letter will be placed in the Nuclear Regulatory Commission's Public Document Room.

These followon questions affect nine or fewer respondents, and therefore is not subject to review by the Office of Management and Budget under P.L. 96-511.

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If you have any questions regarding this matter, you can contact me at (301) 415-1132.

Sincerely,

original signed by:

Joseph M. Sebrosky, Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 52-003

Enclosure: As stated

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AP600

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AP600 PRA REVIEW
REQUEST FOR ADDITIONAL INFORMATION

720.327 Followon RAIs related to DSER Open Item 19.1.3.1-1. In a previous RAI Westinghouse was asked to evaluate the impact of several issues raised, by the staff on the estimated PRHR tube rupture frequency. The staff reviewed Westinghouse's response and identified the need for the following additional information:

a. It is stated that "The Technical Specifications will allow plant operation with a small PRHR HX leak and will require that the plant be shutdown before a PRHR HX leak could degrade into a tube rupture." What are the criteria (and supporting analyses) that Westinghouse is proposing to use in deciding when to shut the plant down before a PRHR HX leak could degrade into a tube rupture. Do these criteria take into account the much higher stresses that would be established in the PRHR HX tubes in case of an accident that requires the PRHR to operate? These higher stresses could cause the pre-existing defect (which causes the leak) to reach its "critical" size and become a rupture, thus adversely affecting the availability of the PRHR when demanded to operate to mitigate an accident. Please explain.

b. According to Revision 4 of the SAR (see pp. 16.1-459 through 16.1-465), it appears that the plant will be allowed to operate for an as-yet unspecified period of time even if the PRHR is declared inoperable. The implication of Action D.2, p. 16.1-464, is that this period of time may be indefinite if it is verified that the startup feedwater system (SFWS), in addition to the steam generators, is operable. Please clarify. If the Technical Specifications allow for T/M unavailability of the PRHR due to leaks, such unavailability should be included in the fault tree model.

c. Although several design features which reduce the likelihood of primary side corrosion are listed, none address the issue of secondary side corrosion which could accelerate under stagnant conditions by allowing local concentrations of ions or oxygen. Please list (with adequate explanation) the AP600 design and operational features that aim at preventing secondary side corrosion (e.g., how proper chemistry is ensured?). How do such features compare to features used to prevent secondary side corrosion in steam generator tubes?

d. Item (e) of Westinghouse's response lists several PRHR HX leak detection features. Please clarify the location of the pressure transmitter, its function and the type of information it provides about the leak. Also, please provide documentation showing that the RCS leak detection instruments stated in your response (i.e., containment sump level, containment radiation, and RCS mass balance) can be used to quantify reliably small leaks, such as those that would be allowed by your proposed technical specification.

Enclosure

e. Westinghouse argues that choosing a PRHR HX tube rupture event frequency of 5E-4/yr (a factor of 10 lower than the EPRI PRA KAG-recommended ALWR SGTR frequency of 5E-3/yr) is conservative. Some of these arguments seem to be valid. However, it is not clear that a factor of 10 reduction is justified, let alone conservative. One could attempt to rationalize the factor of 10 reduction by looking at the SGTR events that have occurred and screening out those events that are not applicable to the PRHR HX tubes. Due to these uncertainties in the assumed PRHR HX tube rupture frequency, please evaluate the sensitivity of PRA results to the PRHR HX tube rupture initiating event frequency and report the results in Chapter 59 of the PRA (Results and insights).

f. One of the arguments used to show that the PRHR HX tube rupture frequency is smaller than the frequency of SG tube ruptures, item (c), is that the primary side water has low oxygen content while the secondary side is "not really stagnant" and its temperature is normally low. Although, the lower oxygen content and low water temperature of the primary side do greatly reduce the problem of primary water stress corrosion cracking (PWSCC), it is less clear that the argument holds for the outside diameter stress corrosion cracking (ODSCC). The data for low temperature behavior of Alloy 690TT exposed to secondary water chemistry and crevice conditions are very limited or nonexistent. It is not clear what Westinghouse is referring to regarding the statement that the secondary side is "not really stagnant." Is there a circulating pump? If so, is the operation of this pump necessary to reduce the likelihood of corrosion on the secondary side of the tubes? Please explain.

720.328 Followon RAI related to DSER Open Item 19.1.3.1-2. Westinghouse's response to RAI #2, related to this open item, did not address the question. The question was: "The staff was unable to find in the revised PRA submittal a description of the analysis **with enough details** to understand how the contributions to intermediate, medium and large LOCA, reported in Section 3.5.3, were calculated. Please provide a clear description of the analysis (including assumptions, data and associated bases) used to calculate ADS spurious actuation frequencies and their contributions to the various LOCA initiating event frequencies." Please explain how the methodology, given in Section 26.5.3, for calculating the frequency of spurious ADS actuation from a 2 out of 2 signal train applies to the fault tree ADS-IC83 which uses a 2 out of 4 logic.

720.329 Followon RAI related to DSER Open Item 19.1.3.1-4. DSER Open Item 19.1.3-4 concerns LOCA sequences with impaired containment. These sequences (leading to endstate #2) were not quantified in Revision 0 (pre-DSER) of the PRA. The staff requested Westinghouse to either modify the event trees by modeling recovery actions or

count these sequences as leading to core damage with open containment. Westinghouse responded by removing the top event CI (containment not impaired) from the event trees in the revised PRA. According to Westinghouse, top event CI is not needed because analyses show that sufficient water for long-term recirculation cooling of the core is available for at least 2.7 days when containment isolation fails. Westinghouse argued that the use of a 24 hour mission time for long-term cooling was adequate for all accident scenarios.

In a follow-up RAI the staff asked Westinghouse to either show (e.g., through a bounding analysis) that the residual risk (beyond 24 hours) is not significant or extend the event tree models beyond 24 hours (to a point in time where it can be argued that the residual risk is not significant). Although statements made in Westinghouse's response to the follow-up RAI seem to agree with the staff regarding the need to look beyond 24 hours (e.g., "core damage is assumed....if core damage is anticipated following 24 hours without further system or operator action"), the residual risk issue was not addressed. If long-term cooling must continue (e.g., beyond the estimated 2.7 days), what actions are needed to be performed by the operator and what systems must be available to perform these actions? How important are such actions and systems to plant risk? Please provide documentation, including important assumptions.

720.330 Followon RAIs related to DSER Open Item 19.1.3.1-6. DSER Open Item 19.1.3-6 concerns the mission time (assumed to be 24 hours) for long-term cooling in sequences such as those where the reactor is initially maintained at high pressure (i.e., non-LOCA sequences with the start-up feedwater or the passive RHR available). Examples are:

a. <u>Sequences ending with startup feedwater system operating</u>: Operator action is needed to replenish the condensate storage tank (CST).

b. <u>Sequences ending with passive RHR operating</u>: The IRWST water starts boiling (Westinghouse analyses show that it reaches saturation in about one to two hours). If the evaporated IRWST inventory does not return to the IRWST, which is probable, the heat exchanger will be uncovered at some time (estimated by Westinghouse to be beyond 24 hours) and the IRWST inventory must be replenished or the plant must be depressurized to continue core cooling by recirculation. What actions are needed to be performed by the operator to bring the plant to cold shutdown conditions and what systems must be available to perform these actions? How important are such actions and systems to plant risk? Please provide documentation, including important assumptions.

720.331 Followon RAIs related to DSER Open Item 19.1.3.1-6. Another example of sequences, categorized as successful in Revision 2 of the PRA, which need additional development or explanation are sequences with an open path outside containment (e.g., sequences initiated by a

steam line break or a stuck open secondary side valve with consequential SGTR) and normal RHR available for long-term core cooling. These "success" sequences, as modeled in Rev 2, end with normal RHR operating. This scenario eventually requires replenishing the IRWST or sump inventory (because it is lost through the open path outside containment). This is true (although to a lesser extent) also in sequences when IRWST injection and passive sump recirculation is used instead of normal RHR. Can recirculation (either using the normal RHR pumps or by gravity) be established for long-term cooling when a considerable amount of inventory has been lost (and in some sequences continues to be lost during passive recirculation) through the open path to the atmosphere? What actions are needed to be performed by the operator and what systems must be available to perform these actions? How important are such actions and systems to plant risk? Please provide documentation, including important assumptions.

720.332 Followon RAI related to DSER Open Item 19.1.3.1-10. The staff requested Westinghouse to assess and document the applicability of generic failure data to the AP600 design. While check valves are not unique to the AP600, the conditions under which they will be operating in the plant are substantially different from those in current generation nuclear plants. The concern is that they will have to open on demand under very low differential pressures after long periods of being held closed by fluid at RCS temperature, pressure and chemistry. Westinghouse responded that this is not an issue anymore because some check valves in the IRWST injection line have been replaced with squib valves which "reduces the number of check valves and eliminates the high differential pressure normal operating environment that the valves in the IRWST injection and recirculation lines would experience in the previous design." However, Westinghouse's response does not fully address the fact that these valves will have to open on demand under very low differential pressures (because gravity is being used instead of pumps). Useful information could be obtained by looking at failure histories of check valves at operating nuclear power plants that must open on demand under small differential pressures, such as the check valves used as vacuum breakers at the turbine exhaust lines for BWR HPCI and RCIC systems (the lines that go from the turbine exhaust to the suppression pool). Please address this question in your next response. Also, as part of the insights section (Chapter 59), please include sensitivity studies that assess the impact of potentially higher failure rates for such check valves to risk.

720.333 Followon RAI related to DSER Open Item 19.1.3.1-13. The staff requested Westinghouse to explain why in calculating the common cause failure (CCF) probability of the IRWST injection line check valves, MGL factors from Revisions 5 and 6 of EPRI's Utility Requirements Document (URD) were used. A beta factor of 0.026 is recommended in Revisions 5 and 6 of the URD which is much lower than the value recommended in previous revisions of the URD (i.e., 0.17) as well as in previous PRAs (e.g., System 80+). Westinghouse responded that the reduced value of the beta factor for check valves reported in Revisions 5 and 6 of EPRI's URD, as compared to the

value recommended in previous revisions of the URD, was due to better understanding of individual events involving failure of check valves at nuclear power plants. It is further stated in Westinghouse's response that "EPRI found no common cause failures to open of check valves (other than failure modes unique to testable check valves)." Please explain what you mean by "failure modes unique to testable check valves" and why such failure modes do not apply to check valves used in the AP600 design. An NRC-sponsored evaluation of LER and NPRDS events, which occurred between 1980 and 1993 at operating nuclear power plants, has found about twenty (20) events involving common cause failure of check valves. Such events should be reviewed for applicability to the AP600 design. Please state the AP600 design and operational features which ensure that such events cannot occur with AP600 check valves.