52-003

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

March 27, 1996

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

SUBJECT: FOLLOWON QUESTIONS CONCERNING THE AP600 LEVEL 1 PROBABILISTIC 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 level 1 PRA results and insights associated with steam generator tube rupture accident sequences.

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 NRC'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
Standardization Project Directorate
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Office of Nuclear Reactor Regulation

Docket No. 52-003

Enclosure: As stated

cc w/enclosure:
See next page

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DOCUMENT NAME: A:SGTR PRA.SPL (3C AP600 DISK)
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Docket No. 52-003
AP600

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Enclosure to be distributed to the following addressees after the result of the proprietary evaluation is received from Westinghouse:

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AP600 PRA REVIEW
REQUEST FOR ADDITIONAL INFORMATION

720.324 As part of the AP600 PRA review, the staff is performing confirmatory re-quantification of selected accident sequences. To compare with Westinghouse's results, lists of top minimum cutsets for each of the sequences quantified by Westinghouse are needed. Each list should provide an adequate number of cutsets to be used for a meaningful comparison of results (e.g., top 50 cutsets or, if less than 50, top cutsets contributing to 99 percent of the sequence frequency). Please provide such lists.

720.325 An important insight, reported by Westinghouse in Chapter 59 (Results) of Revision 6 of the PRA, is that the contribution of the SGTR event to the at power core damage frequency (CDF) is very small (about 1.5 percent). If true, given the low total CDF estimate for the AP600 design, this is a significant improvement with respect to operating pressurized water reactors (PWRs). One of the reasons, reported by Westinghouse, for the small contribution of SGTR to CDF is that "the first line of defense is the startup feedwater system (SFWS) and chemical and volume control system (CVCS)." Please provide documentation showing that operation of CVCS only provides adequate flow for inventory control and that there is sufficient time to stabilize the plant before core uncovery occurs. Such documentation should clearly state all major assumptions made in the analysis.

720.326 Please provide the following information concerning the SGTR event tree model.

a. The description of event CVCS in Section 4.10.2 (page 4-29), entitled "Event Tree Model and Nodes," is not referring to the function (i.e., inventory control) CVCS is assumed to serve in the SGTR event tree. Please explain.

b. Event SGISO (failure to isolate the ruptured steam generator (SG)), as modeled by Westinghouse, does not include the possibility of an unisolable leak (e.g., stuck open SG power-operated relief valves (PORVs)/safety valves or atmospheric dump valves (ADVs)). If an unisolable path exists from the ruptured SG to the atmosphere, the differential pressure between the primary and the secondary side will remain high since the ruptured SG could be at or near atmospheric pressure. This scenario would require decreasing the primary pressure down to the atmospheric pressure to terminate the leak prior to depletion of the available RCS inventory. Please explain why this scenario is not modeled in the SGTR event tree. If your answer is that unisolable leaks cannot occur, include appropriate documentation to support this assumption.

Enclosure

c. Please list and describe the specific AP600 design features that reduce the probability of SGTR events resulting in containment bypass with respect to operating reactors. Such features should improve SGTR diagnosis, increase the time available for operator actions, lead to less reliance on operator actions and reduce the likelihood of challenging the secondary side safety valves. Please refer to applicable event tree models and related analyses.

d. It is stated in page 6-13 (Chapter 6, Success Criteria Analysis, of the PRA) that "the passive response paths on the SGTR event tree pessimistically model active SG isolation, which would not be required, since turbine trip would provide an alternative to active SG isolation." However, this is not a "pessimistic" modeling of the "passive response paths." On the contrary, credit is taken for SG isolation in all "passive response paths," as indicated by the multiplication of the frequency of these paths by the probability of failure to isolate the ruptured SG (events CIB and CIB/SGHL). Please clarify.

e. Following the SGTR event, an Emergency Safeguard Features (ESF) actuation signal is generated due to low pressurizer pressure. The ESF signal is supposed to trip the RCPs and actuate the CMTs. However, a statement made in page 4-27 implies that the event can be terminated by use of nonsafety systems and operator actions only. Please explain.

f. It seems that there are discrepancies between the PRA modeling of the SGTR accident sequences and the AP600 Emergency Response Guidelines (see Guideline AE-3, Revision 1, July 28, 1995). For example, Table 2-1 of the ERGs shows CMT and PRHR actuation and need for operator action to isolate them when certain conditions are met. Please provide documentation explaining how applicable emergency procedures are incorporated into the PRA models of SGTR accident sequences.

g. The following statement is made (see Chapter 4, page 4-28 of Revision 2 of the PRA): "Analyses show that no overfilling occurs and no automatic depressurization is actuated, even if multiple tubes have ruptured in the steam generator." Please provide documentation justifying the reason for not modeling in the PRA multiple SGTR events. This should include, in addition to the frequency of the initiating event, time windows available for required operator actions to isolate the faulted SG and stabilize the plant.

h. Please provide documentation justifying the reason for not modeling in the PRA SGTR coincident with loss of offsite power.