September 7, 1995

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATED TO THE AP600 PROBABILISTIC RISK ASSESSMENT (PRA)

Dear Mr. Liparulo:

Enclosed are the Nuclear Regulatory Commission's (NRC) staff comments on the AP600 PRA. The enclosure contains RAIs related to the level 1 PRA for internal events and power operation. You are requested to provide a response to these questions and comments within sixty days of receipt of this letter.

These followon questions affect nine or fewer respondents, and therefore, this request is not subject to review by the Office of Management and Budget under P.L. 96-511. If you have any questions regarding this matter, you may contact me at (301) 415-8465.

Sincerely,
Original signed by
Michael X. Franovich, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 52-003

Enclosures:
As stated

cc w/enclosures:
See next page

DISTRIBUTION:

Central File	PDST R/F	DCrutchfield
PUBLIC	BGrimes	TQuay
RArchitzel	MFranovich	TKenyon
WHuffman	DJackson	JMoore, 0-15 B18
WDean, EDO	MSiemien, OGC	GSuh (2), 0-12 E4
SDinsmore, 0-10 E4	NSaltos, 0-10 E4	RPalla, 0-8 H7
ACRS (11) w/o encl.	JFlack, 0-10 E4	

DOCUMENT NAME: A: PRARAI2.MXF
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PDST:DRPM		SC:PDST:DRPM	E		
NAME	MXFranovich		RArchitzel			
DATE	09/ 7 /95		09/ /95			

DF03

cc: Mr. B. A. McIntyre
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
P.O. Box 355
Pittsburgh, PA 15230

Mr. John C. Butler
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
Box 355
Pittsburgh, PA 15230

Mr. M. D. Beaumont
Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
One Montrose Metro
11921 Rockville Pike
Suite 350
Rockville, MD 20852

Mr. Sterling Franks
U.S. Department of Energy
NE-42
Washington, DC 20585

Mr. S. M. Modro
EG&G Idaho Inc.
Post Office Box 1625
Idaho Falls, ID 83415

Mr. Frank A. Ross
U.S. Department of Energy, NE-42
Office of LWR Safety and Technology
19901 Germantown Road
Germantown, MD 20874

Mr. Ronald Simard, Director
Advanced Reactor Programs
Nuclear Energy Institute
1776 Eye Street, N.W.
Suite 300
Washington, DC 20006-3706

STS, Inc.
Attn: Lynn Connor
Suite 610
3 Metro Center
Bethesda, MD 20814

Mr. James E. Quinn, Projects Manager
LMR and SBWR Programs
GE Nuclear Energy
175 Curtner Avenue, M/C 165
San Jose, CA 95125

Mr. John E. Leatherman, Manager
SBWR Design Certification
GE Nuclear Energy, M/C 781
San Jose, CA 95125

Barton Z. Cowan, Esq.
Eckert Seamans Cherin & Mellott
600 Grant Street 42nd Floor
Pittsburgh, PA 15219

Mr. Ed Rodwell, Manager
PWR Design Certification
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Mr. Charles Thompson, Nuclear Engineer
AP600 Certification
U.S. Department of Energy
NE-451
Washington, DC 20585

AP600 PRA REVIEW
REQUEST FOR ADDITIONAL INFORMATION
SEPTEMBER 5, 1995

RAIs Related to DSER Open Item 19.1.3.1-1

1. The Passive Residual Heat Removal (PRHR) tube rupture frequency was chosen by Westinghouse to be 5.0E-4/year on the basis that it should be approximately an order of magnitude lower than the frequency of a Steam Generator Tube Rupture (SGTR) event. If Westinghouse's approach, based on a pipe break failure rate of 4.25E-10 per year per section, was followed, this frequency would be 5.0E-3/year. If the failure rate for PRHR heat exchangers of 1.0E-7/year (recommended by EPRI in its Utility Requirements Document) were used, the PRHR tube rupture frequency would be 2.0E-3/year.

   Please re-evaluate the PRHR tube rupture frequency by taking into account the following: (1) it is not possible to isolate and repair a single leaking PRHR heat exchanger without a plant shutdown, (2) the possibility of stress corrosion which accelerates under stagnant conditions by allowing local concentrations of ions or oxygen, (3) the efficiency of detecting very small leaks to a very large body of water (in the IRWST), under stagnant conditions, may not be better than the leak detection capability of circulating primary in the steam generators, (4) the potential impact of mechanical loads on heat exchanger tubing and supports, including potential steam hammer load caused by phase separation within the tubes under accident conditions, and (5) the smaller heat transfer area of PRHR heat exchanger, as compared to steam generators, combined with the potential for two-phase flow in the IRWST side of the tubes during accident conditions where critical heat flux and vapor blanketing of the tubes may be of concern.

2. The primary system pipe break analysis assumes a certain apportionment of the failure rate, according to pipe sizes, into "large", "medium", "intermedium" and "small" LOCAs. Although such apportionment is logical, the assumed percentages are rather arbitrary. Sensitivity analyses are needed to assess the impact of this apportionment on the PRA results and insights.

3. The next PRA revision should reflect the PRHR design change, i.e., one instead of two heat exchangers.

RAIs Related to DSER Open Item 19.1.3.1-2

1. Westinghouse is requesting the extension of the testing interval, from quarterly to semi-annually, for the ADS stage 1, 2 and 3 motor operated valves (MOVs). The FSAR states in 3.9.6.3.1 that the ADS stage 1 through 3 valve exercise testing represents a risk of loss of coolant and depressurization of the reactor coolant system if the test sequence is not followed. Operator error during exercise testing of ADS MOVs (e.g., failure to follow test sequence) must be addressed in the PRA.

2. A methodology is given in Section 26.5.3 for calculating the frequency of spurious ADS actuation from a 2 out of 2 signal train. Section 11.1.2, however, indicates that ADS actuation is based on 2 out of 4 level detectors in either of the 2 CMTs, which includes 12 possible combinations of 2 signals. 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.

<u>RAIs Related to DSER Open Items 19.1.3.1-4 and 19.1.3.1-6</u>

1. Westinghouse assumes a mission time of 24 hours for long-term cooling independently of plant condition. This assumption must be justified by showing (e.g., through a bounding analysis) that the remainder risk (beyond 24 hours) is not significant. Otherwise the event tree models must be extended beyond 24 hours (to a point in time where it can be argued that the remainder risk is not significant).

<u>RAIs Related to DSER Open Item 19.1.3.1-7</u>

1. Westinghouse needs to correct several inconsistencies or provide an explanation indicating that the apparent inconsistency resulted from a misunderstanding. Several entries in the "System Dependency Matrix" tables, at the end of each specific system chapter, are inconsistent with the "AP600 Support System Interdependency Matrix" table located in Chapter 5. Examples are:

   - For the Passive Containment Cooling System (PCS), Table 13-4 on page 13-9 of the PRA, shows that IDS is the support system required to operate AOVs and MOVs. However, PCS-PCT in Table 5-6 on page 5-30 of the PRA does not show that the IDS system is a support system.

   - For the Normal Residual Heat Removal System (RNS), Table 17-4 on page 17-10 states that PLS system provides manual actuation logic for pumps, MOVs, etc. However, RNS-RHR and RNS-RNP (Table 5-6 on page 5-33) indicate the PMS system (not the PLS system) provides support.

   In addition, Section 21.4.2 refers to subsection 8.3.1 of reference 21-1. Reference 21-1 is the revision 1 fault trees and there is no subsection 8.3.1. The correct reference should be given.

<u>RAIs Related to DSER Open Item 19.1.3.1-10</u>

1. 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. For example, 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. In the revised PRA submittal the failure rate of the IRWST check valves was changed, as suggested in EPRI's Utility Requirement Document, to account for "less than ideal conditions" which may exist at the time the valves are demanded. However, no discussion is included in the submittal which shows that this change addresses the failure data applicability concern for the IRWST check valves or for any other components. Please provide this information and/or perform sensitivity studies to assess the impact of changes in failure rates of risk-important components to risk.

RAIs Related to DSER Open Item 19.1.3.1-11

1. The staff was unable to find in the revised PRA submittal a complete response to DSER Open Item 19.1.3.1-11. Please provide documentation of I&C failure data derived from Westinghouse data or identify specifically where this information can be found.

RAIs Related to DSER Open Item 19.1.3.1-13

1. 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. This 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+). No explanation for this is provided. Please explain the reasons for assuming a value for such beta factor which is considerably lower than values used in previous PRAs.

RAIs Related to DSER Open Item 19.1.3.1-14

1. The staff was unable to find in the revised PRA submittal the beta factor, or MGL parameter, values used in calculating common cause failure probabilities of I&C hardware components (as requested in the DSER Open Item 19.1.3.1-14). Please provide this information, including sources and related documentation. In addition, please provide detailed documentation of the calculation of probabilities for the most risk-important CCF events (in terms of both baseline and focused PRA results) related to I&C hardware components.