Docket File

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

March 12, 1997

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

SUBJECT: FOLLOWON QUESTIONS REGARDING THE AP600 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. Specifically, the staff has reviewed a December 13, 1996, letter from Westinghouse that provided a sensitivity study on the baseline PRA in response to a request from the staff. The study assumes that systems needed for normal plant operation (e.g., ac power) can be available if not affected by the initiating event (Open Item Tracking System #3969). This review was integrated with (1) the focused PRA documented in Chapter 52 of the PRA, (2) assumptions made in the baseline PRA which are likely to have a significant impact on the focused PRA results and (3) changes made in common cause failures (documented in Chapter 29 Revision 7).

As a result of this review some preliminary questions (Enclosure 1) were faxed to Westinghouse and a teleconference was held on February 12, 1997, to discuss these questions. Based on this teleconference it was decided to turn the Enclosure 1 questions into formal request for additional information. Enclosure 2 contains the questions from Enclosure 1 as well as additional questions that were not discussed during the teleconference. Therefore, it is requested that Westinghouse formally respond to the questions in Enclosure 2.

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 Public Document Room.

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

Sincerely,

original signed by:

Joseph M. Sebrosky, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 52-003

Enclosure: As stated

cc w/enclosure:
See next page

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Mr. Nicholas J. Liparulo
Westinghouse Electric Corporation

Docket No. 52-003
AP600

cc: Mr. B. A. McIntyre
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
P.O. 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

Ms. Cindy L. Haag
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
Box 355
Pittsburgh, PA 15230

Mr. S. M. Modro
Nuclear Systems Analysis Technologies
Lockheed Idaho Technologies Company
Post Office Box 1625
Idaho Falls, ID 83415

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

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

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

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

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

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

Ms. Lynn Connor
DOC-Search Associates
Post Office Box 34
Cabin John, MD 20818

Mr. Robert H. Buchholz
GE Nuclear Energy
175 Curtner Avenue, MC-781
San Jose, CA 95125

Mr. Sterling Franks
U.S. Department of Energy
NE-50
19901 Germantown Road
Germantown, MD 20874

Mr. Charles Thompson, Nuclear Engineer
AP600 Certification
NE-50
19901 Germantown Road
Germantown, MD 20874

DISCUSSION ITEMS FAXED TO WESTINGHOUSE

1. Cut set #2: IEV-SGTR * ADF-MAN01 * RPX-CB-GO * ADN-MAN01C

   a. Are there any T-H analyses to support this sequence? Can the leak be stopped before uncovering the core or passing water through the secondary side safeties? How fast must the operator act to open ADS stage #1 valves, given that this is not the preferred means? (Notice that the operator will try to align CVCS in auxiliary spray mode first; according to W HRA, p. 30-29 of PRA, this action requires a long procedure and about 10 minutes of actual implementation time). Are there any procedures to follow?

   b. This scenario assumes that even when the operator action ADF-MAN01 fails, the accident can be mitigated by manually depressurizing the RCS using ADS. How is this done? How much time does the operator have to perform this action to avoid uncovering the core or overfilling the SG? What I&C system can the operator use to manually actuate ADS?

2. Cut sets # 3, 6, 7, and similar including CCX-SFTW. It is not clear why a CCF across both PMS and PLS is considered while PLS is not supporting any system credited in the analysis.

3. Cutset #11. Credit for DAS is taken (ATW-MAN04). Explain the reason.

4. Cutsets #34 and #65 (SGTR). Similar comments as for cutset #2. Are there any T-H analyses supporting the modeling of this sequence?

5. Cutsets # 40, #55, #59 and similar cutsets including more than one CCF of sensors and transmitters together with operator action(s). Need to understand how all this I&C failures impact the human error probability.

6. A change in the modeling of RCS leak events in the focused PRA was recently made which has a significant impact on the results. This change was not brought to the attention of the staff. A failure probability of CVCS of about 4E-3/d was assumed even though one of the two CVCS pumps is in standby during normal plant operation. This implies a high reliability/availability of this system. How is this assured?

7. In the latest revision of the PRA, the failure rate of IRWST check valves was changed from 1E-6/hr to 2E-7/hr. Same is true for the failure rate of explosive (squib) valves (changed from 3E-3/d to 5.8E-4/d). These changes, which are not backed up by adequate data or analyses, have a significant impact on the focused PRA results. In addition, common cause failure data either are not available (e.g., squib valves) or could be much higher than those used in the AP600 PRA (e.g., check valves). The staff needs to understand the bases for the above mentioned changes in previously used data in the PRA.

Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION CONCERNING THE AP600 PRA

720.371 Please explain the events and assumptions of cut set #2 (IEV-SGTR * ADF-MAN01 * RPX-CB-GO * ADN-MAN01C). In particular, the staff requests the following:

a. Are there any T-H analyses to support this sequence? Can the leak be stopped before uncovering the core or passing water through the secondary side safety valves?

b. How fast must the operator act to open ADS stage #1 valves (event ADF-MAN01), given that this is not the preferred means? As is documented in the PRA, the operator will try to align CVCS in auxiliary spray mode first. According to the HRA, p. 30-29 of PRA, this action requires a long procedure and about 10 minutes of actual implementation time. Are there any procedures the operator must follow? Does event ADF-MAN01 correspond to a system level actuation or to actuation of individual stage 1 valves using PLS?

c. This scenario assumes that even when the operator action ADF-MAN01 fails, the accident can be mitigated by manually depressurizing the RCS using ADS (event ADN-MAN01C). On what event(s) is the probability of event ADN-MAN01C "conditional?" How much time does the operator have to perform this action to avoid uncovering the core or overfilling the SG, given the other event(s) will have to be diagnosed and potential actions completed first? Is the modeling of this scenario in agreement with the procedure that operators must follow? Please explain by referring to HRA and other analyses documented in the PRA.

d. Cutset #34 (IEV-SGTR * RPX-CB-GO * ADN-MAN01) and cutset #65 (IEV-SGTR * RPX-CB-GO * LPM-MAN01) imply a different emergency response procedure than cutset #2 for same scenario. What do the emergency response procedures instruct the operator to do when a SGTR event is followed by failure to trip of one or more RCPs? If the operator is instructed to depressurize the RCS, what are the times available for diagnosis and action? Please provide the basis for the assumed success criteria for the systems used to mitigate the accident and for the time windows used in HRA.

720.372 Several cut sets (such as #3, #6 and #7) include common cause software failure across both PMS and PLS (event CCX-SFTW). It is not clear why event CCX-SFTW is considered, given that PLS is not supporting any system credited in the analysis, instead of common cause software failure within PMS only (event CCX-PMXMOD1-SW). Please explain.

720.373 It appears that credit for DAS (a nonsafety-related system) is taken in the focused PRA (see event ATW-MAN04 in cutset #11). Also, documentation is needed to support the assumptions made, with respect to unfavorable exposure time (UET) and related pressure relief capability, in modeling ATWS events in the PRA. It seems that the AP600 ATWS model was based on work performed for operating Westinghouse PWRs (documented in WCAP-11993, December 1988). There are concerns with the applicability of the work documented in WCAP-11993 to the AP600 design. For example:

a. WCAP-11993 indicates that, for a 24-month cycle, the primary pressure will <u>not</u> exceed 3200 psig if both PORVs open (i.e., UET is zero), given manual rod insertion (MRI) is successful and all auxiliary feedwater (i.e., 100 percent flow from both motor driven pumps and the turbine driven pump) is available. This is assumed to be applicable in the AP600 design without the benefit of any thermal-hydraulic and/or neutronic analyses. Please provide the basis for the assumption made in the AP600 PRA that if either the PRHR or both SFWS pumps are available the "all feedwater flow" condition of the WCAP-11993 study is satisfied.

b. According to the WCAP-11993 study, the probability of operator failure to act within one minute to step in the control rods is 0.21 (WCAP-11993 page 4-20) which is much higher than the 3.3E-2 assumed in the AP600 PRA. In addition, as stated in WCAP-11993 page 3-8, SECY-83-293 (the basis for the ATWS rule) does not allow for short-term operator action to manually insert control rods to mitigate the transient. Please explain.

720.374 Several cutsets include more than one CCF of sensors and transmitters together with operator action(s), such as #40 and #59. Please verify that all these I&C common cause failures do not adversely impact the human error probabilities (as calculated in the PRA) and provide documentation of your finding in the focused PRA.

720.375 A change in the modeling of RCS leak events was made in the latest revision of the focused PRA which has a significant impact on the results. A failure probability of CVCS of about 4E-3/d was assumed even though one of the two CVCS pumps is in standby during normal plant operation. This implies a high reliability/availability of this system. How will this reliability/availability be assured?

720.376 In the latest revision of the PRA, the failure rate of IRWST check valves was changed from 1E-6/hr to 2E-7/hr. Same is true for the failure rate of explosive (squib) valves (changed from 3E-3/d to 5.8E-4/d). These changes, which are not backed up by adequate data or analyses, have a significant impact on the focused PRA results. In addition, common cause failure data either are not available (e.g., squib valves) or could be much higher than those used in the AP600 PRA (e.g., check valves). The staff needs to understand the

bases for the above mentioned changes in previously used data in the PRA. Following a telephone conversation with the staff, Westinghouse submitted data (obtained from Sandia National Laboratories) which were used to develop the revised failure rates for squib valves. The Sandia data, however, are for a specific design of standardized mini-valves used in weapons systems. Please explain how the Sandia data can be applied to AP600 squib valves.

720.377 No reason is documented in the PRA for not modeling common cause failure (CCF) of check valves belonging to different systems, such as CMTs and Accumulators. This can have a significant impact on the focused PRA results. Please provide the basis for not including such CCFs in the PRA.

720.378 The reactor vessel failure frequency assumed in the AP600 PRA was recently changed from 3E-8/yr to 1E-8/yr without any explanation. This is an order of magnitude lower than the WASH-1400 value of 1E-7/yr. Please explain.

720.379 The common cause failure (CCF) probabilities for one IRWST injection line (event IWX-EV1-SA for squib valves and event IWX-CV1-AO for check valves) were calculated as the failure of 2 out of 4 valves instead of 2 out of 2 valves. Please explain.

720.380 The staff could not find in the PRA an explanation of the assumed common cause failure probability for the reactor trip breakers (failure to open). In Chapter 32 of the PRA (Data Analysis and Master Data Bank), the failure rate of PWR reactor trip breakers is listed to be 3E-3/d (page 32-13) while the common cause multiplier for a group of four breakers is listed as 6E-2 (page 32-27). This implies a much higher CCF probability for the reactor trip breakers than the 8.1E-6 value currently used in the AP600 PRA. In page 32-13, however, it is mentioned that a different failure rate was used in the PRA and that this was explained in Chapter 26 of the PRA. The staff was unable to find such explanation in Chapter 26. Please explain how the assumed CCF probability for the reactor trip breakers was calculated. Compare the calculated Also, please list the reasons the AP600 reactor trip breakers are assumed to be significantly more reliable than similar breakers in operating and evolutionary PRW reactor designs.

720.381 In Chapter 26 of the PRA (page 26-3) it is stated that "the value of 1.8E-06 failures/demand is used for mechanical failure of multiple rod cluster control assemblies to insert." Please explain why this failure does not appear in the submitted cutsets (for both baseline and focused PRA).

720.382 The probability of failure to trip the reactor through the Motor-Generator set, which involves the failure of both 480 V breakers to open (event MGSET), was calculated assuming a failure rate of

1E-7/hr and a two-year test interval (see page 31-4 of PRA). According to Table 32-1, the failure rate of 1E-7/hr was derived from the demand failure rate of 1E-3/d (recommended in EPRI's URD) by assuming monthly testing (note #4, page 32-18). However, using the above assumptions, the staff calculated a standby failure rate of about 3E-6/hr which is much higher than the 1E-7/hr used in the AP600 PRA. Please explain. Also, list assumptions with associated bases used in the model for converting the demand failure rate to hourly failure rate, such as test interval, failures from standby stresses (e.g., corrosion, dirt, lack of lubrication) and failures from stresses put on the component when it is demanded or operated (e.g., vibration, wear and torque).

720.383 It is assumed that the majority of the transient initiating event categories (grouped as event IEV-ATWS-T) do not require reactor trip for about 10 minutes (see page 6-58). This assumption may be optimistic since event IEV-ATWS-T includes some relatively frequent transients which tend to produce RCS pressure transients, such as loss of RCS flow, turbine trips and loss of main feedwater to one steam generator. Please explain.