January 21, 1997

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

SUBJECT: COMMENTS ON THE AP600 STANDARD SAFETY ANALYSIS REPORT (SSAR) CHAPTER 15 ACCIDENT ANALYSES

Dear Mr. Liparulo:

The Nuclear Regulatory Commission review of AP600 accident analyses has determined that additional information will need to be included in Chapter 15 of the AP600 SSAR for the staff to complete its review. The staff's comments on the Chapter 15 analyses are provided as an enclosure to this letter. It is not necessary for Westinghouse to submit a separate correspondence addressing its action on each item as long as it intends to provide the requested information in the next revision to Chapter 15. However, Westinghouse should provide written explanation to the staff for those comments that it will not incorporate into the SSAR. In addition, it is requested that these comments be included in the open item tracking system so that the status and disposition of these items can be tracked.

We also have included in this letter a formal request for additional information (RAI 440.588) concerning the staff's review of your multiple steam generator tube rupture analyses. This RAI is related to one of the key technical issues identified in our letter to you on December 6, 1996.

If you have any questions regarding this matter, you may contact me at (301) 415-1141.

Sincerely,

original signed by:

William C. Huffman, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 52-003

Enclosure: As stated

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See next page

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Docket No. 52-003
AP600

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NUCLEAR REGULATORY COMMISSION COMMENTS
CONCERNING
WESTINGHOUSE AP600 SSAR CHAPTER 15 ACCIDENT ANALYSES

1. Discuss the NRC review status of all the computer codes used in the transient and accident analyses documented in AP600 SSAR. For those codes previously approved by the NRC, list the NRC approval letters and address the continued acceptability of the approved codes for the AP600 analyses. Address how the AP600 analyses comply with any limitations which my be imposed on code usage.

2. Provide a list of single failures considered in your analyses, and perform a systematic assessment for each transient and accident and justify why the limiting single failure determined is adequate. In the list of single failures requested, identify the items that are non-safety related and explain why a failure of these components would result in the most conservative analysis.

3. GDC 17 in 10 CRF Part 50, Appendix A requires, in part:
"An onsite electric power system and offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specific acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents."
In accordance with the requirements of GDC 17, a loss-of-offsite-power should not be considered as a single failure event and should be assumed in the analysis for each event without changing the event category. Discuss in each transient and accident analyses in the AP600 SSAR how the analyses conform to the GDC 17 requirements given above. If the existing analyses did not conform to the GDC 17 requirements, Westinghouse should reanalyze the transient and accident analyses in accordance with GDC 17.

4. In several Chapter 15 analyses, nonsafety-related equipment was credited for accident mitigation. The equipment includes the main feedwater pump trip, turbine stop and control valves, pressurizer heater block, and main steam branch isolation valves. Westinghouse has summarized these systems in Table 2-2 of WCAP-14477 (The AP600 Adverse System Interactions Evaluation Report). A similar summary table should be included in Chapter 15.

   The staff notes that items designated under item a.5 of Table 2-2 in WCAP-14477 appear to be used to mitigate the consequences of the Chapter 15 design basis analyses. 10 CRF 50.36 specifies the criteria for the systems that are required to establish technical

Enclosure

specification limiting conditions for operation. Specifically, Item (c)(2)(ii)(C), Criterion 3, states that "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." Westinghouse should justify why these items should not be included in the AP600 technical specifications.

The staff notes that WCAP-14477 states that components associated with item a.5 of Table 2-2 will be included in the in-service testing (IST) program. However, the staff finds that the systems are not included in the IST program described in the SSAR 3.9.6. Westinghouse should provide further discussion on this proposed action for the non-safety related systems and explain how it is addressed in the SSAR.

5. Chapter 15 should include a table or similar summary documentation of system actuation times, valves closure time and systems parameters assumed in each safety analysis.

Increase in Feedwater Flow (SSAR Section 15.1.2)

6. This analysis should include the calculated transient DNBR, RCS and steam generator pressures to support the conclusion for the full-power case that meets the acceptance criteria of the condition II events.

Inadvertent Opening of a Steam Generator (SG) Relief or Safety Valve (SSAR Section 15.1.4)

7. Core makeup tanks injection and the associated tripping of the reactor coolant pumps are initiated automatically by low pressurizer pressure safeguards signal(page 15.1-11), while the sequence of event lists the low cold temperature safeguard to actuate CMT (page 15.1-26). This inconsistency should be resolved. SSAR Section 15.0 should include the safeguards setpoints and the associated actuation delay times assumed in the transient and accident analysis (see comment #5), and discuss any measures taken to assure that the safeguards setpoints in technical specifications are conservative when compared with the setpoints assumed in the analyses.

8. Provide the DNBR curve in the SSAR section for the transient.

Steam System Piping Failure (SSAR 15.1.5)

9. Provide the DNBR curve in the SSAR for the transient.

10. The review staff will need the following additional information to complete its review of the steamline break analysis:

    (a) To confirm that the zero-power case is the worst post-trip return-to-power case, a discussion should be provided on why the analysis for a SLB inside the containment at full power condition with and without concurrent loss-of-offsite-power in combination with a single failure is less limiting.

    (b) To identify the worst case that maximizes the potential for fuel degradation and dose at the site exclusion area boundary, provide a discussion of analysis for the following two cases:

    - a SLB outside the containment, from full power conditions, in combination with a LOOP, a single failure, a stuck RCCA, and Technical Specification allowable steam generator leakage, and

    - a SLB outside the containment upstream of the MSIV at zero power with LOOP in combination with a single failure, a stuck RCCA, iodine spike and TS SG leakage.

    The results of sensitivity studies for existing plants (such as WCAP-9226) to identify the worst SLB case for the AP600 are not sufficient to address Questions (a) and (b) above due to the nature of the passive designs and the differences in plant configurations between the existing traditional PWR and the AP600.

11. Explain why the single failures such as failure of the MSIV to close in the intact SG, and failure of feedwater isolation to close are not considered in the SLB analysis in assessment of the cooldown effect or the SG overfilling issue.

Inadvertent Operation of the Passive Residual Heat Removal System (SSAR 15.2.1.6)

12. Provide a discussion of the impact of the most single limiting failure on the system response for this case.

13. Explain how the zero-power case is bounded by the inadvertent opening of a SG relief or safety valve.

Turbine Trip (SSAR 15.2.3)

14. This analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on the turbine trip event should be considered.

Loss of ac Power to the Plant Auxiliaries (SSAR 15.2.6)

15. Provide a DNBR transient curve for a loss of ac power event.

Loss of normal Feedwater Flow (SSAR 15.2.7)

16. Provide a DNBR transient curve for a loss of normal feedwater flow event.

17. This analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on the loss of normal feedwater flow event should be considered.

Feedwater System Pipe Break (SSAR 15.2.8)

18. The loss of offsite power, resulting in a RCS flow coastdown, is assumed at the time of the reactor trip. Westinghouse has stated that this is more limiting than the case where power was lost at the initiation of the event. An earlier RCS flow coastdown, which reduces the capacity of the primary coolant to remove heat from the core, may result in a larger increase in the peak RCS pressure. This analysis should include the technical basis needed to show that a coincident loss of power with a reactor trip will result in a highest peak RCS pressure during a feedwater line break event.

19. This analysis (page 15.2-17) assumes a double-ended rupture occurs after a reactor trip on the low-low steam generator level signal. This is inconsistent with the sequence of events in Table 15.2-1 (sheet 5 of 5) that shows initiation of the feedwater line break before the reactor trip. This inconsistency should be corrected.

20. The staff noted that the non-safety related startup feedwater system (SFWS) and the pressurizer spray (PS) were credited (page 15.2-19) for heat removal to limit the increase in the peak RCS and steam generator pressures. Use of these non-safety related systems to mitigate the consequences of the feedwater line break event is not acceptable to the staff. Either reanalyze the feedwater line break event without credit of the SWFS and PS, or demonstrate that the effects of the initiation of both non-safety related systems on the feedwater line break event is insignificantly small. Also, it was noted that the pressurizer safety valves were set at a minimum value. Justify the use of the lowest pressurizer safety setpoint for overpressurization prediction for an FLB heatup event.

21. The DNBR results for this event should be included in the SSAR. The calculated minimum DNBR should be greater than the safety limit DNBR for acceptance.

22. The Semiscale test data for feedwater line breaks (as discussed in Section 4.3.3.1 of NUREG/CR-4945, dated July 1987) showed that the steam generator heat transfer capacity remains unchanged until the steam generator liquid inventory is nearly depleted. This is followed by a rapid reduction to 0 percent heat transfer with little further reduction in the steam generator water inventory. In light

of these test data, provide a discussion of the steam generator heat transfer model used in the feedwater line break analysis and verify the model is conservative as it is compared with the Semiscale test data. If the model is found to be nonconservative, reanalyze the feedwater line break event by using the model that is supported by the test data including the Semiscale test data. With a heat transfer model consistent with the Semiscale test data, perform a sensitivity study of break sizes to identify the worst break size and provide the results for the staff to review.

23. Figure 15.2.8-6 shows that the water level reaches the top of the pressurizer during a period of 6000 to 20000 seconds after an FLB event. Discuss the function of the pressurizer safety valves (PSV) assumed in the analysis during and after the period when the pressurizer is filled with water. Justification should be provided if the PSVs are assumed to reclosed during or after the period when the pressurizer was filled with water.

Partial Loss of Forced Reactor Coolant Flow (SSAR 15.3.1)

24. The staff found that no discussion was provided for the events with loss of one or three reactor coolant pumps. A discussion of the results of the loss of one and three reactor coolant pumps events should be provided.

Reactor Coolant Pump Shaft Break (SSAR 15.3.4)

25. No transient DNBR curve was included in the documentation, and the basis for assuming 18 percent of the fuel rods are failed for the radiological release assessment was not provided. A discussion on how the amount of the failed fuel was determined should be included in the SSAR along with justification of the adequacy of core coolability under the core conditions calculated.

    SRP Section 4.4 states that if the DNBR falls below the safety limit DNBR, fuel failure must be assumed for all rods that violate the safety limit DNBR. Any fuel damage calculated to occur must be sufficiently limited so that the core will remain in place and intact with no loss of core coolability. The staff will review the analysis to confirm compliance with the above acceptance criteria for the fuel performance.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (SSAR 15.4.1)

26. The SSAR should include a DNBR transient curve for this event.

27. The control rod withdrawal analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on this event should be considered.

28. DNBR results are not presented for this events with a dropped or misaligned RCCA (the original figure 15.4.3-5 was deleted). The figure should be included in the SSAR analysis of this event.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (SSAR 15.4.2)

29. The withdrawal at power analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on this event should be considered.

Rod Cluster Control Assembly Misalignment (SSAR 15.4.3)

30. The sequence of the events is not provided for the rod assembly drop event. This should be provide the in the SSAR.

31. Provide the calculated DNBR curve in the SSAR for this event.

32. The staff noted that for the single assembly withdrawal event, an infrequent event, no analytical results including the transient DNBR curve were provided. In addition, no basis was provided for the estimate of 5 percent failed fuel rods. Provide the analytical results for this event including a discussion for how the amount of the failed fuel was determined, and justify the adequacy of core coolability under the core conditions calculated.

    SRP Section 4.4 states that if the DNBR falls below the safety limit DNBR, fuel failure must be assumed for all rods that violate the safety limit DNBR. Any fuel damage calculated to occur must be sufficiently limited so that the core will remain in place and intact with no loss of core coolability. The staff will review the analysis to confirm compliance with the above acceptance criteria for the fuel performance.

33. The single RCCA withdrawal-at-power analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on this event should be considered.

Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (SSAR 15.4.4)

34. This analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on this event should be considered.

Boron Dilution (SSAR 15.4.6)

35. Recent non-conservatisms were identified at Comanche Peak related to the input assumptions and boundary conditions (inverse count rate ratio data and multiplication setpont) in the analyses of the

licensing basis boron dilution event. Based on this background, justify the use of the automatic actions to terminate the dilution and start boration which were assumed in the boron dilution analyses for AP600.

36. Discuss the technical bases for the charging flow of 200 gpm assumed in the analysis.

37. The value of the setpoint (60 percent increase/10 min) of the source range nuclear flux instrumentation used in the analysis to isolate the demineralized water storage isolation valves is inconsistent with that specified in item 15 of technical specification Table 3.3.2 (60 percent increase /50 min).

38. Discuss the time it would take from the start of the dilution to the loss of shutdown margin and then to reach criticality for modes 3 through 5. Discus the margins (the times from the DWS isolation to core criticality) with consideration of uncertainties associated with the calculational methods, input parameters, signal processing time and valves closing time.

Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (SSAR 15.4.7)

39. The staff has reviewed the consequences of the spectrum of postulated fuel loading errors. The analyses show for each case considered either the error would be detectable by the available instrumentation (and hence remediable) or the error would be undetectable, but the offsite consequences of any fuel rod failures would be a small fraction of 10 CFR 100 guidelines. A COL action item should be included in the SSAR to use the available in-core instrumentation before the start of a fuel cycle to search for fuel loading errors.

Rod Ejection Accident (SSAR 15.4.8)

40. Provide the analysis to show how 15 percent of the fuel experiencing DNB was calculated. Also, a loss-of-offsite-power should be considered in this analysis.

LOCA outside containment

41. Describe the program to assure that the isolation valves in the CVS discharge line and the sample line will be qualified to close upon demand during the piping break conditions. Also, discuss the location of the flow orifices which were credited to limit the blowdown flow for the LOCA events outside containment to 130 gpm. These orifices should be included in the ITAAC. A delay time of 30 minutes for the operator actions was assumed in the analysis. Justify the assumption of the operator delay time by considering the display of the instrumentation and procedures available for the accident mitigation.

Steam Generator Tube Rupture (SSAR 15.6.3)

42. The staff notes that for the AP600, the offsite power was assumed to be available in the SG overfill bounding analysis (page 15.6-9) while a loss-of-offsite-power (LOOP) was identified as one of the worst initial conditions in the analysis for existing plants (as documented in page 4-22 of WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill"). With offsite power available, the steam release from the affected steam generator will be released to the condenser. With the LOOP, the condenser will be unavailable and the steam will be released to the atmosphere through steam generator safety or power-operated relief valves. Explain the difference in the assumptions for offsite power conditions in the bounding SG overfill analyses for the AP600 design and existing plants.

43. The calculated DBNR curve should be included in the SSAR for this analysis.

Loss of Coolant Accidents (15.6.5)

<u>SBLOCA</u> (SAR 15.6.5)

44. Provide a technical basis for defining the break of 1.0 ft^2 as the largest SB size.

45. Expand the break size spectrum study to cover the full range of the SBLOCA up to the largest break size of 1.0 ft^2 at any locations in the RCS pressure boundary by using small break methods, and expand LBLOCA analyses from double-ended guillotine cold leg break to the lower end of break size of 1.0 ft^2 using the LBLOCA methods.

46. Provide the basis for the "judgment" that determined the failure of one of the fourth stage ADS valve to open as the limiting single failure. Discuss what sensitivity study of single failures was performed to determine the limiting failure.

47. The steam isolation valves are assumed to close one second after the reactor trip. Provide the basis for this assumption and confirm whether the steam isolation actuation is consistent with the design. Also, there are two numbers specified for each event in SSAR Tables 15.6.5-15 and -16 reporting the time for steam isolation valves to close. Delete the incorrect values from the tables.

48. Address why injection of ECCS into the RCS hot legs is not needed in the AP600 to prevent boron stratification in the reactor vessel.

<u>Large Break LOCAs</u>

49. Provide technical bases for the break size of 1.0 ft^2 that was used as the criterion to distinguish LB and SB LOCAs.

50. Figures 15.6.5A-1 through 64 for the analytical results of a LBLOCA were removed from Revision 3 of the SSAR chapter 15. No equivalent figures have been provided in subsequent SSAR revisions. The figures for the results of LBLOCA analysis should be included in the next update to the SSAR.

51. Item (a)(1)(i) of 10 CFR 50.46 requires that "... ECCS cooling performance must be calculated... for a number of postulated loss-coolant-accidents of <u>different sizes, locations,</u> (underlined for emphasis) and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated...." The scope of the Westinghouse LBLOCA analysis is very limited. The analysis provided was performed for a double-ended cold leg guillotine (DECLG) and a cold leg split (CLS) with a break area of one-half of that of a DECLG using three different flow discharge coefficients. Westinghouse is requested to perform a break size sensitivity analysis for the LBLOCA from the break size of a DECLG to the smallest LBLOCA (1.0 ft^2 cross section area - or the smallest break to which the WCOBRA/TRAC method can be applied).

For the analysis presented in the SSAR, the break was assumed to occur at one of the cold legs in the loop containing the CMT. The selection of the worst break location is based on the sensitivity studies in WCAP-10924, Westinghouse Large Break Best Estimate methodology, Volume 1 Model description and Validation, Volume 2, Revision 2, Application to Two-Loop PWR Equipped with Upper Plenum Injection, which showed that locating the cold leg break in the loop does not contain the pressurizer is conservative. Westinghouse is requested to extend the sensitivity study to cover the whole range of breaks at various locations. The results of sensitivity for the existing plants are not sufficient to justify for the selection of size and location for the worst LOCA case because the AP600 ECCS designs are significantly different from that of the conventional Westinghouse plant and the sensitivity studies for the existing plants may not be directly applicable to the AP600 design. Westinghouse is requested to justify the adequacy of selection of the break size and location for the limiting case and address its compliance with 10 CFR 50.46(a)(1)(i).

ATWS (SSAR 15.8)

52. Preform an ATWS analysis and include the results in the SSAR to demonstrate that the AP600 response is within the bounds covered by the ATWS Rule (10 CFR 50.62).

Shutdown Risk Assessment

53. To address Open Item 1002 in the DSER regarding the need of automatic containment isolation for non-safety related penetrations, Westinghouse responded, in a letter dated October 10, 1996, that normal RHR and CVS charging isolation valves will isolate on a containment radiation signal permitting these non-safety systems to continue to operate when in service unless there is an elevated

radiation level present. The staff is concerned that a spurious containment high radiation signal caused by radiation release spikes during outage activities may inadvertently isolate the normal RHR. Westinghouse is requested to discuss its analysis and/or plant procedures to justify the mitigation strategies for this inadvertent isolation of the RHR event during the refueling outage.

For a postulated RHR pipe break outside the containment, the containment radiation isolation signal will not actuate to isolate the RHR and terminate the break. Explain how this LOCA outside the containment is mitigated and discuss the need for automatic isolation of the RHR based on a low reactor water level signal.

AP600 MULTIPLE SGTR - CONTAINMENT BYPASS ISSUE:

RAI 440.588

SECY-93-087 required the design certification applicant to assess design features to mitigate containment bypass due to SGTR events. In a previous submittal, Westinghouse performed realistic SGTR analysis of up to 5 ruptured tubes with the results showing that the secondary pressure never reached the main steam safety valve setpoint. This analysis included an analysis of 5-tube rupture assuming the PORVs fail to open, and the result also showed that the secondary pressure did not exceed the MSSV setpoint.

Westinghouse, in August 22, 1995, letter, provided new SGTR analysis using the pressurizer volume of 1600 ft^3, which is consistent with the AP600 design. The result showed that, if the PORVs fail to open, the secondary pressure exceeds the main steam safety valve setpoints at various times for different number of tubes ruptured. The main concern of the SECY-93-087 is that once an MSSV lifts, there is a possibility of its failure to reseat, resulting in an unisolable LOCA outside containment. For this situation, the SECY paper recommended remedial actions. The Westinghouse analysis result of MSSV opening is based on the assumption of PORV failure to open in the event of an SGTR. Westinghouse should examine the probability of the PORV failure to open, especially since the PORV is a non-safety design.

Westinghouse should address what actions are necessary to conform with SECY-93-087. In addition, the staff requests justification for the use of MAAP4 in analyzing this event.