February 11, 2002

Mr. Gordon Biscoff, Project Manager
Westinghouse Owners Group
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: WESTINGHOUSE OWNERS GROUP - WCAP-15666, REV. 0, "EXTENSION OF REACTOR COOLANT PUMP MOTOR FLYWHEEL EXAMINATION"

Dear Mr. Bischoff:

By letter dated August 24, 2001, the Westinghouse Owners Group (WOG) submitted for staff review Topical Report WCAP-15666, Rev. 0, "Extension of Reactor Coolant Pump Motor Flywheel Examination." The staff has completed its preliminary review of WCAP-15666 and has identified a number of items for which additional information is needed to continue its review. On January 17, 2002, the NRC staff discussed an informal set of questions with the WOG during a conference call.

The staff is now forwarding the enclosed request for additional information (RAI). Please provide the requested information so that the review can be completed in a timely manner. Partial submittals would be welcomed to minimize delays.

Pursuant to 10 CFR 2.790, we have determined that the enclosed RAI does not contain proprietary information. However, we will delay placing the RAI in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

If you have any questions, please call me at (301) 415-1436

Sincerely,

/RA/
Drew Holland, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Request for Additional Information

cc w/encl: See next page

cc:
Mr. H. A. Sepp, Manager
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Pittsburgh, PA 15230-0355

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REQUEST FOR ADDITIONAL INFORMATION

WCAP-15666, REV. 0, "EXTENSION OF REACTOR COOLANT PUMP

MOTOR FLYWHEEL EXAMINATION"

PROJECT NO. 694

1. The parameter on the pipe size shown in Table 3-4 of the Structural Reliability Risk and Assessment (SRRA) Benchmarking Study considers a range of pipe sizes (pipe outside diameter (OD) and wall thickness) that are much smaller than the characteristic dimensions of flywheels discussed in WCAP-15666. How will the fracture mechanics model used in the SRRA code be applicable to validate the results obtained in Table 3-8 using the flywheel-specific fracture mechanics model discussed in Section 2 of the submittal? Please explain what parts of the flywheel failure probability code calculations (e.g., the PROF object library) are validated by the SRRA results by comparing the results of the SRRA code with that of the pc-PRAISE code as shown in Tables 3-3 and 3-4.

2. Were the failure probabilities of ductile and excessive deformation failure modes stated in Regulatory Guide (RG) 1.174, "An Approach for Using Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis" calculated? If not, please provide your rationale.

3. Section 3.3 states that the nominal rotation per minute (rpm) for a flywheel is 1200 rpm and discusses "peak speed[s]" of 1500 rpm or 3321 rpm that are "used in the evaluation" of failure frequency. The entries in Table 3-8 identify the failure frequencies by these peak speeds. Table 3-5 includes input parameters as "Number of Transients per Operating Cycle" and "Speed Change per Transient (rpm)."

   a. What is the relationship between the "peak speed" and the probability of failure at 40 and 60 years? For example, does the calculation in Table 3-8 for 1500 rpm assume that the flywheel runs continuously at 1500 rpm during the life of the plant? Does the calculation for 3321 rpm assume that the flywheel runs continuously at 3321 rpm during the life of the plant?

   b. If the flywheel is assumed to run at 1200 rpm and only increases speed to 1500 rpm or at 3321 rpm after an event, please describe the input frequency of each event and the development of the frequency. If an event frequency is included in the failure frequency calculation in Table 3-8 for either of the cases, 1500 rpm or 3321 rpm, how does this comport with the multiplication of the failure frequencies with initiating event frequencies in Tables 3-12 and 3-13?

4. The submittal's estimate of an initiating event frequency of 2E-6/yr for large break loss-of-coolant accidents (LOCAs) is two orders of magnitude lower than the estimates used in the individual plant examinations (IPEs). The submittal uses 2E-2 from NUREG/CR-6538 (Reference 1) for the conditional probability of getting a loss of offsite power (LOOP) following a LOCA. The same NUREG suggests that the frequency of a

large LOCA in a pressurized water reactor (PWR) is in the range of 5E-4/year to 2E-4/year. These estimates of LOCA frequency are also much higher than the submittal's estimate of 2E-6/year. Additionally, although the 3321 rpm speed is discussed as the bounding speed, Table 3-11 and the following paragraph indicate that the 2E-6/yr LOCA frequency only includes large cold leg breaks which would, on loss of the reactor coolant pump (RCP) power, result in the 3321 rpm. Therefore, although the speed may be a bounding value, the frequency is not a bounding value because the lesser equivalent break area equal to 60 percent of the double-ended break area (resulting in 2609 rpm) is not included in the risk calculation. The analysis should include consideration of the uncertainty in the LOCA frequency as part of an overall uncertainty or bounding evaluation.

5. The sensitivity study in Table 3-9 indicates that the results are quite sensitive to some input parameters when the values of the parameters are individually varied. There is no discussion on how the results could vary when the input parameters are simultaneously varied. Furthermore, Table 3-5 includes other parameters that appear to represent highly uncertain parameters, particularly "Fatigue Crack Growth Rate," that are not included in the sensitivity study. Another parameter with large uncertainties that is not included in the sensitivity analysis is the LOCA frequency. RG 1.174 requires that uncertainty be considered in any risk-informed analysis. If the calculated metric is sufficiently small, "a simple bounding analysis may suffice." Please provide an analysis that bounds the final result (change in risk) based on the potential variation in all the input parameters. Alternatively, a systematic uncertainty analysis as described in Reference 2 and illustrated in Reference 3 may be performed.

6. How is the safe shutdown earthquake load factored into the evaluation of nonductile, ductile and excessive deformation?

References

1. "Evaluation of LOCA with Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," BNL, NUREG/CR-6538, July 1997.

2. "Best Estimate Calculations of Emergency Core Cooling System Performance," U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, May 1989.

3. "Quantifying Reactor Safety Margins," U. S. Nuclear Regulatory Commission, NUREG/CR-5249, December 1989.