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

May 3, 1984

Docket Nos: 50-390
50-391

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500 A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION CONCERNING STEAM GENERATOR TUBE RUPTURE AT THE WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

Since the staff first issued the Watts Bar SER, considerable insight has been gained by the staff and industry in the analysis and mitigation of steam generator tube rupture (SGTR) events. Plant experience at Ginna and other PWRs has disclosed inconsistencies between the actual events and FSAR assumptions. In particular, the assumption that the tube leak can be terminated by the operator within 30 minutes may not be valid under loss of offsite power and single failure assumptions. Emergency guidelines have been developed which, under some instances, recommend continued steaming of the ruptured steam generator during the cooldown process.

Members of the NRC staff, Westinghouse technical staff, and several Westinghouse owners (NTOLs) met in Bethesda, Maryland, on February 23, 1984 to discuss plans for resolution of the SGTR issue. The general approach presented and schedule for complete resolution appear promising. The staff made comments which are reflected in the attached meeting minutes (Enclosure 1). In addition, the staff has prepared the attached (Enclosure 2) request for additional information.

We request that your responses to these questions be submitted by May 31, 1984. If you have any questions concerning this matter, please contact the project manager, T. J. Kenyon, at (301) 492-7266.

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A PDR

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P. L. 96-511.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
As stated

cc: w/enclosures
See next page

WATTS BAR

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ENCLOSURE (2)

STEAM GENERATOR TUBE RUPTURE QUESTIONS

Steam generator tube rupture (SGTR) events at R. E. Ginna and other PWRs indicate the need for a more detailed review of the analysis for this accident. In view of this plant experience, the staff's review of FSAR section 15.4.3 has resulted in several questions and a need for the following additional information and clarification:

1) FSAR section 15.4.3 indicates equalization of primary and secondary pressure 30 minutes after the SGTR event, with consequent termination of steam generator tube leakage. Justify that this can be achieved under loss of offsite power/natural circulation conditions under which a steam bubble might form in the reactor vessel head.

2) Clarify whether you have analyzed a case which considers the radiological effects of a SGTR with the highest worth control rod stuck out of the core, with equilibrium iodine concentration, including the effects of any additional fuel failure caused by this event.

3) Discuss whether as a result of possible modification to your analysis including consideration of longer leak times, liquid can enter the main steam lines. If so, discuss the effects on the integrity of the steam piping and supports. Consider both the liquid dead weight and the possibility of water hammer.

4) Describe the sequence of events which includes an identification of all operator actions and when these actions are expected to occur. Justify the analytical assumptions regarding operator action to (1) open steam line atmospheric relief valves and (2) open a pressurizer PORV. Also include descriptions of the automatic initiations and actuation as they occur chronologically.

5) Provide the following parameters as a function of time, until releases from the ruptured steam generator are terminated:

   (a) the primary system pressure;

   (b) intact and ruptured hot leg fluid temperature;

   (c) the secondary liquid water mass and level in each steam generator;

   (d) the primary system liquid mass;

   (e) the secondary system pressure in each steam generator;

(f) the integrated mass released out of the atmospheric relief valves or safety valves for the intact steam generators and for the ruptured steam generator;

(g) pressurizer level;

(h) the tube rupture flow rate and integrated tube rupture flow;

(i) the extent of upper head voiding if predicted, and

(j) the steam flow rate for all steam generators.

6) Describe or reference the computer code utilized to calculate the primary and secondary system response. Justify that the code is appropriate for SGTR analysis.

7) Provide the noding diagram used in the analysis. Justify that sufficient noding is provided to predict head bubble formation or loss of natural circulations in loops for which the steam and feedwater flow has been isolated.

8) Include in the analysis of the SGTR accident the most limiting single active failure. If the most limiting single active failure is failure of an atmospheric relief valve to close, operator action to close the block valve may be assumed if justified. The analysis should assume that the accident

begins with the primary coolant iodine concentrations at the technical specification limit and that an iodine spike occurs as a result of the primary system depressurization.

9) Consider the case that the loop with the ruptured steam generator is also the one to which the RHR drop line is connected. Justify that this loop can be brought to RHR cut in conditions without additional steaming of the ruptured unit or provide the additional leakage and release to the environment from the additional steaming.

10) Identify all equipment which is relied upon to mitigate a design basis SGTR event. Justify that this equipment meets NRC requirements for safety related equipment.