Docket No. 50-272

May 21, 1984

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Mr. Richard A. Uderitz, Vice President - Nuclear
Public Service Electric and Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Uderitz:

SUBJECT: ASYMMETRIC LOCA LOADS OPERATING REACTOR LICENSING ACTIONS - SUMMARY OF CONCERNS OF INDEPENDENT PLANT SAFETY EVALUATIONS

Re: Salem Unit 1

Generic Letter 84-04, dated February 1, 1984 subject "Safety Evaluation of Westinghouse Topical Reports Dealing With Eliminating of Postulated Pipe Breaks in PWR Primary Main Loops" indicated an acceptable technical basis has been provided for the 16 Westinghouse Owners Group plants so that the asymmetric blowdown loads resulting from double ended pipe breaks in the main coolant loop piping need not be considered as a design basis provided certain specified conditions are met.

However, as you are one of several independently represented nuclear plant facilities currently under review for their ability to withstand asymmetric loadings from postulated loss-of-coolant accident (LOCA), we are transmitting this request for additional information in order to complete our Safety Evaluations which are scheduled for completion in the fourth quarter of fiscal 84. Under review are the plant-specific LOCA analysis submitted by the independent utilities for the following plants: Salem Unit 1, Trojan, Beaver Valley Unit 1, Prairie Island Unit 1 and 2, Kewaunee, Maine Yankee and St. Lucie Unit 1.

The asymmetric LOCA load submittals reviewed to date are referenced at the end of the enclosed summary. All submittals were evaluated with the guidelines set forth by NUREG-0609 and the summary lists the major areas of concern that have not met these established guidelines. Detailed questions and requests have previously been submitted in the form of initial reviews and additional requests for information. It is our understanding that a substantial portion of the information which would resolve these concerns is available with the NSSS vendors.

Please respond to the concerns identified in the enclosure in time for the staff to complete the Safety Evaluation as scheduled above, i.e., by July 31, 1984.

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

/s/SVarga

Steven A. Varga, Chief
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Enclosure:
As stated

cc w/enclosure:
See next page

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A. Cavity Pressurization Analysis

1. The TMD input parameters are required, especially those concerning contraction from losses and vena contracta effects.

2. The methodology used in the calculation of the mass and energy release rates utilized in the analysis is required.

B. Thermal Hydraulics Analysis

1. The MULTIFLEX node-volume spatial schematics for each postulated break analyzed are required.

2. A list of the MULTIFLEX thermal-hydraulic input parameters is required for each MULTIFLEX analysis performed.

3. The resulting absolute and differential pressure transients across the core support barrel are required for each postulated pipe break analyzed.

C. Structural Analysis

1. The applied loads and the use of the design loads in the primary shield wall analysis require clarification.

2. THe structural analysis of the following components requires quantitative results, allowable values, and the bases for the allowables.

   a. fuel assemblies
      (1) assembly component stresses and grid impact forces
      (2) derivation and value of the minimum grid strength (P_{crit})

   b. reactor vessel internal components other than the core barrel

c. steam generator and pump supports

3. The structural analyses of the following components require allowable values for the submitted results and the associated acceptance criteria:

   a. core barrel
   b. control rod drive mechanism
   c. primary shield wall (acceptance criteria only)

4. A complete LOCA analysis adhering to the criteria outlined in NUREG-0609 is required for the ECCS piping and supports

REFERENCES

1. "Dynamic Analysis of the Reactor Coolant System for Loss-of-Coolant Accidents: Salem Nuclear Generating Stations I and II," NS-TMA-1996, Westinghouse Electric Corp., December 1, 1978. (Westinghouse Proprietary)

2. "Evaluation of the Reactor Coolant System Considering Subcompartment Pressurization Following a LOCA for Salem Units No. 1 and 2," enclosure to R. L. Mittl to O. D. Parr letter, Public Service Electric and Gas Company, March 6, 1979.

3. "Reactor Vessel Support Analysis for the Trojan Nuclear Plant," PGE-1014, Portland General Electric Company, January 1977.

4. "Analysis of the Effects of LOCA on Reactor Coolant System Supports," Beaver Valley Power Station Unit 1, January 1976.

5. "Structural Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident--Prairie Island/Kewaunee Nuclear Power Plant," Preliminary Issue, Westinghouse Electric Corp., February 1980. (Proprietary)

6. "Evaluation of Asymmetric LOCA-Related Loadings on the Reactor Coolant System at Maine Yankee," Enclosure (A) of letter B3.2.1, WMY80-75, Maine Yankee Atomic Power Company, May 14, 1980.

7. "Reactor Coolant System Asymmetric LOCA Loads Evaluation", Revision 1, Enclosure to letter L-80-263, Florida Power and Light Company, August 8, 1980.

8. St. Lucie Unit No. 1 Final Safety Analysis Report, Amendment No. 36, Docket No. 50-335, December 20, 1974.