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

October 25, 1979

Docket No.: 50-438
and 50-439

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

Dear Mr. Parris:

SUBJECT: 10 CFR 50.54 REQUEST REGARDING THE DESIGN ADEQUACY OF BABCOCK & WILCOX NUCLEAR STEAM SUPPLY SYSTEMS UTILIZING ONCE THROUGH STEAM GENERATORS (BELLEFONTE NUCLEAR PLANT)

Several hardware and procedural changes have been made to operating B&W plants to reduce the likelihood of recurrence of a TMI-type accident. These changes have been in the area of auxiliary feedwater systems, integrated control system, reactor protection system, small-break loss-of-coolant accident analysis and operator training and procedures. However, at this time, we are beginning to look more deeply into additional design features of B&W plants to consider if any further system modifications are necessary.

The use of once-through-steam-generators (OTSG) in B&W plants has an operational advantage in that it provides a small degree of steam superheat, as contrasted with the conventional saturated U-tube steam generator. In addition, it provides for less water inventory thus making a steam line break less severe. However, the relatively low water inventory with the presence of a liquid-vapor heat transfer interface in the active heat transfer zone closely couples the primary system to the steam generator conditions with a consequently high sensitivity to feedwater-flow rate perturbations. Enclosure 1 to this letter addresses system problems and staff concerns in this area. At present, we are investigating whether B&W plants are overlysensitive to feedwater transients, due to the OTSG concept, as coupled with the pressurizer sizing, ICS design, and PORV/reactor trip set points.

As part of the post TMI-2 effort, detailed analyses have been made of undercooling transients for B&W plants. However, due to the sensitivity of the OTSG design, B&W plants have also been experiencing a number of relatively severe overcooling events.

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For your information, NRC is initiating a research task to quantitatively assess B&W system designs, including the integrated control system, aimed at identifying obvious accident sequences leading to core damage having a high frequency as compared to the Reactor Safety Study, see Enclosure 2. (A complete determination of risk will not be attempted). The objective of this assessment is to identify high-risk accident sequences (including TMI implications) utilizing event tree and simplified fault tree analyses. Included will be estimation of release categories, approximate quantification of expected frequency of selected event sequences and sensitivity studies for reliability of operator response. The study will focus on the risk implications of the sensitivity of the B&W design and on the potential interactions arising from the integrated control system. We estimate this study to be completed in about six months. We will use the Crystal River, Unit 3 plant as the referenced facility to be analyzed.

We have been holding generic discussions with Babcock and Wilcox Company concerning this matter. However, system sensitivity to feedwater transients involves balance-of-plant equipment and systems as well as the nuclear steam supply system, and such plant-specific characteristics must be considered.

We are also considering whether it is necessary to halt portions of the construction of B&W plants, pending the outcome of the reliability assessment. As a preliminary consideration, we have identified those systems and components that may be impacted by possible design changes as a result of this study. Enclosure 3 is a preliminary listing of such systems and components.

Under the authority of Section 182 of the Atomic Energy Act of 1954, as amended, and Section 50.54(f) of 10 CFR Part 50, additional information is requested to allow us to determine whether it is necessary to halt all or portions of the construction of your plant pending the results of our study. We request you provide:

a) Identify the most severe overcooling events (considering both anticipated transients and accidents) which could occur at your facility. These should be the events which causes the greatest inventory shrinkage. Under the guidelines that no operator action occurs before 10 minutes, and only safety systems can be used to mitigate the event, each licensee should show that the core remains adequately cooled.

b) Identify whether action of the ECCS or RPS (or operator action) is necessary to protect the core following the most severe overcooling transient identified. If these systems are required, you should show that its design criterion for the number of actuation cycles is adequate, considering arrival rates for excessive cooling transients.

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c) Provide a schedule of completion of installation of the identified systems and components.

d) Identify the feasibility of halting installation of these systems and components as compared to the feasibility of completing installation and then effecting significant changes in these systems and components.

e) Comment on the OTSG sensitivity to feedwater transients.

f) Provide recommendations on hardware and procedural changes related to the need for and methods for damping primary system sensitivity to perturbations in the OTSG. Include details on any design adequacy studies you have done or have in progress.

We are sending similar letters to all utilities holding construction permits for plants with B&W nuclear steam supply systems.

We request your reply by December 3, 1979. We believe that a meeting with you and the other utilities together with the staff and the Babcock and Wilcox Company to discuss this matter would be beneficial to all parties. At that time, we will provide further details on the Crystal River Study. We are scheduling such a meeting for November 6, 1979 at 10:00 a.m. in Room P-422 at our offices in Bethesda, 7920 Norfolk Avenue, Bethesda, Maryland.

Please call Dr. Anthony Bournia at (301) 492-7200 if you have any questions concerning this letter.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

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Mr. H. G. Parris

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 Commerce Avenue, E11B33
Knoxville, Tennessee 37902

Mr. E. G. Beasley
Tennessee Valley Authority
400 Commerce Avenue, W10C131C
Knoxville, Tennessee 37902

Mr. D. Terrill
Licensing Engineer
Tennessee Valley Authority
400 Chestnut Street Tower - II
Chattanooga, Tennessee 37401

Mr. Dennis Renner
Babcock & Wilcox Company
P. O. Box 1260
Lynchburg, Virginia 24505

Mr. Robert B. Borsum
Babcock & Wilcox Company
Suite 420
7735 Old Georgetown Road
Bethesda, Maryland 20014

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