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52-004

January 27, 1995

Brian W. Sheron
Director Division of Engineering
Nuclear Reactor Regulation
Nuclear Regulatory Commission
Mail Stop O7E25
Washington, DC 20555

Subject: Transmittal of Requested Documentation

Reference: Kontani and Shah, Report of Thermal Properties of Concrete at High Temperature, NSF Center for Advanced Cement Based Materials, Northwestern University, (Final Report) February 17, 1994.

Dear Dr. Sheron,

This letter transmits the reference report as you requested (telecon with Jim Quinn of January 17, 1995). It should be clearly noted that the report does not form part of the SBWR design basis documentation. Further GE does not endorse the report, but we are providing it to the NRC at your request. We wish to provide you some background on this report and some cautions related to its use.

This report was prepared by the NSF Center for Advanced Cement Based Materials, Northwestern University, based on work performed between August 1992 and January 1994. The contract for this work was administered and monitored by Burns and Roe. The study was commissioned by GE Nuclear Energy to support conceptual studies of the SBWR program.

The SBWR employs a passive system for containment heat removal. Using this passive system, the containment temperature may be as high as 170°C. Previous studies of concrete exposed to high temperature indicated that, over a period of days, this might lead to increased pressure behind the liner plate due to non-condensable gases and steam released from the concrete. This raised a concern that, after the containment was depressurized, the strains in the liner plate might exceed the ASME allowable limits. In contrast, earlier plant designs use containment sprays to keep the containment temperature low. Therefore, this concern does not relate to design basis conditions in earlier designs.

One of the possible design alternatives studied to prevent damage to the liner was the use of dry (low water cement ratio) concrete for the containment structure. To obtain insight into this idea, testing was performed using low water cement ratio concrete as documented in the referenced report. Note that the test does not simulate the containment configuration or conditions, rather they duplicate a previous test program. It was observed that the water cement ratio has little effect on pressurization. The conclusion of the study

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was that there appears to be sufficient quantities of gel water, chemically bound water, and aggregate absorbed water that pressurization may occur above 105°C. Note that low water cement ratio concrete is not being pursued further for the SBWR.

It should be understood that, although the results of the test program were helpful to GE in deciding to pursue an alternate design strategy, GE does not endorse the report for design application. There are technical concerns with the report and it does not meet GE quality standards. The technical concerns include the following:

1. There is considerable discussion of the uncertainties related to instrumentation and calibration problems which were encountered during the study, but the main report does not clearly indicate whether these issues were resolved for the final testing, nor does it indicate any impact they may have had on the results.

2. The bulk of the report documents theoretical study of concrete pressurization under high temperature conditions. Although a number of hypotheses are made regarding the differences between the theoretical and experimental results, there is little effort to determine if the hypotheses could explain the differences, nor is a path for the resolution of the issues discussed. It is simply stated that the problem is difficult and the cause for the differences cannot be determined.

Since there is no suggested path to resolution of the differences between the theoretical and experimental results, and since the test conditions are not typical of conditions in either operating reactors or reactors now under design, GE does not feel that the report, in its current form, is useful for assessing design basis or severe accident conditions in a nuclear power plant. Although we would not normally release this report without resolving these technical concerns, it is being given to the NRC at their request and with these stated reservations.

Very Truly Yours,

Robert C. Mitchell

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Manager, Safety Evaluations

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