TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

JUL 29 1991

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - 10 CFR 50.46 PEAK CLAD TEMPERATURE (PCT) STATUS REPORT

Reference: TVA letter to NRC dated February 27, 1991, "10 CFR 50.46 Annual Report"

In accordance with 10 CFR 50.46 paragraph (a)(3)(ii), the following information is being provided for PCT changes that have occurred since those reported in the above reference.

On June 27, 1991, TVA received Westinghouse Electric Corporation letter TVA-91-181, which documented the resolution of several issues related to the emergency core cooling system evaluation model used to determine PCT for Westinghouse-modeled plants. The Westinghouse letter also identified which of the potential issues were applicable to Sequoyah and whether the issue had been resolved. The issues, their resolution, and the associated PCT impacts are presented in the enclosure.

A summation of the PCT impacts for the large break loss of coolant accident (LOCA) analysis results in an increase of 30 degrees Fahrenheit (F), leaving the PCT at 2,043 degrees F. For the small break LOCA PCT, a summation of those issues that result in an increase in PCT indicates a change of 37 degrees F, leaving the PCT at 2,142 degrees F. However, Westinghouse indicates that resolution of the NOTRUMP convergence reliability issue may result in a decrease in the calculated PCT, which exceeds 50 degrees F for some plants. While a plant-specific analysis has not been performed, Westinghouse recommends reporting the NOTRUMP convergence reliability issue since the NOTRUMP evaluation model has already been revised.

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The range of PCT impact for the NOTRUMP convergence reliability issue, coupled with the quantified impact of 37 degrees F, results in an accumulation of changes and errors such that the sum of the absolute magnitudes of respective temperature changes is greater than 50 degrees F. Thus, this report also serves to conservatively report these changes within the 30-day timeframe required by 10 CFR 50.46 for significant changes.

The above-identified changes do not result in exceeding the limits of 10 CFR 50.46, and PCT margin allocations will ensure these limitations are not exceeded. Therefore, further reanalysis or actions are not planned at this time.

Additionally, it should be noted that other potential issues are still under investigation by Westinghouse, which may impact the PCT for both large and small break LOCAs. The potential issues have had PCT margin temporarily allocated to ensure the cumulative effects are tracked such that the 10 CFR 50.46 PCT limit of 2,200 degrees F is not exceeded. Upon their resolution, these issues will continue to be reported as appropriate.

Please direct questions concerning this issue to J. D. Smith at (615) 843-6672.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Mark J. Burzynski for

E. G. Wallace, Manager
Nuclear Licensing and
Regulatory Affairs

Enclosure
cc: See page 3

MAC:JDS:PMB

cc (Enclosure):

Ms. S. C. Black, Deputy Director
Project Directorate II-4
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Mr. D. E. LaBarge, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

Mr. B. A. Wilson, Project Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE

PCT Issues Summary

Fuel Rod Model Revisions

It was found that the large and small break LOCA code versions were not consistent with fuel design codes in the following areas:

1. The LOCA codes were not consistent with the fuel rod design code relative to the flux depression factors at higher fuel enrichment.

2. The LOCA codes were not consistent with the fuel rod design code relative to the fuel rod gap gas conductivities and pellet surface roughness models.

3. The coding of the pellet/clad contact resistance model required revision.

In addition, it was determined that integration of the cladding strain rate equation used in the large break LOCA Evaluation Model was being calculated twice each time step instead of once. The coding was corrected to properly integrate the strain rate equation.

PCT Impact: Large Break: 10^{o}F Small Break: 37^{o}F

NOTRUMP Code Solution Convergence

In the development of the NOTRUMP small break LOCA ECCS Evaluation Model, a number of noding sensitivity studies were performed to demonstrate acceptable solution convergence as required by Appendix K to 10 CFR 50. However, since the initial studies, modifications were made to the NOTRUMP computer program to enhance code performance and implement necessary modifications. Subsequent to the modifications, solution convergence was not re-confirmed.

Sensitivity studies were performed for the time step size selection criteria which culminated in a revision to the recommended time step selection criteria inputs. Fixed input values originally recommended for the steady state and all break transient calculations were modified to assure converged results. The NOTRUMP code was re-verified against the SUT-08 Semiscale experiment and it was confirmed that the code adequately predicts the key small break phenomena.

PCT Impact: Not Quantified (Small Break Only)

Steam Generator Flow Area

Analyses performed by Westinghouse combined the most severe LOCA loads with the plant specific SSE. This was in support of the requirement that licensees are normally required to provide assurance that there exists only an extremely low probability of abnormal leakage or gross rupture of any part the reactor coolant pressure boundary. Generally, these analyses showed that while tube integrity was maintained, the combined loads led to some tube deformation. This deformation reduces the flow area through the steam generator. The reduced flow area increases the resistance through the steam generator and thus, the flow of steam from the core during a LOCA, which potentially could increase the calculated PCT.

The combination of LOCA and SSE loads led to the following:

1. LOCA and SSE loads cause the steam generator tube bundle to vibrate.

2. The tube support plates may be deformed as a result of lateral loads at the wedge supports at the periphery of the plate. The tube support plate deformation may cause tube deformation.

3. During a postulated large LOCA, the primary side depressurizes to containment pressure. Applying the resulting pressure differential to the deformed tubes cause some of these tubes to collapse, and reduces the effective flow area through the steam generator.

4. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the calculated PCT.

The effect of potential steam generator area reduction on the cold leg break LOCA PCT has been either analyzed or estimated for each Westinghouse plant. A value of 5% area reduction has been applied, unless a detailed non-linear analysis is available. The effect of tube deformation and/or collapse can be taken into account by allocating the appropriate PCT margin, or by representing the area reduction by assuming additional tube plugging in the analysis. SQN has chosen to allocate PCT margin.

PCT Impact: 20°F (Large Break Only)