Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

September 26, 1995

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

Gentlemen:

In the Matter of	)	Docket No. 50-327
Tennessee Valley Authority	)	

SEQUOYAH NUCLEAR PLANT (SQN) - ADDITIONAL INFORMATION FOR TECHNICAL SPECIFICATION (TS) CHANGE 95-15, REVISION 1

Reference: TVA letter to NRC dated September 7, 1995, "Sequoyah Nuclear Plant (SQN) - Technical Specification (TS) Change 95-15, Revision 1, 'Alternate Plugging Criteria for Steam Generator (S/G) Tubing - Unit 1'"

Enclosed is additional information for TS Change 95-15, Revision 1 that was discussed with the NRC staff during a September 19, 1995, telephone call. The additional information provides the new assumed initial value for secondary coolant iodine activity in the accident analysis associated with the subject TS change. The new value (0.1microcuries per gram dose equivalent Iodine-131) replaces the current value ($2x10^{-6}$ microcuries per gram dose equivalent Iodine-131) that was assumed in Westinghouse Electric Corporation WCAP-13990. Accordingly, with the new assumed initial secondary side activity, the allowed S/G leakage rate from the faulted loop has been revised from 4.3 gallons per minute (gpm) to 3.7 gpm.

This change was evaluated by TVA and Westinghouse and is discussed in Enclosure 1. Enclosure 2 provides a revised TS Bases page (Insert I) that contains the new 3.7 gpm value. The new Insert I information supersedes the Insert I previously provided in Enclosure 1 of the referenced letter.

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Please direct questions concerning this issue to D. V. Goodin at extension 7734.

Sincerely,

R. H. Shell

R. H. Shell
Manager
SQN Site Licensing

Enclosures
cc (Enclosures):

Mr. D. E. LaBarge, Project Manager
Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
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NRC Resident Inspector
Sequoyah Nuclear Plant
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Soddy-Daisy, Tennessee 37379-3624

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

ENCLOSURE 1

STEAM GENERATOR ALLOWABLE

LEAK RATE RECALCULATION

TVA-95-195
September 22, 1995

Mr. Mark Burzynski, Manager
Department of Nuclear Engineering
Tennessee Valley Authority
Sequoyah Nuclear Plant
P.O Box 2000
Soddy Daisy, TN. 37379

Tennessee Valley Authority
Sequoyah Nuclear Plant
S/G Allowable Leak Rate Recalculation

Dear Mr. Burzynski:

The following provides the documentation for completion of task N95-024.
An evaluation has been performed to determine the maximum permissible steam generator primary to secondary leak rate during a steam line break for Sequoyah Nuclear Plant.Unit 1 and 2.This evaluation is identical to the evaluation presented in WCAP-13990,"Sequoyah Units 1 and 2 Steam Generator Tube Plugging Criteria for Indications at Tube Supports Plates" with the exception of the initial secondary coolant iodine activity. The WCAP used a calculated concentration of 2 E-06 micro curie/gram of dose equivalent I-131. This evaluation uses a value of 0.1 micro curie/gram of dose equivalent I-131, to be consistent with the Standard Westinghouse Technical Specifications.

The result of this change is to slightly reduce the allowable leak rate . The previous leak rate was 4.6 gpm total or 4.3 gpm to SG on the faulted loop. The new leak rate is 4 gpm total or 3.7 gpm to the faulted loop.

If you have questions, please contact the undersigned.

Very truly yours,
Kelli Agostino for.
D. W. Salak
Sequoyah Projects Manager
TVA Projects

cc: D.Lafever
H.Cothron

ENCLOSURE 2

REVISED INSERT I

FOR

TECHNICAL SPECIFICATION

CHANGE 95-15, REVISION 1

Insert I

The total steam generator tube leakage limit of 600 gallons per day for all steam generators and 150 gallons per day for any one steam generator will minimize the potential for a significant leakage event during steam line break. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 3.7 gpm in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm.