TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

May 30, 1984

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

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

The Sequoyah Nuclear Plant unit 1 and unit 2 operating license conditions 2.C(23).j(2) and 2.C(16).i, respectively, require TVA to "conform to the EPRI test program" and to "provide documentation for qualifying (a) reactor coolant system relief and safety valves, (b) piping and supports, and (c) block valves in accordance with the review schedule in SECY 81-491. . . ." Also, NUREG-0737 item II.D.1, as revised by D. G. Eisenhut's September 29, 1981 letter, required documentation to be submitted for items (a), (b), and (c) above. A response for item (a) was submitted by letter from me to you on April 1, 1982. A response for item (b), piping and supports, and item (c) block valves, which satisfied the requirements of the license conditions and NUREG-0737 was submitted by my June 30, 1982 letter. A supplemental response was submitted by my January 7, 1983 letter to you.

As a result of telephone conversations on May 9 and 10, 1984 with members of the NRC staff, enclosed is a revision to section 4.0 of the supplemental response submitted to you on January 7, 1983.

If you have any questions concerning this matter, please get in touch with Jerry Wills at FTS 858-2683.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 30th day of May 1984

Paulette H. White

Notary Public
My Commission Expires 9-5-84

Enclosure

cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attn: Mr. James P. O'Reilly Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

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ENCLOSURE

REVISION TO SECTION 4.0 OF SUPPLEMENTAL RESPONSE
SUBMITTED TO NRC ON JANUARY 7, 1983

SEQUOYAH NUCLEAR PLANT

4.0 Piping/Support Evaluation

As indicated in item 3.0 of our June 30, 1982 response, a further evaluation of safety/relief valve discharge piping support loads was necessary, and if design modifications were required, a schedule for implementation would be provided.

Through the application of RELAP 4/MOD 5 and conservative assumption on the impact of the water slug, evaluations to date indicate that changes to the support loads, because of subcooled water slug flow, will probably involve several major support modifications. To eliminate this concern, TVA decided to delete the power-operated relief valve loop seals and drain the safety valve loop seals. The Crosby safety valves were modified by installing steam internals on unit 1 and scheduled for modification in the next unit 2 refueling outage. Subsequent to this modification on SQN unit 1, high tailpipe temperatures were observed on the safety valve discharge piping and a safety valve actuation may have occurred while at normal operating pressure. Following numerous valve replacements, TVA decided, in accordance with 10 CFR 50.59, to reestablish the loop seals on the safety valves. It is TVA's engineering judgement (based on the support loads and safety factors) that the system will function to mitigate an overpressure transient for interim operation with the loop seals established.

TVA plans to reevaluate the options available to resolve this issue. These options include, but are not limited to, the following:

1. Evaluation of the problems associated with draining of the loop seals and continued operation with the leaking safety valves.

2. Modification of the supports on the safety valve discharge piping.

3. Heating the safety valve loop seal.

4. Modifying the safety valve discharge piping to reduce support loads.

Qualification of the piping/support system will be verified by the thermal-hydraulic code RELAP 5 MOD 1. TVA will implement modifications to resolve the safety valve discharge piping support loads by the end of the third refueling outage on unit 1 and the second refueling outage of unit 2. Because of the approaching unit 2 cycle 2 refueling outage and depending on the complexity of the analysis and modifications, the unit 2 modifications may slip to no later than the end of the third refueling outage.