TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

February 15, 1985

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 the Application of) Docket Nos. 50-390
Tennessee Valley Authority 50-391

Please refer to TVA's letter dated February 8, 1985 which provided proposed revisions to technical specification 3/4.4.6.2 regarding Reactor Coolant System pressure isolation valve leakage criteria.

Due to oversight, the first page of the enclosed Technical Basis discussion was inadvertently omitted. Enclosed is the missing page. NRC representative K. Jabbour was notified of this matter on February 14, 1985.

If you have any questions concerning this matter, please get in touch with D. B. Ellis at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. W. Hufham

J. W. Hufham, Manager
Licensing and Regulations

Sworn to and subscribed before me
this 15th day of Feb. 1985.

Paulette H. White

Notary Public

My Commission Expires 8-24-88

Enclosure

cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attn: Mr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

B001
1/1

ENCLOSURE

TECHNICAL BASIS

These changes to LCO 3.4.6.2.f, SR 4.4.6.2.2, and table 3.4-1 are proposed in order to preclude undue down time during plant recovery f om cold shutdown conditions without adversely affecting the safety of the public (i.e. safe operation of the reactor). The proposed changes are consistent with the approach to primary pressure boundary leakage quantities currently under review by the NCR staff for possible inclusion in the Standard Tech Spec and are similar to those accepted by the staff in Amendment No. 50 to Facility Operating License No. NPF-2 and Amendment No. 41 to Facility Operating License NPF-8 for the Joseph M. Farely Nuclear Plant Units Nos. 1 and 2. (Attachment 1)

The 1 gpm limit currently required by the Watts Bar Unit 1 draft Technical Specification is more or less an arbitrary criterion which was imposed on all nuclear plants following the TMI-2 accident. It was based on a very conservative estimate of the pressure relief capacity for the plant and is not an indicator of imminent accelerated deteriation or potential valve failure. Since July, 1984 at Sequoyah Nuclear Plant there have been two failures of particular interest in check valve leakage measurements. Two valves of similar construction failed to meet the 1 gpm criteria. One valve failed at 2 gpm and the other at 6 gpm. Under the proposed criteria, the 2 gpm leak rate would have been acceptable; the 6 gpm leak rate would have been unacceptable. When both valves were disassembled, no cause for the leakage could be detected and no signs of imminent failure were found. Both valves were reassembled and tested successfully. The inability to find significant valve degradation upon failure of the 1 gpm limit was also reported by Joseph M. Farley Plant in support of the above mentioned amendments.