

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 4, 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 the Application of) Docket Nos. 50-390
Tennessee Valley Authority) 50-391

Please refer to TVA's letters dated September 14, 1981 and June 11, 1982, which provided responses to NUREG-0737 item III.D.1.1, "Integrity of Systems Outside Containment" for the Watts Bar Nuclear Plant. Also, refer to TVA's letter dated July 20, 1984 which provided the results of the leak test performed on the Waste Gas System and the procedure governing that test.

At the NRC's request, conference calls were held between TVA and NRC representatives on August 22 and 29, 1984 to discuss the leak test results (15.5 standard cubic feet per hour (std ft³/hr) at a system pressure to 100 psig) transmitted by the referenced letter of July 20, 1984.

NRC representatives involved in the conference calls indicated that efforts had been undertaken to determine allowable leak rates to assure compliance with General Design Criterion 19 of 10 CFR 50, Appendix A, and that by using an internal NRC procedure based upon Three Mile Island release data, a leak rate of approximately 2 x 10⁻⁸ std ft³/minute on a per leak basis had been calculated. The underlying concern is over whether there is a potential, due to fuel failure, for a significant release of radioactive gas outside containment through leaks in the Waste Gas System.

TVA representatives involved in the conference calls indicated that in accordance with the criteria of NUREG-0737 item III.D.1.1, the identified leakage of 15.5 std ft³/hr represents the as-low-as practical level for the Waste Gas System. TVA further indicated that the Waste Gas System would not be used to process highly radioactive gases during an accident and therefore should be excluded from the NUREG-0737 leak reduction program. The August 29, 1984 conference call concluded with TVA committing to formally submit its justification for excluding the Waste Gas System from the leak reduction program. Enclosed is TVA's revised response to NUREG-0737 item III.D.1.1.

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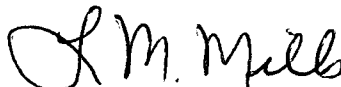
Director of Nuclear Reactor Regulation

October 4, 1984

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



L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 4th day of Oct. 1984

Bryant M. Lowery
Notary Public
My Commission Expires 4/8/86

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 30323

ENCLOSURE

WATTS BAR NUCLEAR PLANT
NUREG-0737 ITEM III.D.1.1

"INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT"

REVISED RESPONSE

INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO
CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED WATER REACTORS

TVA RESPONSE (Revised October 4, 1984)

The Watts Bar Nuclear Plant (WBN) design was reviewed to evaluate ways to minimize radioactive fluid leakage. Plant systems that were reviewed included Residual Heat Removal (RHR), Containment Spray, Safety Injection (recirculation mode), Chemical and Volume Control (CVC), Sampling, and Waste Gas. The examination included valve stem packing leakoffs, rotating seals, gasket connections, vents, and drains.

As a result of the review, a second pressure boundary has been incorporated on about 20 vents and drains found on pump suction lines and pump casings. The second pressure boundary is a second valve in most cases and an occasional blind flange.

An additional review was conducted with regard to the North Anna 1 incident and no similar release path was found. The WBN design routes the over-pressure relief from the volume control tank to the pressurizer relief tank and all relief paths from high pressure systems vent back into containment to the pressurizer relief tank. All tanks containing radioactivity in the Radwaste System and the CVCS vent to a contained release path which is continuously monitored.

A periodic leak check program will be implemented for the following systems which have been identified as requiring leakage checks.

1. RHR
2. Containment Spray
3. Safety Injection
4. CVC
5. Sampling

Procedures for reducing and quantifying leakage from liquid systems were provided by TVA's letter dated July 19, 1982. These procedures were written in compliance with the guidelines listed below.

1. Visual inspection with the system in operation is required.
2. Closed loop systems, such as Component Cooling Water, will not be inspected.
3. Inspection will be performed annually.
4. Leakage will be quantified and specifically located by valve number, pump flange or other similar means.
5. Leakages will require immediate attention. All leakage identified will be "tracked" inplant until the leakage is stopped or controlled (i.e., normal pump seal leakage per manufacturer's spec).
6. Initial leak test results will be provided to the NRC.

INTEGRITY SYSTEMS OUTSIDE CONTAINMENT RELY TO
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The initial test results of the leak reduction program for the identified liquid systems will be provided by fuel loading of unit 1.

The Waste Gas System will be excluded from the leak test program since the system would not be used to process highly radioactive gases during an accident event. A more detailed justification for its exclusion is provided in Attachment 1.

Containment isolation will result in all but two of the Condition 3 and 4 events discussed in Chapter 15, "Accident Analysis" of the WBN Final Safety Analysis Report (FSAR). The two events for which isolation does not occur are: (1) Inadvertent Loading of a Fuel Assembly Into an Improper Position (FSAR 15.3.3) and (2) Single Rod Cluster Control Assembly Withdrawal at Full Power (FSAR) 15.3.6).

Inadvertent loading of a fuel assembly into an improper position is unlikely due to WBN administrative procedures. In the event that it did occur, the resulting power distribution effects will either be readily detected by the incore moveable detector system or will cause a sufficiently small disruption to be acceptable within the uncertainties allowed between nominal and design power shapes.

In the event of an inadvertent rod cluster control assembly withdrawal, it would require a double failure of electrical or mechanical systems. The reactor would eventually trip on overtemperature ΔT with an upper limit on the number of rods with a departure from nucleate boiling ratio (DNBR) less than 1.3 being about 5 percent. This does not necessarily mean that 5 percent of the fuel will fail and since operation is allowed with about a 1-percent fuel failure, TVA does not believe a significant amount of fission gases will be released before the reactor will trip. -

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