TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

August 5, 1983

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

Enclosed for NRC review is TVA's final response to NRC question 212.35.

This information will be incorporated into the Final Safety Analysis Report by future amendments.

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

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 5th day of Aug. 1983

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, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

8001
1/1

ENCLOSURE

WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2

Response to NRC Question 212.35

212.35 Question
feedwater
(6.3)

During long-term cooling following a steamline break, feedwater line break, or small LOCA, the operator must control primary system pressure to preclude overpressurizing the pressure vessel after it has been cooled off.

a. Describe the instructions given the operator to perform long-term cooling.

b. Indicate and justify the time frame for performing the required action.

c. List the instrumentation and components needed to perform this action and confirm that these components meet safety grade standards.

d. Discuss the safety concerns during this period and the design margins available. This should include potential adverse hydraulic conditions leading to inadequate cooling or mechanical damage.

e. Provide temperature, pressure, and RCS inventory graphs that would show the important features during this period.

The above discussion should account for the following:

a. loss of offsite power

b. operator error or single failure

c. small LOCA's may occur in the cold leg or in the hot leg/pressurizer.

d. small LOCA's may result in nitrogen blanketing of the steam generators.

e. long-term cooling for a small LOCA may depend on alternating forced convection and vaporization depending on the break location and size.

Response

Emergency operating procedures for Watts Bar units 1 and 2 include consideration of safety concerns for the vessel during those specified events that are categorized as pressurized thermal shock (PTS) events. The PTS issue is being addressed as a broad based safety concern by the NRC and report SECY-82-465 reviews current position. The developments to date, which are based on generic evaluations, are sufficient to show that there is no near-term safety concern related to PTS for either Watts Bar unit 1 or 2 reactor vessels.

The current methodology proposed by the NRC includes evaluation of plants based on a fracture-related parameter known as RT_{NDT}. The use of RT_{NDT} is relevant since it represents a reactor vessel material's susceptability to brittle fracture at any time in the vessel's life. As a vessel ages, the RT_{NDT} for the vessel increases due to the effect of irradiation. The lower a vessel's RT_{NDT}, the higher its resistance to brittle fracture due to pressurized thermal shock. SECY-82-465 also describes the methodology used to evaluate this effect. As a consequence of these recent developments, the NRC has proposed screening criteria limits based on RT_{NDT}. The intent of the screening criteria is to enable identification of lead plants that are plants for which RT_{NDT} approaches the limit within the next three years, so that these plants can address the issue in order to mitigate the effects through actions such as plant and procedural modifications. For other reactor vessels, no immediate safety concern exists and current design and procedures are sufficient.

The RT_{NDT} screening criterion proposed by the NRC for the longitudinally oriented flaw is the criterion relevant to the Watts Bar units 1 and 2. The value of RT_{NDT} criterion is 270°F. The evaluation should be made for the material at the reactor vessel beltline which has the most susceptability to increasing RT_{NDT}. The material considered for the Watts Bar units 1 and 2 is the vessel forged ring at the reactor vessel beltline shell course.

For Watts Bar unit 1 based on methods described in SECY-82-465, the RT_{NDT} predicted for three years of operation is 173°F RT_{NDT}, which is considerably below the screening criterion limit. In addition, the End of Life RT_{NDT}, based on the currently projected fluence rate is also below the screening criterion at 242°F.

For the Watts Bar unit 2 reactor vessel, the material properties of the beltline forging are again used, the

RT_{NDT} predicted for three years of operation is 80°F and End of Life RT_{NDT} is predicted to be 96°F. Both of these values are considerably below the screening criterion limit as well.

Through the use of the RT_{NDT} screening criteria and comparison of calculated values of RT_{NDT} for Watts Bar units 1 and 2 reactor vessels, which establishes that these units do not approach the screening criteria limits, it is concluded that there is no near-term safety concern for the reactor pressure vessels based on brittle fracture due to pressurized thermal shock transients.

Current emergency operating procedures at Watts Bar units 1 and 2 are based on the generic W Emergency Operating Instructions (EOI) (Ref. 1-2). These guidelines provide the necessary operator actions to preclude pressurized thermal shock of the reactor vessel for a small LOCA or secondary system break transient.

A revised set of generic guidelines designated the BASIC version of the Emergency Response Guidelines (ERG) (Ref. 3-7) has recently been completed and issued under Westinghouse Owner's Group sponsorship. These guidelines provide the required operator actions, a basis for these actions, and contingency actions to be taken should an expected response not be obtained at any step in the recovery procedure. The operator actions required by the ERG's to respond to a small LOCA or a secondary system break are virtually the same as those of the existing EOI's. However, included in the set of guidelines is a system of accident management based on critical safety function monitoring designed to direct operator response to any challenge to a plant safety function independent of initiating event. The vessel overcooling concern is monitored separately as the Integrity Critical Safety Function, but operator action is based on a prioritization of all critical safety functions to insure proper action to respond to core cooling, subcriticality, heat sink, or containment concerns as well as vessel overcooling.

The operator actions prescribed if a vessel overcooling condition exists and is of highest safety priority are listed in two Function Restoration Guidelines. They are 'Response to Imminent Pressurized Thermal Shock Condition' and 'Response to Anticipated Pressurized Thermal Shock Condition.' These guidelines contain appropriate operator actions to respond to the situation and also provide instructions on restrictions for subsequent cooldown to a shutdown condition.

The BASIC version of the ERGs has undergone an extensive step by step review specifically related to vessel integrity (Ref. 8). This review explains in detail the thermal shock effect of each step in the ERGs and also summarizes the important plant equipment and instrumentation used to minimize the potential for thermal shock. Use of equipment is not limited to safety grade equipment but includes all available resources for responding to the overcooling event.

A material evaluation of the Watts Bar reactor vessels has concluded that based on current methodology there is no near-term safety concern for brittle fracture due to pressurized thermal shock transients. A schedule for implementation of upgraded emergency procedures based on the ERGs has been submitted as part of the April 15, 1983 requirement for NUREG-0737 Supplement 1 items. This schedule will result in the timely implementation of emergency procedures based on the ERGs at Watts Bar well before any potential brittle fracture concern due to overcooling is reached for the Watts Bar reactor vessels.

<u>References</u>

1. 'Reference Emergency Operating Instructions, Revision 2, April 1980,' (OG-37, July 15, 1980). Correspondence to D. F. Ross (U.S. NRC) from C. Reed (WOG).

2. 'Inadequate Core Cooling Guidelines E^{2}OI-1, E^{2}OI-2 (OG-44, November 10, 1980). Correspondence to D. F. Ross (U.S. NRC) from R. Jurgeson (WOG).

3. 'Emergency Response Guideline Program (Volumes 1-111 of ERG set),' (OG-64, November 30, 1981). Correspondence to D. G. Eisenhut (U.S. NRC) from R. Jurgeson (WOG).

4. 'Transmittal of Low-Pressure Version of Emergency Response Guidelines (Volumes 1, 11A and 11B of ERG set),' (OG-76, July 21, 1982). Correspondence to D. G. Eisenhut (U.S. NRC) from O. G. Kingsley (WOG).

5. 'Transmittal of Volume III for the High-Pressure Version of Emergency Response Guidelines,' (OG-83, January 4, 1983). Correspondence to D. G. Eisenhut (U.S. NRC) from O. G. Kingsley (WOG).

6. 'Transmittal of Additional Material for the Low-Pressure Version of Emergency Response Guidelines (Additional Material for Volumes 11A and 11B Plus a Complete Volume III of LP ERG Set),' (OG-84, January 4, 1983). Correspondence to D. G. Eisenhut (U.S. NRC) from O. G. Kingsley (WOG).

7. 'Transmittal of Emergency Contingency Action Guidelines (ECA-4, ECA-5, and ES-2.3),' (OG-85, January 13, 1983). Correspondence to D. G. Eisenhut from O. G. Kingsley (WOG).

8. 'PTS Review of ERGs,' (OG-72, June 22, 1982). Correspondence to H. R. Denton (U.S. NRC) from O. G. Kingsley (WOG).