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

March 11, 1982

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

Dear Ms. Adensam:

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

My letters to you dated September 14, 1981 (original submittal) and October 29, 1981 (revised responses to certain items) provided TVA's responses to NRC concerns specified in NUREG-0737 for Watts Bar Nuclear Plant. Enclosed are revised responses to the following items: I.A.1.2, II.B.2, II.D.1, II.D.3, II.F.2, II.G.1, II.K.3.5, II.K.3.30, and III.A.1.1.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 11th day of March 1982

Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

Enclosure
cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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TVA Response (Revised March 9, 1982)

The requirements are to be implemented by fuel loading for Watts Bar units 1 and 2.

1. The duties of the shift supervisor, as discussed in NUREG-0578, are performed by the assistant shift engineer. The V. P. for Operations is the Manager of Power Operations. TVA's administrative procedures, shift supervisor job descriptions, and training programs emphasize the primary management responsibility of the shift engineer. In addition, periodic retraining acts to reinforce his command responsibilities. While these existing measures provide a high level of confidence that the shift supervisor has primary management responsibility for safe operation of the plant, TVA will periodically issue a management directive which emphasizes this assignment of responsibility.

2a. Plant administrative procedures have been reviewed to ensure that they clearly define the authority and responsibilities of each position on shift. The duties and responsibilities of the shift supervisor, as specified in the job description, are consistent with position statement 2a. Administrative instruction, the shift supervisor's (assistant shift engineer's) responsibilities, and the Watts Bar standard practices show TVA's current training program.

2b. The shift crew in TVA plants consists of the following: (1) a shift engineer who has an SRO license and who has overall responsibility for the plant when higher level 'in-line' management employees are not present, (2) an assistant shift engineer (also has an SRO license) for each unit who has supervisory responsibility for all normal, abnormal, and emergency activities on his assigned unit, (3) a unit operator (with an RO license) for each unit, and (4) other employees as appropriate. The duties of the shift supervisor as discussed in NUREG-0578 and -0737 are performed by the assistant shift engineer on each unit. For purposes of our responses, we will use the term assistant shift engineer for shift supervisor.

The assistant shift engineer's normal work station is in the control room, but he periodically makes inspections of plant equipment. He will immediately go to the control room during emergency situations.

He remains in the control room at all times during accident situations to direct the activities of the unit operator unless formally relieved of this function by the shift engineer. The shift engineer may, in turn, be formally relieved by the assistant operations supervisor or the operations supervisor (both also hold an SRO license).

2c. In the event that the assistant shift engineer is absent from the control room, the shift engineer or another SRO will be required to be in the control room if the unit is in modes 1, 2, 3, or 4. For multiple-unit plants, an additional licensed operator will be available in the control complex to act as an assistant to the unit operator in abnormal or emergency situations. The line of command is clearly specified in administrative procedures.

3. The shift engineer and assistant shift engineers will receive such training.

4. The administrative duties of the shift supervisor have been reviewed by the senior officer of TVA responsible for plant operations. Administrative functions that detract from or are subordinate to ensuring safe operation of the plant will be assigned to other employees. The following actions have already been taken:

    1. A clerk has been assigned to the shift engineer's office on each shift to perform administrative details formerly done by the shift engineer.

    2. Part of the routine 'non-management' duties of the assistant shift engineer have been assigned to other employees.

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS

<u>TVA RESPONSE</u> (Revised March 9, 1982)

The Watts Bar design bases include the assumption of TID 14844 sources. TVA plants are specifically designed to mitigate major design basis events with no access outside the main control room (MCR) being required. With this goal in mind, the plants were not specifically designed for any access outside the MCR. To specifically design for guaranteed access at any time in most parts of the auxiliary building is not feasible. However, the current designs allow considerable capability for access for short times if the entry time into the area can be selectively chosen.

The current arrangements and shielding for normal operation will help minimize the impact from post-accident contained sources even though the shielding was not intended for that purpose. In certain instances, TVA has provided some shielding for post-accident access. TVA has performed a shielding review for Watts Bar. The review included generation of radiation source terms for primary system water and containment sump water based on TID 14844. These fluids were assumed to circulate in the plant systems designed for accident response and also in systems used in normal plant operation but which might be called upon for accident recovery. From the analyses performed, radiation doses can be determined at locations in the plant near accident recovery equipment.

Watts Bar is designed to mitigate major accidents without access to the plant outside the MCR. One area outside the MCR was identified which would be helpful in responding to an accident situation. This area is the normal plant sampling station in the auxiliary building at elevation 713.0. Dose rates in the sample room were evaluated for various times into the accident. a representative value at one hour into the accident is 900 mr/minute. Sampling procedures for accident situations in the interim period until a redesigned sampling facility can be installed take into account these calculated values. If samples are ever needed in an accident, the procedure will also utilize actual dose rate measurements to evaluate accessibility and occupancy times.

As a result of this study, it has been determined that no additional shielding is necessary at Watts Bar, except for lead blankets around sample lines in the sample room to improve its accessibility in an accident situation.

A detailed report on TVA's shielding review for Sequoyah was transmitted to A. Schwencer by L. M. Mills letter dated June 16, 1980. This information is also applicable to Watts Bar.

Watts Bar Nuclear Plant will meet the requirements of GDC 19.

PERFORMANCE TESTING OF PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES

<u>TVA RESPONSE</u> (Revised March 9, 1982)

TVA is a participant in the EPRI PWR Relief and Safety Valve Test Program. Recently, NRC, by a letter dated September 29, 1981, from Darrell G. Eisenhut to all Operating License/Construction Permit Holders, revised the schedules for completion of this program and submittal of all generic and plant specific information for Staff review. It is TVA's intent to provide this information in a timely manner to NRC before unit 1 fuel loading. However, our ability to complete the necessary plant specific evaluations required by NRC is dependent upon the preparation and submittal of the final valve test reports to the utilities by EPRI.

TVA RESPONSE (Revised March 9, 1982)

The power operated relief valves have a reliable direct, stem-mounted position indication in the main control room. Valve position of the pressurizer safety valves is currently provided in the following manner.

1. Temperature is sensed downstream of the valves and displayed in the main control room including high temperature alarms.

2. The pressurizer relief tank has temperature, pressure, and fluid level indication and alarms in the main control room.

3. The pressurizer has high pressure alarms in the main control room.

An acoustic monitoring system for the three safety relief valves and Power-operated Relief Valves has been provided on unit 1 and will be provided on unit 2 before fuel loading. An accelerometer is mounted on the valve discharge line just downstream of each valve. The accelerometer signals go to a charge converter inside containment mounted in a NEMA-4 enclosure. A valve flow indicator module is located in the main control room. The flow indicator module will be calibrated to detect indication of failure of a valve to reclose. An alarm in the main control room will indicate when any valve is not in the fully closed position.

This design provides the operator with unambiguous indication of valve position as specified in the above response.

Valve position is indicated in the main control room and alarmed as discussed in the above response.

Valve position indication for Watts Bar Nuclear Plant meets seismic and environmental qualification requirements as specified for Sequoyah. Technology for Energy Corporation (TEC), the vendor for the monitoring system is currently conducting a qualified life test program.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

TVA RESPONSE (Revised March 9, 1982)

Analysis and procedures for the detection of inadequate core cooling using existing instrumentation have been developed in conjunction with the Westinghouse Owners Group. This guidance will be incorporated into plant procedures by fuel load.

In addition to the primary method for detecting inadequate core cooling described above, TVA will provide instrumentation to measure water level from the top to the bottom of the reactor vessel. Refer to Figure II.F.2-1. This instrumentation will be designed and qualified in accordance with safety grade, Class IE, requirements including redundancy and emergency power.

The Reactor Vessel Level Instrumentation System was designed by Westinghouse to provide direct readings of vessel level which can be used by the operator. This Reactor Vessel Level Instrumentation System does not replace existing systems and is not coupled to safety systems, but acts only to provide additional information to the operator.

The Upper Range Reactor Vessel Level Instrumentation has differential pressure measurement across the upper region of the reactor vessel. The system utilizes two differential pressure cells measuring the differential pressure from the reactor coolant hotleg piping to the top of the reactor vessel head. The system provides an indication of reactor vessel water level above the hotleg pipe when the pump in the loop with the hotleg connection is not operating. A pump operating in a loop without the hotleg connection will have an effect of less than 10 percent on this indication's range. When the pump is operating in the loop with the hotleg connection, the instrument reading will be offscale.

The narrow range reactor vessel level instrumentation measures vessel level from the hotleg to the bottom of the reactor vessel when no reactor coolant pumps are running.

The wide range reactor vessel level instrument measures the reactor core internals and outlet pressure drop for any combination of pumps running. Comparison of any measured pressure drop with the measured pressure drop during normal operation will provide an approximate indication of the relative void content or density of the circulating fluid.

To provide the required accuracy for water level measurement, temperature measurements of the reference legs are provided.

These measurements together with the reactor coolant temperature and pressure measurements are used to compensate the differential pressure particularly during the environment inside the containment structure following an accident.

The Reactor Vessel Level Instrumentation System utilizes differential pressure cell instrumentation in two of the hotleg pipes. The instrumented hotleg piping will not be adjacent, but with respect to the plant layout, will be on opposite sides of the reactor vessel. The differential pressure cells are to be located outside of containment so that calibration cell replacement, reference leg checks and filling, and operation are made more easily and the overall system accuracy is improved.

Instrumentation for the operator for the Reactor Vessel Level Instrumentation System is intended to be unambiguous and reliable so that operator error or misinterpretation is avoided.

Upper range, narrow range, and wide range level signals are available from each train for display on standard VX-252 type vertical scale voltage meters. Thus, the indication is compatible with existing control board layouts. The indication signals are electrically isolated from the protection set and are suitable to serve as either a standard control grade or postaccident monitoring output.

The control board displays provide the following information:

1. An indication of reactor vessel level (narrow range) for each instrumented set displaying vessel level in percent from 0 to 60 percent after compensation for the effects of the reactor coolant and capillary line temperature and density, when reactor coolant pumps are not operating.

2. An indication of reactor differential pressure (d/p) (wide range) from each instrumented set displaying d/p in percent from 0 to 100 percent, after compensation for the effects of the reactor coolant and capillary line temperature and density effects, when reactor coolant pumps are operating.

3. An indication of upper range vessel level on each of the two instrumented sets displaying vessel level in percent from 60 to 100 percent after compensation for any reactor coolant and capillary line density effects, when the reactor coolant pump in the loop with the hotleg connection is not operating. A status light will indicate the operation of the reactor coolant pump with the hotleg connection.

Redundant displays are provided for the two sets. Level information based on all three d/p measurements is presented. Correction for reference leg densities is automatic.

The Reactor Vessel Level Instrumentation is to be used in conjunction with a coolant subcooling readout and incore thermocouple readout to determine the state and transient behavior of the reactor coolant system. The reactor vessel wide range level indication will read onscale with all four reactor coolant pumps running during normal operation from 0 to 100 percent full power. With all pumps shut down, the narrow and upper range indicators will provide a direct indication of water level in the reactor vessel.

Incore Thermocouples

1. The Watts Bar Nuclear Plant incore thermocouples are located at the core exit for each quadrant and, in conjunction with core inlet RTD data, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. Sixteen (four per quadrant) of the core-exit thermocouples will be designated as PAM sensors.

2. The primary operator display is a computer-driven printer. This system has the following capabilities:

   a. A spatially oriented core map is available on demand which indicates the temperature at each core exit thermocouple location.

   b. An example of the Watts Bar selective readings is an on-demand tabular listing of all instantaneous incore thermocouple values.

   c. A printout of average, instantaneous, and maximum values is provided for all T/C temperatures. The range will meet the suggested range of 200°F to 2300°F.

   d. Trend capability showing temperature time histories is designed into the system. Strip chart recorder points are available to assign to any incore thermocouple on demand. In addition, a point value trend printout is available on the control room printer.

   e. Alarm capability is provided in conjunction with the subcooling monitor which uses the average of all the T/C readings in the calculations.

   f. The control room displays are designed for rapid operator access and ease of viewing data. Also, the incore program has a validity-check comparison which reduces the probability of accessing false readings.

3. A backup analog readout is provided with the capability of selective reading of any T/C in the system. The range of the system is 0-700°F. However, TVA will extend the temperature range of the backup analog readout to 2300°F per the requirements of item II.F.2.

   Another means of obtaining this data can be obtained by reading the raw signals (T/C and reference junction output) with portable test equipment. This data is available in the control building and would be accessible under all conditions should the primary and backup display devices fail.

4. This system will be reviewed by the human factors review group as a part of NUREG-0700 task.

5. Conformance to Appendix B.

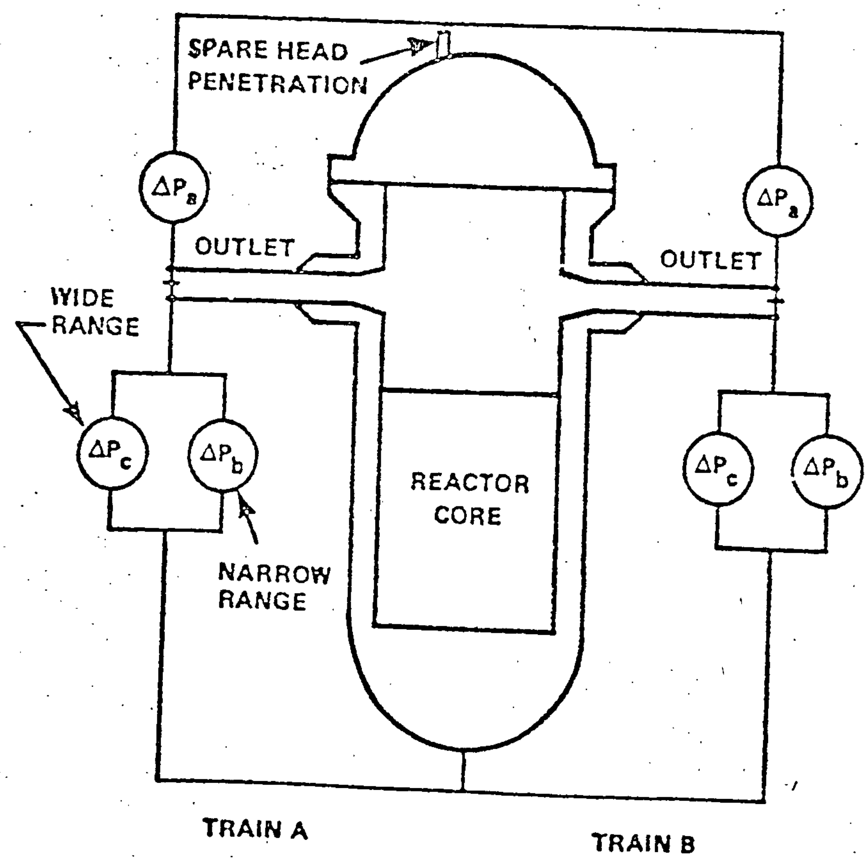
   The existing system does not meet the requirements of Appendix B. Evaluations are being performed to determine to what extent modifications should be made to upgrade this system. This evaluation, along with an implementation schedule, will be available by fuel loading on Watts Bar unit 1.

6. Present isolation between the primary and backup channels is implemented in the form of electrical switches. The primary and backup display channels are powered by a reliable battery-backed power source.

7. The existing incore T/C system is a very simple set of hardware which should, by virtue of its simplicity, be a highly reliable and accessible system.

8. Same as item 7.

9. Same as item 7.



Reactor Vessel Level Instrument System

Figure II.F.2-1

EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

TVA RESPONSE (Revised March 9, 1982)

The power-operated relief valves (PORV) and their associated block valves and control components are supplied from the emergency onsite power supply if offsite power is lost. The relief valves and their associated block valves are powered from opposite power trains. All connections to the emergency power supply are through devices that are qualified in accordance with safety grade requirements. For a description of the PORV and block valves, see FSAR Section 5.2.

The pressurizer level indication instrumentation power is taken from the vital power bus (see FSAR Section 7.5). These buses are supplied from the emergency power source when offsite power is unavailable.

Since the Watts Bar design meets NRC recommendations, no changes are anticipated and therefore, the capability to open PORV/block valves will not be affected.

AUTOMATIC TRIP OF REACTOR COOLANT PUMPS
DURING LOSS-OF-COOLANT ACCIDENT

<u>TVA RESPONSE</u> (Revised March 9, 1982)

In response to IE Bulletin 79-06C, Westinghouse (in support of the Westinghouse Owners Group) performed an analysis of delayed reactor coolant pump (RCP) trip during small break loss-of-coolant accidents (LOCA). This analysis is documented in WCAP-9584 and WCAP-9585. Westinghouse (again in support of the Westinghouse Owners Group) has performed test predictions of the LOFT Experiments L3-1 and L3-6. The results of these predictions were submitted to the NRC by Westinghouse Owners Group Letters OG-49 (dated March 3, 1981), OG-50 (dated March 23, 1981) and OG-60 (dated June 15, 1981). The results constitute both a best estimate model prediction with the NOTRUMP computer program and an evaluation model prediction with the Westinghouse FLASH computer program using the supplied set of initial boundary assumptions.

Based on: 1) the Westinghouse analysis, 2) the excellent prediction of the LOFT Experiment L3-6 results using the Westinghouse analytical model, and 3) Westinghouse simulator data related to operator response time, the Westinghouse and TVA position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Our understanding of the schedule for final resolution of this issue is:

A) Once the NRC formally approves the Westinghouse model, a 3-month study period will ensue during which the Westinghouse Owners Group will attempt to demonstrate compliance with some NRC acceptance criteria for manual RCP trip. The NRC acceptance criteria will accompany their formal approval of the Westinghouse models.

B) If, at the end of the 3-month period, the Westinghouse Oweners Group cannot show compliance with acceptance criteria, the NRC will formally notify utilities that they must submit an automatic RCP trip design.

REVISED SMALL-BREAK LOSS-OF-COOLANT ACCIDENT
METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

TVA RESPONSE (Revised March 9, 1982)

It is both Westinghouse's and TVA's position that the small break LOCA analysis model currently approved by NRC for use on Watts Bar is conservative and in conformance with 10 CFR 50, Appendix K. However, as documented in Westinghouse letter NS-TMA-2319 (dated September 26, 1980), Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has committed to revise its small break LOCA analysis model to address NRC concerns (e.g., NUREG-0611, NUREG-0623, etc.). This revised Westinghouse model is currently scheduled for submittal to the NRC by April 1, 1982.

TVA RESPONSE (Revised March 9, 1982)

TVA has met milestones 1 through 4. The remaining items are addressed below.

1. Primary meteorological measurements program -

   The meteorological measurements program is designed to conform to the intent and guidance of Regulatory Guide 1.23.

   a. Wind direction and wind speed are measured at three levels, low (10 meters), high (90-110 meters), and at mid-range (40-60 meters). Standard deviations of wind direction fluctuations (sigma theta) are calculated every 5 minutes for all 3 levels, based upon wind direction readings every 5 seconds. The 5-minute values are then averaged to produce an hourly estimate of sigma theta. Wind speed readings are taken every 15 seconds.

   b. Air temperatures are measured at the same levels at 1-minute intervals. From these values the average hourly vertical temperature difference for the layers between the low and mid-range levels, the low and high levels, and the mid-range and high levels are calculated. The temperature difference is used to estimate the Pasquill stability class.

   c. Hourly average dewpoint temperatures at the low level are calculated from dewpoint temperatures read at 1-minute intervals.

   d. Precipitation is measured by a standard recording rain gauge and reported at hourly intervals.

Hourly and 15-minute average meteorological data are transmitted from the Sequoyah Environmental Data Station to the TVA emergency centers, State Emergency Centers, and to the Contract Meteorological Forecast Service. Average values of wind direction, wind speed, and temperature are transmitted each 15 minutes. Hourly averages of these parameters, in addition to sigma theta, precipitation, and dewpoint, are transmitted each hour. More specific information on the meteorological measurements program can be found in the Final Safety Analysis Report (FSAR).

The meteorological measurements station is equipped with a backup electrical power generation system.

2. Backup meteorological data estimation procedures -

   TVA is preparing objective backup procedures to provide estimates for missing or garbled data needed to perform dose calculations and to determine transport estimates. The procedures will be designed to accommodate both partial and total primary system outages. They incorporate available onsite and offsite data (from other TVA nuclear plants and National Weather Service first-order stations) and conditional climatology. Each procedure has an accompanying statement of reliability.

If primary system data are missing or garbled, backup procedures are applied within five minutes (as feasible) to provide estimates for the unavailable variables. The backup procedures can be applied by the contract meteorological forecaster or by the Muscle Shoals Emergency Control Center (MSECC) staff.

Whenever backup procedures are used in lieu of primary system data, the MSECC meteorologist informs the staff of this and advises on the reliability of the procedures being used, so that additional protective actions may be taken, if necessary.

3. Real-time and forecast meteorological data for transport and diffusion -

Within 30 minutes of notification of a real or simulated emergency condition, the contract forecaster issues a forecast for 1 and 2 hours of wind direction, wind speed, and stability conditions, for indicated levels and layers, and of precipitation conditions. This information along with current meteorological parameters is used by the MSECC dose assessment staff to project offsite doses and dose rates.

The projection of doses is discussed in detail in MSECC-IPD, IP-10, 'Emergency Dose Assessment Procedures for Atmospheric Releases of Radioactivity,' and in OHS-20-80-05, Rev. 2, 'Emergency Dose Assessment Procedures for Atmospheric Release of Radioactivity - Technical Basis.' Within one hour after notification and at hourly intervals thereafter, or more often if changing meteorological circumstances warrant, a forecast is made of wind direction, wind speed, and stability conditions, for the selected levels and layers, and of precipitation type and intensity. These forecasts predict conditions at 1, 2, 4, 6, and 8 hours into the future with an outlook particularly addressed toward changing meteorological conditions for the following 16 hours.

In addition, the latest wind direction, wind speed, stability, precipitation information, and plume trajectory is available to the MSECC staff at 15-minute intervals. The plume trajectories are based on the observed 15-minute and hourly average wind information.

More detailed information on the meteorological aspects of assessment actions is provided in MSECC-IPD, IP-7, 'Air Resources Program Procedures.'

4. Remote access of meteorological data -

Near real-time access of meteorological data is available to authorized users through the Central Emergency Control Center computer. The remote access system gathers data from TVA nuclear plants, performs units conversion and reformats data. The system stores up to 168 hours of 15-minute and hourly observations.