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

October 23, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

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

By a letter dated August 10, 1981 from R. L. Tedesco to H. G. Parris, TVA was requested to address the 13 "Unresolved Safety Issues" pertaining to Watts Bar Nuclear Plant units 1 and 2. Information supporting TVA's position that Watts Bar Nuclear Plant can be operated safely until final resolution of these generic issues was provided by my letter to you dated September 11, 1981.

The enclosed information provides the same supporting information. However, minor modifications have been made to items A-1, A-3, A-43, and A-44.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

M. R. Wisenburg
Nuclear Engineer

Sworn to and subscribed before me
this 23rd day of Oct, 1981
Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

Enclosure

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ENCLOSURE
WATTS BAR NUCLEAR PLANT UNITS 1 AND 2
GENERIC UNRESOLVED SAFETY ISSUES

A-1 - Water Hammer

Water hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions. Since 1971 there have been over 100 incidents involving water hammer in pressurized water reactors and boiling water reactors. The water hammers have involved steam generator feedrings and piping, decay heat removal systems, emergency core cooling systems, containment spray lines, service water lines, feedwater lines and steam lines. However, the systems most frequently affected by water hammer effects are the feedwater systems. System design changes and testing requirements necessary to prevent this type of water hammer have been implemented in accordance with recommendations provided in Westinghouse report 'Minimization of Series D2/D3 Steam Generator Preheating Pressure Transient' dated June 30, 1977. TVA has concluded that, subject to confirmation during the preoperational test program, the feedwater system and steam generator design for Watts Bar units 1 and 2 with respect to this potential water hammer concern is acceptable.

With regard to protection against other potential water hammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. TVA will conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

In the unlikely event that a large pipe break did result from a severe water hammer event, core cooling is assured by the emergency core cooling systems described in Section 6.3 of the FSAR and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided as described in Sections 3.6 of the FSAR.

Task A-1 may identify some potentially significant water hammer scenarios that have not explicitly been accounted for in the design and operation of nuclear power plants, including the Watts Bar units. The task has not as yet identified the need for requiring any additional measures beyond those already required in the short term.

TVA has concluded that Watts Bar units 1 and 2 can be operated until there is an ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-3 - Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. In addition, the requirement for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers. Corrosion resulting in steam generator tube wall thinning (wastage) has been observed in several Westinghouse plants. However, plants operating exclusively with an all-volatile secondary water treatment process have not experienced this form of degradation. Another observed corrosion-related phenomenon has been buildup of support plate corrosion products in the annulus between the tubes and support plates. This buildup may eventually cause a diametric reduction of the tubes, called "denting" and deformation of the tube support plates. This phenomenon may lead to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

The following is a technical summary of the Watts Bar Nuclear Plant (WBN) design information related to this issue:

Tube Material

Tubes are .750" od X .043" wall, Inconel Alloy 600 to ASME SB-163.

Environmental Control

WBN will operate on an all-volatile secondary water teatment process (AVT). Water chemistry is addressed in WBN FSAR chapter 10.

Dissolved and suspended impurities and corrosion products are removed from the system by means of six mixed-bed demineralizer vessels, five of which are in service and one in standby. Additional detail is addressed in WBN FSAR, chapter 10.

Tube Surveillance

Tube/tubesheet inspection is provided for in the following manner:

(1) Four 2" diameter ports, 90°F apart approximately 4.75" above the tube plate. These were designed and installed for sludge lancing.

(2) Two 6" diameter handholes, 180°F apart, approximately 31" above the tube plate.

(3) One 2.6" diameter inspection port approximately 234 inches above the tube plate.

Design

WBN consists of two units, four steam generators per unit (total 8). The SG tubes have been expanded full length in the tubesheet, thereby eliminating the crevice and opportunity for contaminants to enter this area and become a potential source of intergranular corrosion. There are 1,162 "U" bent tubes per SG. Tube bend radius is 3.5.

Technical Cognizance

TVA is a member of EPRI and the Steam Generator Owners Group (SGOG). TVA representatives regularly participate in these meetings to maintain awareness of industry experience and obtain updates on research in progress.

Model D SG Industry Experience

There is no commercial experience of any duration with Model D generators at this time. However, WBN is not scheduled for fuel loading until August 1982. By that time information may be available from other plants that can be factored into TVA procedures and controls.

Crack Preventing Measures

Water chemistry will be controlled as described previously. Sludge lancing will be used to control sludge if it forms. State-of-the-art chemical cleaning will be considered on a case-by-case basis.

Additional measures will be taken as circumstances dictate. Industry experience will be constantly monitored, as well as the results of the numerous Department of Energy (DOE), NRC, EPRI, and SGOG programs now underway.

Task A-3 is expected to result in improvements in current requirements for inservice inspection of steam generator tubes. Pending completion of this task, the design and operational provisions implemented at Watts Bar should minimize the occurrence of steam generator tube problems. Therefore, we believe that Watts Bar units 1 and 2 can be operated before ultimate resolution of the generic issue without undue risk to the health and safety of the public.

A-9 - Anticipated Transient Without Scram (ATWS)

TVA has set up an analytical program to examine the effects of the most severe postulated ATWS transients and to determine the mitigating effects of turbine trip and auxiliary feedwater initiation on the Sequoyah Nuclear Plant. The sensitivity of the effects of various nuclear and thermal parameters also are being evaluated. This program is being coordinated with Westinghouse, the aim being to come to eventual agreement on the magnitude and effects of these transients. Due to plant design similarities, we believe that these studies are also applicable to Watts Bar.

TVA has developed criteria to be used for the design of additional control rod trip (scram) gear. This design will be fully vendor-developed and will be implemented if shown to be needed to reduce the probability of failure to scram.

TVA is completing design criteria for systems to be used in mitigating the effects of scram failures. Installation of these systems could give further assurance of turbine trip and auxiliary feedwater initiation should a failure to scram occur. Preliminary design studies toward implementation of this alternate mitigating systems actuation circuitry (AMSAC) are also planned.

TVA will continue to remain informed on issues related to NRC rulemaking on USI A-9. Plans to resolve the ATWS issue for WBN will be discussed with the NRC as this program progresses.

Based on the foregoing, we believe that the Watts Bar Nuclear Plant is safe to operate until such time that the ultimate resolution of this generic issue can be reached.

A-11 - Reactor Vessel Materials Toughness

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as fracture toughness. Fracture toughness has different values and characteristics, depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important: first, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation

In recognition of these considerations, power reactors are operated within pressure restrictions imposed by the technical specifications in accordance with 10CFR50, Appendix G, during heatup and cooldown operations. These restrictions ensure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations.

The reactor coolant pressure boundary materials meet the fracture toughness requirements of Appendix G of 10CFR Part 50 except that paragraph IV.B of Appendix G has not been met by the intermediate shell forging for the unit 1 reactor vessel. This paragraph requires that the reactor vessel beltline materials have a specified minimum charpy V-notch upper shelf energy, unless it can be demonstrated to the Commission that a lower value will still provide an adequate margin against deterioration from irradiation. On the basis of our evaluation, we have concluded that the calculated fracture toughness values, based on the conservative Regulatory Guide 1.99 curves, are sufficiently high to assure that the safety margins specified in Appendix G, Section III, of the ASME Code will be maintained for unit 1 at operating temperatures and pressures during the first five effective full power years of plant life. Before this time, removal of the first material surveillance capsule will provide an opportunity to refine the material property estimate. The unit 2 reactor vessel meets the fracture toughness requirements of Appendix G to 10CFR Part 50. Therefore, it is expected to meet the specified safety margins throughout its life.

Since the unit 1 reactor vessel may have marginal fracture toughness based on our current conservative assessment after five effective full power years of operation, its available fracture toughness will be reassessed before allowing operation beyond this point in plant life. The improved engineering method and safety criteria being developed under Task A-11 are expected to allow a more accurate assessment of the available safety-margin over plant life and accordingly are expected to be used for the reassessment of the unit 1 vessel. Test A-11 results will be available long before they are needed for application to the analysis of the Watts Bar unit 1 vessel.

In addition to the evaluation for normal operating conditions, we have evaluated the integrity of the units 1 and 2 reactor vessels during accident conditions. These evaluations demonstrate that the acceptance criteria for accident conditions are met for periods of operation well in excess of the limitations imposed by the Appendix G criteria discussed above.

Based on the foregoing, it is concluded that Watts Bar units 1 and 2 can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-12 - Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

TVA is currently reviewing the materials used in all primary reactor coolant system component (reactor vessel, steam generators, reactor coolant pumps, and pressurizer) supports to determine the degree of compliance with the proposed criteria contained in NUREG-0577, 'Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports' (issued by the NRC in November 1979 for comment). Also, we are participating in the recently formed Atomic Industrial Forum (AIF) Subcommittee on Material Requirements and with the Metals Property Council to develop an industry response to the generic issue. A test program has been initiated at Sequoyah to monitor actual support temperatures. This information will be extrapolated to the Watts Bar supports and will be utilized in our repsonse to USI A-12.

In the interim, until the fracture toughness of the support materials at Watts Bar units 1 and 2 is evaluated and corrective measures implemented, if needed, operation is justified based on the following:

Support failures from inadequate fracture toughness are not expected to occur except under the unlikely combination of:

(1) The initiating event (i.e., a large pipe break) which has a low probability of occurrence.

(2) The existence of nonredundant and critical support structural member(s) with low fracture toughness (many supports contain redundant members).

(3) The existence of support structural members at operating temperatures low enough that the fracture toughness of the support material is reduced to a level at which brittle failure could occur if a large flaw existed.

(4) The existence of a flaw of such size that the stresses imparted during the initiating event could cause the flaw to rapidly propagate, resulting in brittle failure of the member(s).

Accordingly, we have concluded that Watts Bar can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

A-17 - Systems Interaction

The NRC implemented an investigation, phase 1 of System Interactions (A-17), to confirm that their present review procedures and safety criteria provide an acceptable level of redundancy for nuclear safety features. These nuclear safety features, systems essential to bring the plant to a safe shutdown condition in the event of a major accident, are referred to as engineered safety features (ESF). Phase 1 consists of an analysis structured to identify areas where adverse interactions are possible between and among systems and have potential for negating or degrading the performance of ESF.

The NRC has completed their review of our Sequoyah Nuclear Plant (SQN) with regard to the A-17 Phase 1 task. In NUREG-0011, Supplement No. 1, dated February 1980, the NRC stated that their review plan and acceptance criteria currently provided adequate assurance that an acceptable level of system independence and redundancy is provided in the SQN design. Since the ESF design at SQN and WBN are essentially identical, we believe that the level of redundancy and independence within the ESF at WBN are acceptable to the present NRC review procedures and safety criteria. Adherence to the principle ESF design requirements such as physical separation, independence of redundant safety features, emergency power and protection against fire, high energy line breaks, missiles, natural phenomena, and sabotage is the same within SQN and WBN.

We have performed extensive studies of various engineered safety features at both the SQN and WBN plants. These include pipe break, fire protection, ATWS, and IE Bulletin 79-27 studies. Most of the design reviews were prompted by NRC IE bulletins, generic NRC concerns, and new licensing requirements. They served to confirm the independence, redundancy, and adequacy of the WBN ESF with any deficiencies being resolved for both SQN and WBN when necessary. Differences between the plants were identified and determined to be acceptable. Based on the similarity of the plants, we have determined that the potential for undesirable system interactions which may have an adverse effect on the engineered safety features at WBN is at an acceptable level. Thus, we have concluded that Watts Bar units 1 and 2 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-40 - Seismic Design Criteria

NRC regulations require that nuclear power structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in regulatory guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to the Standard Review Plan sections and regulatory guides to bring them more in line with the state-of-the art are expected.

The seismic design basis for this facility is described in FSAR Section 3.7. Pending the identification of new issues by the NRC Staff that could alter the plant design basis, we do not expect the results of Task A-40 to affect our conclusions. Accordingly, we conclude that Watts Bar Nuclear Plant can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-43 - Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, i.e, a break in the reactor coolant system piping, the water flowing from the break would be collected in the emergency sump. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems.

One postulated means of losing the ability to draw water from the emergency sump could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release to the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing blockage.

Mirror insulation is the only type on the primary system piping and vessels. It has been determined from an examination of the possibility of blockage of the screens of the Sequoyah sump that the insulation blown off by a LOCA could block only a small area, much less than 50 percent, of the sump screens. TVA has performed out-of-plant scale model tests of the Watts Bar containment sump design (reference J. E. Gilleland's letter to S. A. Varga dated May 23, 1979) which demonstrated that a 50-percent blockage of the screen area had a negligible effect on the sump operation.

Peeling paint has also been identified as a possible hazard to the operation of the containment sump. To prevent this hazard, surface coatings have been qualified to withstand possible upset DBA conditions per ANSI Standard N101.2-1972, 'Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities.'

Currently, regulatory positions regarding sump design are presented in Regulatory Guide 1.82, 'Sumps for Emergency Core Cooling and Containment Spray Systems,' which address debris (insulation). The Regulatory Guide recommends, that two protective screens be provided. A low approach velocity in the vicinity of the sump is required to allow insulation to settle out before reaching the sump screening; and it is required that the sump remain functional assuming that one-half of the screen surface area is blocked.

A second postulated means of losing the ability to draw water from the emergency sump could be abnormal conditions in the sump or at the pump inlet such as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps.

Currently, regulatory positions regarding sump testing are contained in Regulatory Guide 1.79, 'Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors,' which addresses the testing of the recirculation function. In-plant tests will be performed to demonstrate that circulation through the sump can be reliably accomplished. The out-of-plant tests mentioned above were performed utilizing several design modifications identified during similar sump model tests for the Sequoyah Nuclear Plant. These modifications have been incorporated into the Watts Bar design. From the tests, we concluded that there was reasonable assurance that the sump design would perform as expected following a LOCA and, therefore, was acceptable. Accordingly TVA believes that Watts Bar Units 1 and 2 can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 - Station Blackout

A loss of all ac power is not a design basis event for Watts Bar units 1 and 2. Nonetheless, the combination of design, operation, and testing requirements that have been imposed on the plant will assure that these units will have substantial resistance to this event and that even if a loss of all ac power should occur, there is reasonable assurance that the core will be cooled. These are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in section 8.2 of the Watts Bar units 1 and 2 FSAR. Also, a preliminary grid stability analysis was performed, portions of which may be found in the Watts Bar FSAR, section 8.2.

If offsite ac power is lost, two independent and redundant onsite diesel generators per unit and their associated distribution systems will deliver emergency power to safety-related equipment. The design, testing, surveillance, and maintenance provisions for the Watts Bar units 1 and 2 onsite emergency diesels are described in section 8.3.1 of the FSAR. TVA's requirements include preoperational testing to assure the reliability of the installed diesel generator in accordance with the provisions of Regulatory Guide 1.108.

Even if both offsite and onsite ac power are lost, cooling water from the condensate storage tank can still be provided to the steam generators by the auxiliary feedwater system by employing a steam turbine driven pump that does not rely on ac power for operation. Our description of the auxiliary feedwater system design and operation is described in section 10.4.9 of the Watts Bar FSAR.

In addition, as part of the Sequoyah unit 1 low power test program, TVA performed a 'blackout' test, which simulated loss of all offsite and onsite ac power. The test results showed that during the blackout period, the turbine-driven auxiliary feedwater pumps delivered flow to the steam generators as required. Therefore, while the designs for the ac power and auxiliary feedwater systems at Sequoyah and Watts Bar are not identical, it is TVA's opinion that sufficient similarities exist in the designs such that the test results (from a system operation standpoint) could be considered applicable to Watts Bar. Additional information on this test and other concerns (operator training, procedures, etc.) related to this issue is provided in Supplement No. 2 to the Sequoyah safety evaluation report and TVA's response (L. M. Mills June 5, 1981, letter to E. Adensam) to the NRC's generic letter No. 81-04 'Emergency Procedures and Training for Station Blackout Events.'

Based on the above, we have concluded that there is reasonable assurance that Watts Bar units 1 and 2 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-45 - Shutdown Decay Heat Removal Requirements

This unresovled safety issue is concerned with the ability of Watts Bar Nuclear Plant to remove decay heat in the event normal decay heat removal systems are unavailable, for an unspecified reason, during a plant transient.

WBN is designed to be normally cooled down by the steam generators (SG) to reactor coolant system (RCS) conditions of approximately 350°F and 400 lb/in^2, at which point the cooldown is continued by the residual heat removal system (RHR). Feedwater supply during this normal cooldown is from the condensate-feedwater system (described in FSAR section 10.4.7) or the auxiliary feedwater (AFW) system (FSAR section 10.4.9); heat removal is to the condenser through the main steam system (FSAR seection 10.3). The RHR system design is discussed in FSAR section 5.5.7.

In the event normal feedwater is unavailable, the AFW has sufficient stored water capacity to hold both reactors at hot standby for two hours and then cooldown to the RHR initiation conditions using condensate water from the condensate storage tank. In addition, a backup connection from the essential raw cooling water (ERCW; see FSAR section 9.2) ensures an indefinite supply of water. These systems and the RHR system, plus all essential supporting systems, are fully qualified and redundant and are able to perform their essential functions without offsite power or dependence on unqualified systems.

If a total loss of ac power, both from offsite and from the onsite emergency diesel generators, is assumed, each reactor is served by a turbine-driven AFW pump that supplies sufficient feedwater to the SG to remove decay heat for a minimum of two hours. The water supply for each pump is gravity-fed from the condensate storage tank, and all power requirements are furnished by the vital 125-V dc control power system (FSAR section 8.3.2). Energy removal from the SG in this event can be accomplished by use of the atmospheric relief valves or the safety valves (described in FSAR section 10.3). The ability to operate in this mode was confirmed by a special low power testing program for SQN unit 1 (see response to Unresolved Generic Safety Question A-44).

In the event of a total loss of feedwater, 'feed and bleed' is conceivable at WBN, but is not a system design basis. This emergency operation was discussed with the ACRS on June 2, 1980. In addition, the EPRI testing program currently (August 1981) underway will verify the ability of the pressurizer power-operated relief valves to sustain the hydraulic conditions associated with water relief during the 'feed and bleed' operation.

Based on the foregoing, we believe that WBN can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-46 - Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program.

Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shut down the plant as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Watts Bar Nuclear Plant was designed using the seismic criteria and analytical methods described in sections 3.7, 3.9, 3.10, and 5.2 of the FSAR. These criteria and methods have been reviewed by the Staff and any remaining outstanding issues are being resolved in accordance with current design criteria for seismic qualification. Therefore, we conclude that Watts Bar can be operated before resolution of this generic issue without undue risk to the health and safety of the public.

A-47 - Safety Implications of Control Systems

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. The independent control system failure or 'single failure' is of some concern due to the potential for a failure such as loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those analyzed as anticipated operational occurrences. Another concern is for a postulated accident to cause a control system failure which could make the accident more severe than analyzed. This latter failure could also result from a single failure-induced accident. These accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment.

The Watts Bar Nuclear Plant control and safety systems have been designed with the goal of ensuring that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment required for accident mitigation and/or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or 'accident.' This has been accomplished by providing independence and physical separation between safety system trains and between safety and nonsafety systems. For the latter, as a minimum, isolation devices were provided. These devices preclude the propagation of nonsafety system equipment faults to the protection systems. Also, to ensure that the operation of safety system equipment is not impaired, the single failure criterion has been applied in the plant design. Design reviews are being performed (IE Bulletin 79-27) to ensure that the loading on certain class 1E power boards maintains the separation and independence of plant systems as designed.

A systematic evaluation of the control system design, as contemplated for this Unresolved Safety Issue, has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, a wide range of bounding transients and accidents is presently analyzed to assure that the postulated events such as steam generator overfill and overcooling events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems are being performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action. These reviews are part of an ongoing evaluation program to qualify class 1E plant equipment to function for all postulated service conditions to which it is subjected (NUREG-0588).

Based on the above, we have concluded that there is reasonable assurance that the Watts Bar Nuclear Plant can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-48 - Hydrogen Control Measures and Effects of Hydrogen Burn on Safety Equipment

The research and development work being performed by TVA for the Sequoyah Nuclear Plant is directly applicable to the Watts Bar Nuclear Plant. A description of both the TVA research program and the TVA hydrogen control system was included in the NRC Safety Evaluation Report, Supplements No. 3 and 4 for Sequoyah.

TVA is conducting research into hydrogen combustion phenomena and mitigation measures in cooperation with Duke Power, American Electric Power, the Electric Power Research Institute, and on its own. Information about this research program has been submitted by TVA to the NRC in three quarterly reports (December 16, 1980, March 16 and June 16, 1981).

Information on the Interim Distributed Ignition System (IDIS) that is currently installed at Sequoyah and on the technical and analytical bases for its design was provided to the NRC by TVA in Volume II of our Hydrogen Control Study (September 2, 1980), revision 1, to Volume II (December 15, 1980), and Resolution of Equipment Survivability Issues (May 29, 1981). Although these submittals were for the IDIS, which is not going to be installed at Watts Bar, the information formed the basis for the NRC Sequoyah SER, Supplement No. 4, and also pertains to the Permanent Hydrogen Mitigation System (PHMS) which is to be installed at both Sequoyah and Watts Bar.

Information on the PHMS selection and preliminary design was provided by TVA to the NRC in early July 1981. In response to NRC questions, more submittals on the PHMS are scheduled by TVA for fall 1981.

The ongoing TVA research effort is confirmatory in nature, and results to date are supportive of the mitigation system selected. No alternative courses of action are planned pending completion of the generic degraded core rulemaking. Accordingly, pending resolution of this unresolved safety issue and the rulemaking proceeding on hydrogen generation, we believe that Watts Bar can be operated without undue risk to the health and safety of the public.