NUREG-0847
Supplement No. 1

Safety Evaluation Report
related to the operation of Watts Bar Nuclear Plant, Units 1 and 2

Docket Nos. 50-390 and 50-391

Tennessee Valley Authority

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1982

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ABSTRACT

This report supplements the Safety Evaluation Report, NUREG-0847, issued June 1982 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the open items identified in the Safety Evaluation Report and discusses recommendations of the Advisory Committee on Reactor Safeguards in its report dated August 16, 1982.

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APPENDICES

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (the applicant or TVA) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. This report is the first supplement to that Safety Evaluation Report (SER).

This supplement provides more recent information regarding resolution of some of the open items identified in the SER. This supplement also provides and discusses the recommendations of the Advisory Committee on Reactor Safeguards (ACRS) in its report on the Watts Bar Nuclear Plant, dated August 16, 1982. Another supplement to the SER will be issued before fuel loading of Unit 1 to discuss the resolution of the other open and confirmatory items identified in the SER.

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography. Appendix E is a list of principal contributors to this supplement. Appendix F is a copy of the ACRS report. Appendix G is a list of errata for the SER. No changes in SER Appendices C and D have been made by this supplement.

The Project Manager is Thomas J. Kenyon; he may be reached on (301) 492-7266.

1.7 Summary of Outstanding Issues

In Section 1.7 of the SER the staff noted that certain items remained outstanding at the time the report was issued. This supplement updates some of those items for which the staff has completed part or all of its review. These items, and the sections of this supplement discussing the staff's review conclusions, are

(4) Seismic qualification of equipment (3.10)
(7) Model D-3 steam generator preheater tube degradation (5.4.2.2)
(15) Physical Security Plan (13.6)

1.9 License Conditions

In Section 1.9 of the SER the staff identified 37 license conditions. There is an additional issue for which a license condition may be desirable to ensure that staff requirements are met during plant operation. This item, with the appropriate reference to its section in this supplement, is

(38) Physical Security Plan (13.6.4)

3 DESIGN CRITERIA - STRUCTURE, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

The following evaluation applies to Unit 1 of the Watts Bar facility only. The Unit 2 review and evaluation will be performed at a later date, and the results will be reported in a future supplement to the SER.

The staff's evaluation of the adequacy of the applicant's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT) has reviewed the equipment dynamic qualification information contained in the pertinent Final Safety Evaluation Report (FSAR) Sections 3.9.2 and 3.10. The SQRT made a site visit on April 26 through April 30, 1982 to determine the extent to which the qualification of equipment (as installed in Watts Bar Unit 1) meets the current licensing criteria as described in Regulatory Guides 1.92 and 1.100 and the Standard Review Plan (SRP) Section 3.10 (NUREG-0800). A representative sample of seismic Category I mechanical and electrical equipment, as well as instrumentation, included in both nuclear steam supply system (NSSS) and balance-of-plant (BOP) scopes, were selected for the plant site review. The review consisted of field observations of the actual equipment configuration and its installation, followed by the review of the corresponding test and/or analysis documents.

In instances where components have been qualified by test or analysis to other than current licensing criteria such as Regulatory Guides 1.92 and 1.100 and SRP Section 3.10, the applicant is currently re-evaluating the acceptability of the qualification program in the light of current criteria.

In the SQRT onsite audit exit meeting and the trip report of the SQRT site visit, the staff concluded that in order to complete the review, it would need additional information and clarification of the details of the qualification for some pieces of equipment. The trip report includes both generic and equipment-specific concerns. The applicant will be required to address and resolve the specific concerns before initial fuel load.

The generic concerns are significant, in that they apply to all safety-related equipment and can potentially affect a large number of equipment items. Therefore, an acceptable approach and a plan to implement the resolution of generic issues will also be required before fuel load. A list of the generic concerns is summarized below:

Generic Open Items

(1) Single-axis and single-frequency tests were performed to qualify equipment. For equipment in the flexible range (below 33 Hz) these tests may not challenge the multi-axis and multi-frequency response of the equipment. The extent to which this issue can affect the qualification status of equipment at the plant is the thrust of this concern.

(2) In numerous cases the required response spectra (RRS) were not broadened at the peaks to account for the uncertainty in the prediction of natural frequencies of the supporting structures. Also, sufficient margins must be included in the test response spectra (TRS) to account for the uncertainty in manufacturing process and the test apparatus.

(3) In numerous cases the field mounting of the equipment is by welding of various lengths, whereas the mounting for the qualification testing is by bolting. Changes in the field mounting could alter the dynamic characterristics of the system invalidating the results of the original qualification testing. Devices within electrical cabinets could be more susceptible to this type of problem.

(4) Many safety-related equipment are age sensitive with respect to their seismic performance; for example, the insulation of motors, transformers and other electric devices, and diaphragms for valves may limit the qualified life. To ensure that seismic resistance of safety-related equipment is available throughout the plant life, a detailed program of surveillance and maintenance must be developed.

Resolution of the specific and generic items as they progress will be reported in a future supplement to the SER.

As discussed in Section 3.10 of the SER, the seismic hazard for the Watts Bar site was redefined by the staff during the operating license review. The seismic hazard is now specified in terms of a site-specific spectrum, which corresponds to the 84th percentile spectrum shape derived from a number of recorded time histories of ground motion. Whether or not the seismic input used for qualification of safety-related equipment is conservatively bounded by the input actually used for qualification was a concern. By submittals dated December 9, 1981, the applicant provided a comparison between the Watts Bar spectrum against the 84th percentile spectrum. These comparisons were made for higher values of damping representative of the structural damping and they established the adequacy of the Watts Bar spectrum. Because equipment damping values are somewhat lower than the structural damping values, the applicant, in a submittal dated April 21, 1982, provided additional comparisons between the 84th percentile spectrum and the Watts Bar spectrum for lower damping values representative of equipment damping values. During the site audit, the staff examined the selected equipment input spectra incorporated in the purchase specification against the 84th percentile spectrum and confirmed that the Watts Bar spectrum was properly used and that it adequately covers the 84th percentile spectrum. Thus there is reasonable assurance that the seismic input used for the qualification for safety-related equipment is at least as severe as can be expected from the redefined seismic hazard for the Watts Bar site.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.4 Component and Subsystem Design

5.4.2 Steam Generators

5.4.2.2 Westinghouse Model D Steam Generator Tube Degradation Potential

As discussed in the SER, a generic problem concerning the potential for tube degradation caused by flow-induced vibration in the preheater section of Westinghouse Model D steam generators has been identified. Because the Watts Bar facility uses the Model D-3 steam generator, the applicant is working with Westinghouse, Duke Power Company, and South Carolina Electric and Gas Company (the latter two being owners of facilities with Model D-2/D-3 steam generators) in resolving this concern.

During a June 30 - July 1, 1982 meeting, Westinghouse made presentations regarding their proposed modifications to the Model D-2/D-3 steam generators to the Design Review Panel (DRP) with the NRC staff in attendance. The 19 member DRP was established by TVA, Duke Power Company, and South Carolina Electric and Gas Company to (1) evaluate the final proposed modifications to the preheater to ensure that applicable regulatory criteria are met, (2) verify the functional adequacy of the proposed design, and (3) ensure the safety acceptability of the design. The DRP is comprised of representatives of the above utilities (or their consultants) with expertise in such disciplines as materials engineering, structural mechanics, radiation exposure, quality assurance, cooling and insulation engineering, and thermal-hydraulics.

On August 25, 1982, the staff again met with representatives of the DRP and Westinghouse to discuss the scope and content of the DRP's safety evaluation of the proposed Westinghouse modifications to the model D-2/D-3 steam generators. The NRC staff made recommendations to the DRP to assist them with their review of these modifications. During these and other meetings with the DRP and Westinghouse, the staff has monitored the analysis and development work by Westinghouse to resolve this matter. The vendor has developed a manifold to be installed in the feedwater inlet nozzle of the steam generator, based on the model testing and analytical programs. The manifold will disperse the fluid flow so that impingement of the water on the rows of tubes in the preheater region will be reduced and thereby eliminate the vibration responsible for the accelerated wear of the steam generator tubes. The final design review of this manifold and related modifications is anticipated to be completed by the DRP in mid-September and will be presented to the staff at that time.

The procurement of materials, the tooling, and the procedures for the proposed modifications already have been completed by Westinghouse. Training is expected to be complete by mid-September 1982.

During the August 13, 1982 ACRS full committee meeting, TVA stated that the utility has determined that it would be to their advantage to install the manifold and complete any necessary modifications before hot functional testing at the Watts Bar facility. However, it has not been decided which facility (Watts Bar, Summer, or McGuire) will be the first domestic facility to receive the modifications.

The staff will continue to monitor developments of the Westinghouse program and will evaluate the proposed modifications and the DRP report upon final submittal. The staff will address this matter in a future supplement to the SER.

13 CONDUCT OF OPERATIONS

13.6 Physical Security Plan

In the SER the staff stated that appropriate revisions to the Watts Bar Physical Security Plan had not been filed. By letter dated July 29, 1982, TVA submitted their formal revisions to the plan. The following is the staff's evaluation of TVA's security program plans and a summary of how the applicant has provided for meeting the requirements of 10 CFR 73.

13.6.1 General

TVA filed the following security program plans with the NRC for the Watts Bar Nuclear Plant Units 1 and 2.

(1) "Watts Bar Nuclear Plant Physical Security Plan," dated June 3, 1982 as revised by letter of July 29, 1982

(2) "Watts Bar Nuclear Plant Contingency Plan," dated December 21, 1980 with revision dated March 30, 1981

(3) "Watts Bar Nuclear Plant Guard Training and Qualification Plan," dated August 17, 1979 with revision dated January 24, 1980, May 21, 1980, October 1, 1980 and March 9, 1981

13.6.2 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), TVA has provided a physical security organization that includes a Public Safety Service Shift Supervisor who is onsite at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, guard training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members have been written and are available for inspection.

The training program and critical security tasks and duties for the security organization personnel are defined in the "Watts Bar Nuclear Plant Guard Training and Qualification Plan," which meets the requirements of 10 CFR 73, Appendix B, for the training, equipping, and requalification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before the individual is properly trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.3 Physical Barriers

In meeting the requirements of 10 CFR 73.55(c), TVA has provided a protection area barrier that meets the requirements defined in 10 CFR 73.2(f)(1). To permit observation of activities along the barrier, an isolation zone of at least

20 ft is provided on both sides of the barrier with certain specified exceptions. The staff has reviewed these exceptions and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 ft-candles is maintained for the isolation zones, protected area barrier, and external portions of the protected area. In areas where illumination of 0.2 ft-candles cannot be maintained, special procedures are applied.

Vital equipment is located within vital areas that are located within the protected area and requires passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier.

Patrols of the protected area are performed at random intervals to detect the presence of unauthorized persons, vehicles, and materials.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows.

13.6.4 Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except applicant designated vehicles, are controlled by escorts. Applicant designated vehicles are limited to onsite station functions and remain in the protected area except for operational, maintenance, repair, security, and emergency purposes. Positive control over these vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel.

A picture badge/key card system, using encoded information, identifies individuals that are authorized unescorted access to protected and vital areas; it is used to control access to these areas. Individuals not authorized unescorted access are issued non-photo badges that indicate an escort is required. Access Authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. Access to the reactor containment(s) is positively controlled to ensure that only authorized individuals are permitted to enter. In addition all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated as a result of the lack of reliability or trustworthiness, or for poor work performance, the keys, locks, combinations, and related equipment to which that person has access are changed.

Section 9.1 of TVA's physical security plan allows designation of the containment as a nonvital area when the fuel is out of the core during major refueling and major maintenance. The staff finds this is in violation with the regulations and is, therefore, unacceptable. The license will be conditioned to ensure that the containment shall not be designated as nonvital under any circumstances.

13.6.5 Detection Aides

In satisfying the requirements of 10 CFR 73.55(e) the licensee has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. The licensee has exceeded the regulation by providing two perimeter-intrusion detection systems at the protected area barrier. Alarms from the intrusion detection systems annnunciate within the continuously manned central alarm station and a secondary alarm station located within the protected area. The central alarm station is located so that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central alarm station is constructed so that the walls, floors, ceilings, doors, and windows are bullet-resistant. The alarm stations are located and designed so that a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other function or duty that would interfere with its alarm response function.

The transmission lines and associated alarm annunciation hardware for the intrusion detection systems are self-checking and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.6 Communications

As required in 10 CFR 73.55(f) the licensee has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio link. All nonportable communiction links, except the conventional telephone system, are provided with an uninterruptable emergency power source backed up by diesel generators.

13.6.7 Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g) the applicant has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices or equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate compensatory measures as defined in the "Watts Bar Nuclear Plant Physical Security Plan" and onsite procedures. The compensatory measures defined in these plans will ensure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures.

The intrusion detection systems are tested for proper performance at the beginning and end of any period that they are used for security. Such testing will be conducted at least once every 7 days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by the Quality Assurance and Audit Staff, which is independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization in implementing the approved security program plans, include, but are not limited to (1) a review of the security procedures and practices, (2) system testing and maintenance programs, or (3) local law enforcement assistance agreements. The Quality Assurance and Audit Staff prepares a report documenting their findings and recommendtions and submits it to the Director of Nuclear Power and the plant superintendent.

13.6.8 Response Requirements

In meeting the requirements of 10 CFR 73.55(h) TVA has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations. In addition, to provide additional response support in the event of security events, liaison with local law enforcement authorities has been established and documented.

The applicants' safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR 73, Appendix C. The plan identifies appropriate security events that could initiate a radiological sabotage event and it identifies TVA's preplanning, response resources, safeguards contingency participants, and coordination activities for each identified event. Through this plan, response activities using the available resources would be initiated upon the detection of abnormal presence or activities within the protected or vital areas. The response activities and objectives include the neutralization of the existing threat by (1) a requirement that the response force members interpose themselves between the adversary and their objective, (2) instructions to use force commensurate with that used by the adversary, and (3) authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities a closed circuit television system, providing the capability to observe the entire protected area perimeter, isolation zones, and a majority of the protected area, is provided to the security organization.

13.6.9 Conclusions

Based on a review of the applicants' security program plans listed in Section 13.6.1 of this report and on visits to the site, the staff has concluded that the protection provided by TVA against radiological sabotage at the Watts Bar Nuclear Plant meets the requirements of 10 CFR 73, provided the license condition discussed above is imposed. Accordingly, the protection provided will ensure that the health and safety of the public will not be endangered.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During its 268th meeting on August 13, 1982, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application of the Tennessee Valley Authority for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. This application also was considered at Subcommittee meetings held on April 30, 1982 in Knoxville, Tennessee, and on August 10, 1982 in Washington, D.C. Members of the Subcommittee toured the facility before the April 30, 1982 meeting. Transcripts of each of these meetings are available from Alderson Reporting, 400 Virginia Avenue, S.W., Washington, D.C. 20024. Copies of transcripts also are available for review at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and at the Chattanooga-Hamilton County Bicentennial Library, 1001 Broad Street, Chattanooga, Tennessee.

A copy of the ACRS Report on the Watts Bar Plant, Units 1 and 2, dated August 16, 1982, is included as Appendix F of this supplement. The report indicates the ACRS' belief that, subject to the satisfactory completion of construction, staffing, and preoperational testing and if due regard is given to the items in the letter, there is reasonable assurance that the Watts Bar Nuclear Plant, Units 1 and 2, can be operated at core power levels up to 3411 MWt without undue risk to the health and safety of the public. A discussion of the items mentioned in the August 16, 1982 letter follows below. Another supplement to the Watts Bar SER will be issued before fuel loading of Unit 1 to provide additional information on those items, as appropriate.

(1) Design and Construction Quality Assurance

The Committee noted that the effects of the breakdown of TVA's quality assurance program in the design and construction areas still persist. Consequently, the Committee has requested to be kept informed of the results of TVA's major quality assurance programmatic changes, including the applicants' plans to have an independent contractor review the design and construction of a typical "vertical section" of the plant to confirm the adequacy and safety of the as-completed plant.

Region II has just completed an inspection effort (see Inspection Report 50-390/82-05) devoted to determining how responsive the staff of the TVA Office of Engineering Design and Construction (OEDC) have been in resolving TVA and NRC audit findings. The inspection effort entailed review of the TVA Nuclear Safety Review Staff (NSRS), OEDC, QA, and NRC inspection report findings related to OEDC activities. In addition, followup inspections were performed to determine the extent of problem identification, documentation, investigation, evaluation, corrective action, and implementation of the corrective action.

The TVA audit group findings indicated that several management control programs were not sufficiently adequate to ensure that requirements or commitments of the QA program would be met. The majority of the problems were judged by TVA to be programmatic. The most serious of the deficiencies were found in the areas of QA program requirements, QA program applicability, engineering procedures for the control of the design process and changes to the design, interface control, and construction activity planning. TVA findings indicate that implementation deficiencies also were identified; however, these specifically identified problems were not considered by the audit group to be as serious as those involving the program. The OEDC has developed, during 1981, an extensive 1982 action plan for quality improvement as a direct result of the problems identified by the TVA audit groups and the NRC.

TVA's 1982 Action Plan is a program to perform an aggressive, indepth investigation and evaluation of the identified problem areas. The plan is designed to focus corrective action on the root causes of the problems that have adversely affected the quality assurance program. The TVA Action Plan consists of separate tasks for the OEDC QA staff, Division of Engineering and Design (ENDES), and Division of Construction (CONST). Certain root causes of inadequacies in the problem areas have been identified in the following categories: (1) positive attitude and approach, (2) authority and responsibility, (3) timeliness and responsiveness, (4) procedures, (5) commitment control and requirements definition, and (6) retaining experienced personnel. Each group having an assigned task in the Action Plan (OEDC QA, ENDES, CONST) has developed a master schedule to provide traceability relating to each root cause, action element, and time schedule. Much of the 1982 Action Plan activities remain in the investigative and evaluation stages. All indications are that the investigation, evaluation, and determination of corrective action activities will be completed by late fall of 1982. Implementation of most corrective action elements should be started by the first of 1983.

The staff has concluded that the past 12-month effort by OEDC to investigate and evaluate the problem areas, identify root causes, develop the TVA Action Plan, and determine the course for corrective action has been a very aggressive and monumental task. Further review of these areas will continue during future NRC inspections.

Based on its evaluation of the TVA audit findings and the extensive ongoing effort associated with the TVA Action Plan, the staff has concluded that the plan is comprehensive and commendable and, when completed and implemented, should result in a well defined interdivisional program that will provide reasonable assurance for both the adequacy of design and construction of future TVA facilities. However, with respect to Watts Bar, the staff's review of both the internal audit findings and TVA Action Plan Task Group findings, to date, as well as previously identified NRC concerns, has reinforced the conclusion that programmatic deficiencies did and possibly still do exist to varying degrees in the following areas: (1) design control, (2) transmission of design information to construction, (3) the construction process, and (4) the ability of QA to verify the adequacy of the as-built product. The staff does not believe the corrective actions associated with the TVA Action Plan alone, when implemented, will provide the necessary assurance that the as-built product at Watts Bar is actually in accordance with the specified design requirements. Therefore, the staff has concluded that further design verification of the as-built conditions at the Watts Bar facility is necessary.

In Section 1.1 of the SER, the staff stated that TVA proposed to have the Institute of Nuclear Power Operations (INPO) perform the audit, but the staff reserved judgment as to the acceptability of this proposal. During the August 13, 1982 ACRS Committee meeting, TVA stated they no longer believed that the INPO review program was appropriate for use on the Watts Bar facility.

By letter dated September 9, 1982 and during a September 17, 1982 meeting with the staff, TVA committed to having an independent organization (Black and Veatch) perform an audit on the safety-related auxiliary feedwater system of the Watts Bar facility to ensure that the system has been designed and constructed in accordance with the license application and license commitments. The review is intended to provide a comprehensive assessment of TVA's design and construction activities at the Watts Bar facility and to provide confirmation of the adequacy of the system. This independent design verification will be compiled with broader, more comprehensive programmatic reviews (e.g., NSRS Review of Watts Bar, United Engineers Design Verfication Program Review, and Theodore Barry and Associates Review of OEDC) to verify that the Watts Bar facility is designed and constructed adequately even though deficiencies in the QA program have been identified.

The auxiliary feedwater system was chosen by TVA to be a representative system of the facility because

(1) it was essential to plant nuclear safety

(2) it was primarily of TVA's design

(3) there was a clearly defined design basis

(4) it was representative of safety-related equipment

(5) the design and construction of the system required interfacing with several disciplines

(6) essential areas have been completed at the plant site

(7) it is moderately complex both physically and functionally

The staff considers these criteria to be reasonable and the selection of the auxiliary feedwater system to be appropriate for this aspect of the design review. The independent design verification is expected to be completed by Black and Veatch by the end of December 1982. Upon receipt and review of this information, the staff will address this matter in a future supplement to the SER.

(2) <u>Model D-3 Steam Generator Preheater Tube Degradation</u>

The Committee noted that the Watts Bar units have Westinghouse Model D-3 steam generators. As discussed in Section 5.4.2.2 of the SER, steam generators of this design have experienced tube degradation in the preheater region. The Committee indicated its desire to be kept informed of the steps being taken to resolve this problem.

The present status of this open item is discussed in Section 5.4.2.2 of this supplement. New developments will be addressed in a future supplement to the SER.

(3) Surveillance Requirements for ERCW Cement Mortar Lining

The Committee recommended that periodic inspections and tests of the cement mortar lining in the essential raw cooling water (ERCW) pipeline be carried out to ensure the ERCW system will not be subject to sudden entrainment of debris resulting from the deterioration of the lining.

Discussion of the seismic qualification of the lining can be found in Section 9.2.1 of the SER.

The staff currently is assessing the need for TVA to develop these surveillance requirements. The staff will address this matter in a future supplement to the SER.

(4) Hydrogen Detection and Mitigation System

The Committee noted that they expect to review the Sequoyah and Watts Bar hydrogen mitigation systems in the near future. By letters dated April 23, 1982 and June 14, 1982, TVA presented additional information regarding their research program on hydrogen control. TVA has selected a 120-V ac thermal ignitor to be installed in place of the 12-V ignitor system currently installed in the Sequoyah facility. Staff review of the information received is expected to be completed by November 1982, at which time a supplement to the Sequoyah SER (NUREG-0011) will be issued. TVA intends to use the same mitigation system in both the Sequoyah and Watts Bar facilities. As stated in Appendix C of the SER, the staff has requested additional justification for TVA's contention that the work performed on this matter for Sequoyah is directly applicable to Watts Bar. The staff will consider the additional justification in its continuing review of this issue for the Sequoyah and Watts Bar facilities.

The Committee recommended that specific attention be given to ensuring the reliability of the hydrogen monitors used in conjunction with this system. The regulatory requirement for providing hydrogen monitors is prescribed in 10 CFR 50.44. The staff's recommended performance features for the hydrogen monitors are contained in Regulatory Guide 1.97. Additional guidance is provided in Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

Staff reviews for compliance with Item II.F.1 are completed in conjunction with the operating license (OL) reviews for all new plants. These same reviews are scheduled for completion by December 1982 for all the operating plants. The staff's evaluation of this item for Watts Bar Units 1 and 2 appears in Section 6.2.6 of the SER.

The ice condenser containments for the Watts Bar facility will be equipped with a distributed ignition system to deal with the hydrogen releases associated with postulated degraded core accidents. The distributed ignition system will be manually actuated by the operator upon receipt of a safety injection signal. The hydrogen monitors will provide data for use in management of the plant following the onset of an accident and for use in postaccident analysis efforts. However, these data are not essential for the operator's decision to actuate the distributed ignition system nor are they essential for the successful performance of the system.

The staff has had a program to evaluate the performance of hydrogen monitors for some time. The staff believes that there is no technological problem that might prevent compliance with the requirements. Recent licensee event reports (LERs) indicate that some licensees are experiencing difficulties with their hydrogen monitoring system. The staff will review these reports to determine whether any change to NRC licensing requirements is warranted. If such a change is needed, the staff will report on this matter in a supplement to the SER.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNITS 1 and 2, OPERATING LICENSE REVIEW

June 15, 1982	Generic Letter 82-12 -- Nuclear Power Plant Staff Working Hours.
June 17, 1982	Generic Letter 82-13 -- Reactor Operator and Senior Reactor Operator Examinations.
June 21, 1982	Letter from applicant concerning submergence of electrical equipment.
June 21, 1982	Letter from applicant concerning options for compensation of the steam generator reference leg.
June 23, 1982	Letter from applicant concerning buried seismic Category I piping systems and tunnels.
June 23, 1982	Letter from applicant concerning askarel-insulated transformers.
June 24, 1982	Letter to applicant concerning revised SER regarding the boron dilution event concern.
June 30-July 1, 1982	Meeting with applicant to review geotechnical engineering information.
July 9, 1982	SER issued.
July 9, 1982	Letter from applicant concerning compliance with Appendix R.
July 12-15, 1982	Meeting with applicant to confirm the findings of the Power Systems Branch in SER.
July 19, 1982	Letter from applicant concerning technical instructions related to the leakage reduction program.
July 19, 1982	Letter from applicant forwarding generic security training and qualification plan.
July 20, 1982	Letter from applicant concerning implementation of the requirements of 10 CFR 73.21.

July 21, 1982	Letter to applicant forwarding flow diagrams for the pump and valve inservice testing program.
July 26, 1982	Letter from applicant concerning proposed modifications to the draft radiological effluent technical specifications.
July 29, 1982	Letter from applicant concerning the Physical Security Plan.
July 30, 1982	Letter from applicant concerning the preservice inspection program.
July 30, 1982	Letter to applicant concerning loose parts monitoring program.
August 9, 1982	Generic Letter 82-14 -- Submittal of Documents to the Nuclear Regulatory Commission.
August 23, 1982	Letter to applicant transmitting ACRS Letter of August 16, 1982.
August 25, 1982	Letter from applicant concerning TVA schedule for completion of the evaluation for control of heavy loads.
August 31, 1982	Letter from applicant concerning schedule for process control program for radioactive waste solidification.
September 9, 1982	Letter from applicant concerning TVA's proposal to perform an independent design review for the Watts Bar facility.
September 17, 1982	Meeting with applicant to discuss TVA proposal for the QA independent design verification.

APPENDIX B

BIBLIOGRAPHY

Tennessee Valley Authority, Final Safety Analysis Report for Watts Bar Nuclear Plant, Units 1 and 2, October 4, 1976.

U.S. Nuclear Regulatory Commission, NUREG-0011, "Safety Evaluation Related to Operation of Sequoyah Nuclear Plants, Units 1 and 2," March 1979.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

---, NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants---LWR Edition" (includes Branch Technical Positions), July 1981.

---, NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," June 1982.

---, Region II Inspection Report 50-390/82-05.

---, Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Rev. 1, February 1976.

---, Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," Rev. 1, August 1977.

APPENDIX E

PRINCIPAL CONTRIBUTORS

NRC Personnel	Branch
M. Duncan	Licensing Branch No. 4
C. Gaskin	Physical Security Licensing Branch
J. Pulsipher	Containment Systems Branch
D. Quick	Region II
G. Bagchi	Equipment Qualification Branch

Appendix F

ACRS REPORT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 16, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

During its 268th meeting, August 12-14, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of the Tennessee Valley Authority (TVA) for authorization to operate the Watts Bar Nuclear Plant, Units 1 and 2. The project was considered at ACRS Subcommittee meetings in Knoxville, Tennessee on April 30, 1982, and in Washington, D.C. on August 10, 1982. Members of the Subcommittee toured the facility on April 30, 1982. In its review, the Committee had the benefit of discussions with representatives of TVA, Westinghouse Electric Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for the Watts Bar Nuclear Plant in a report dated September 21, 1972.

The Watts Bar Nuclear Plant is located in Rhea County in southeastern Tennessee, about 45 miles north-northeast of Chattanooga, Tennessee. Each of the two identical units uses a Westinghouse nuclear steam supply system with a rated core power of 3411 MWt and has an ice-condenser containment with a design pressure of 15 psig. TVA estimates that Watts Bar Nuclear Plant, Units 1 and 2 will be ready for fuel loading by August 1983 and August 1984, respectively.

A number of items have been identified by the NRC Staff as Outstanding Issues, Confirmatory Issues, and License Conditions. These matters should be resolved in a manner satisfactory to the NRC Staff.

Late in the construction program a serious quality assurance breakdown was identified - principally in the construction area, but also in the design area. The effects of the breakdown persist, and corrective work on the plant will continue at least throughout 1982. TVA invoked major quality assurance programmatic changes, including plans to have an independent contractor review the design and construction of a typical "vertical section" of the plant, to confirm the adequacy and safety of the as-completed plant. This issue should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

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Both Watts Bar Nuclear Plant units have Westinghouse Model D-3 steam generators. Steam generators of this design have experienced tube failures, apparently related to flow-induced vibrations in the preheater region. TVA has stated that this problem is being worked on by Westinghouse and that a resolution involving internal modifications is expected before the projected fuel load date for Unit 1. We wish to be kept informed.

TVA is using a cement mortar lining in the essential raw cooling water system piping to reduce the pressure drop from corrosion-induced roughness. We believe that periodic inspections and tests of this lined piping should be carried out so that, if the bonding or quality of the coating should unduly deteriorate, the system will not be subject to sudden entrainment of debris.

TVA is developing a hydrogen ignition system using controlled distributed ignition sources. The system to be used at the Watts Bar Plant will be of the same design as the permanent system to be installed at the Sequoyah Nuclear Plant. We expect to review that system in the near future. We recommend that specific attention be given by the NRC Staff to assuring the reliability of the hydrogen monitors used in conjunction with this system. Acceptability of this system has been designated as a License Condition by the NRC Staff.

The ACRS believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Watts Bar Nuclear Plant, Units 1 and 2 can be operated at core power levels up to 3411 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS member D. Okrent are presented below.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

<u>Additional Comments by ACRS Member D. Okrent</u>

With regard to the seismic design, I recommend that TVA and the NRC Staff conduct studies to evaluate the margins available to accomplish safe shutdown, including long-term heat removal, following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake. I believe it is important that there be considerable assurance that the combination of seismic design basis and margins in the seismic design is such that this accident source represents an acceptably low contribution to the overall risk from this plant.

References:

1. Tennessee Valley Authority, "Watts Bar Nuclear Plant Final Safety Analysis Report," with Amendments 1-46.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," NUREG-0847, dated June 1982.

APPENDIX G

ERRATA TO WATTS BAR SAFETY EVALUATION REPORT

Section	Page	Change
1.3	1-8	Change quantity listed for "containment volume, ft^3" for Watts Bar to 1,191,500
1.3	1-8	Change quantities listed for "Total reactor coolant flow-rate, lb/hr" to 144,800,000 for Watts Bar, 138,100,000 for Sequoyah, and 144,800,000 for McGuire
1.3	1-8	Change quantity listed for "Secondary steam flow-rate, lb/hr" for Sequoyah to 15,140,000
1.7	1-12	In Item (10), change "(6.3)" to "(5.2.2)"
1.8	1-13	Add "(40) Submergence of electrical equipment as result of LOCA (8.3.3.1.1)"
1.9	1-14	In Item (26), change "(13.1.2)" to "(13.1.3)"
4.4.6	4-21	Change quantity listed for "Coolant flow-total flow rate (10^6 lb/hr)" for Sequoyah to 144.8
6.2	6-4	In third paragraph, change "1,191,000 ft^3" to "1,191,500 ft^3" and "110,400 ft^3" to "110,500 ft^3"
6.2	6-5	Change quantities listed for "Ice Condenser" Volume to 110,500 and "Total Containment Volume (ft^3)" to 1,191,500

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BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)* NUREG-0847 Supplement No. 1	
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)* Safety Evaluation Report related to operation of the Watts Bar Nuclear Plant, Units 1 and 2	2. *(Leave blank)*
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH September YEAR 1982
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* U. S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Washington, D. C. 20555	DATE REPORT ISSUED MONTH September YEAR 1982
	6. *(Leave blank)*
	8. *(Leave blank)*
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Same as 9 above	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO.
13. TYPE OF REPORT	PERIOD COVERED *(Inclusive dates)*
15. SUPPLEMENTARY NOTES Docket Nos. 50-390 and 50-391	14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

This report supplements the Safety Evaluation Report, NUREG-0847, issued June 1982 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the open items identified in the Safety Evaluation Report and discusses recommendations of the Advisory Committee on Reactor Safeguards in its report dated August 16, 1982.

17. KEY WORDS AND DOCUMENT ANALYSIS

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