NUREG-0847
Supplement No. 2

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

January 1984

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ABSTRACT

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982) and Supplement No. 1 (September 1982), issued 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 and confirmatory items and license conditions identified in the Safety Evaluation Report.

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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. The Safety Evaluation Report (SER) was supplemented by Supplement No. 1 (September 1982), which discussed the status of some outstanding issues in further support of the licensing activities and addressed the recommendations of the Advisory Committee on Reactor Safeguards (ACRS).

This supplement provides more recent information regarding resolution of some of the open and confirmatory items and license conditions identified in the SER and its supplement. 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 and license conditions 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 G is a list of errata for the SER. No changes in SER Appendices C, D, and F have been made by this supplement.

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1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues that had not been resolved at the time the SER was issued. This supplement updates the status of four of those items. The current status of each of the 17 original issues is tabulated below. For those items discussed in this supplement, the relevant section of this document is indicated. Resolution of those issues that are, to date, unresolved will be addressed in future supplements.

*Avaliability of all material cited is described on the inside front cover of this report.

Issue	Status	Section
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Awaiting information	--
(2) Buckling loads on Class 2 and 3 supports	Under review	--
(3) Preservice and inservice pump and valve test program	Under review	--
(4) Seismic and environmental qualification of equipment	Seismic - under review Environmental-awaiting information	-- --
(5) Preservice and inservice inspection program	Under review	--
(6) Pressure-temperature limits for Unit 2	Awaiting information	--
(7) Model D-3 steam generator preheater tube degradation	Under review	--
(8) BTP CSB 6-4 and containment isolation dependability (II.E.4.2)	Under review	--
(9) H_2 analysis review	Awaiting information	--
(10) Safety valve sizing analysis (WCAP-7769)	Resolved	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDC-17 and 18	Partially resolved	8.2.2.2
(12) Fire Protection Program	Awaiting information	--
(13) Quality classification of diesel generator auxiliary system piping and components	Under review	--
(14) Diesel generator auxiliary system design deficiencies	Under review	--
(15) Physical Security Plan	Resolved in SSER 1*	--
(16) Boron Dilution Event	Under review	--
(17) Q List	Resolved	17

*TVA has recently submitted a revised Physical Security Plan. However, the plan approved in SSER 1 is acceptable for use pending approval of the new plan.

1.8 Confirmatory Issues

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. This supplement updates 15 of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The current status of each of the original issues is tabulated below. For those items discussed in this supplement, the relevant section of this supplement is noted. Resolution of issues that are outstanding, to date, will be addressed in future supplements.

Issue	Status	Section
(1) Design basis ground water level for the ERCW pipeline	Under review	--
(2) Material and geometric damping effect in SSI analysis	Under review	--
(3) Analysis of sheetpile walls	Under review	--
(4) Design differential settlement of piping and electrical components between rock-supported structures	Under review	--
(5) Upgrading ERCW system to seismic Category I	Under review	--
(6) Seismic classification of structures, systems, and components important to safety	Awaiting information	--
(7) Tornado missile protection of diesel generator exhaust	Resolved	3.5.2
(8) Steel containment building buckling research program	Awaiting information	--
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02)	Under review	--
(10) Thermal performance analysis	Resolved	4.2.2
(11) Cladding collapse	Resolved	4.2.2
(12) Fuel rod bowing evaluation	Resolved	4.2.3
(13) Loose-parts monitoring system	Awaiting information	--
(14) Installation of residual heat removal flow alarm	Awaiting verification of installation	--
(15) Natural circulation tests	Awaiting information	--

Issue	Status	Section
(16) Dump valve testing	Resolved	5.4.3
(17) Protection against damage to containment from external pressure	Under review	--
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines	Awaiting information	--
(19) Compliance with GDC-51	Awaiting information	--
(20) Isolation survey (sump debris)	Resolved	6.3.3
(21) Safety system set point methodology	Awaiting information	--
(22) Steam generator water level reference leg	Resolved	7.2.5
(23) Containment sump level measurement	Resolved	7.3.2
(24) IE Bulletin 80-06	Awaiting information	--
(25) Overpressure protection during low-temperature operation	Awaiting information	--
(26) Availability of offsite circuits	Resolved	8.2.2.1
(27) Nonsafety loads powered from the Class 1E ac distribution system	Resolved	8.3.1.1
(28) Low and/or degraded grid voltage condition	Awaiting verification of test results	8.3.1.2
(29) Diesel generator reliability qualification testing	Under review	--
(30) Diesel generator battery system	Resolved	8.3.2.4
(31) Thermal overload protective bypass	Resolved	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2	Awaiting information	--
(33) Sharing of raceway systems between units	Resolved	8.3.3.2.3
(34) Testing Class 1E power systems	Resolved	8.3.3.5.2
(35) Evaluation of penetrations capability to withstand failure of overcurrent protection device	Awaiting information	--

Issue	Status	Section
(36) Missile protection for diesel generator vent line	Awaiting verification of modifications	--
(37) Component booster pump relocation	Awaiting verification of modifications	--
(38) Electrical penetrations documentation	Under review	--
(39) Compliance with NUREG/CR-0660	See License Condition (22)	--
(40) No load, low-load, and testing operations for diesel generator	Awaiting verification of procedure changes	--
(41) Initial test program	Awaiting information	--
(42) Submergence of electrical equipment as result of a LOCA	Awaiting information	--

1.9 License Conditions

In Section 1.9 of the SER and its supplement, the staff identified 38 license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The license conditions are tabulated below, with the corresponding NUREG-0737 item number given in parentheses and the relevant section of this report noted for the updated status.

Condition	Section
(1) Relief and safety valve testing (II.D.1)	--
(2) Preservice/inservice testing of pumps and valves	--
(3) Detectors for inadequate core cooling (II.F.2)	--
(4) Inservice Inspection Program	--
(5) Installation of reactor coolant vents (II.B.1)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)	
(a) noble gas monitor	--
(b) iodine particulate sampling	--
(c) high range incontainment radiation monitor	--
(d) containment pressure	--
(e) containment water level	--
(f) containment hydrogen	--
(7) Modification to chemical feedlines	--
(8) Containment isolation dependability (II.E.4.2)	--
(9) Hydrogen control measures (II.B.7)	--
(10) Status monitoring system	--
(11) Installation of acoustic monitoring system (II.D.3)	--
(12) Diesel generator reliability qualification testing at normal temperatures	8.3.1.6

Condition	Section
(13) DC monitoring and annunciation	--
(14) Possible sharing of dc control power to ac switchgear	--
(15) Testing of associated circuits	--
(16) Testing of non-Class 1E cables	--
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (I.G.1)	--
(18) Testing of reactor coolant pump breakers	8.3.3.6
(19) Postaccident sampling system (II.B.3)	--
(20) Fire Protection Program	--
(21) Performance testing for communications systems	--
(22) Diesel generator reliability	--
(23) Secondary water chemistry monitoring and control program	--
(24) Primary coolant outside containment (III.D.1.1)	--
(25) Independent safety engineering group (I.B.1.2)	--
(26) Use of experienced personnel during startup	--
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2)	--
(28) Review of power ascension test procedures and emergency operating procedures by NSSS-Vendor (I.C.7)	--
(29) Modifications to emergency operating instructions (I.C.8)	--
(30) Report on outage of emergency core cooling system (II.K.3.17)	--
(31) Initial test program	--
(32) Effect of high pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	--
(33) Voiding in the reactor coolant system (II.K.2.17)	--
(34) PORV isolation system (II.K.3.1, II.K.3.2)	--
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	--
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31)	--
(37) Control room design review (I.D.1)	--
(38) Physical Security Plan	--

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

3.5 Missile Protection

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

As discussed in the SER, the diesel generator exhaust stacks, which protrude approximately 2 ft above the roof grade, may be susceptible to damage or incapacitating flow blockage as a result of tornado-missile impact. By letter dated November 24, 1982, the applicant submitted the details of proposed design modifications, which consist of installation of a reinforced concrete curb around the diesel exhaust stacks, to prevent such damage. The staff has reviewed this information and has concluded that the proposed design is acceptable. The applicant has committed to completing the modifications to the exhaust stacks before fuel loading; the staff finds this acceptable. This item, therefore, is closed.

4 REACTOR

4.2 Fuel System Design

4.2.2 Thermal Performance

Thermal Performance Analysis

A Westinghouse fuel thermal performance code known as PAD-3.1, described in attachments to correspondence from Westinghouse to the U.S. Atomic Energy Commission (letters dated December 22 and 29, 1972, and January 1 and 12, 1973), was initially used for the Watts Bar safety analysis. A more recent Westinghouse fuel thermal performance code known as PAD-3.3 (Westinghouse Topical Report WCAP-8720) has also been approved by the NRC. The more recent code, which contains revised models for fission gas release, helium solubility, fuel swelling, and fuel densification, was not used in the applicant's original submittal.

The PAD-3.3 code addresses a concern about enhanced fission gas release at high burnup. Because the earlier version of the code did not contain the models necessary to analyze this effect, the staff's safety evaluation of PAD-3.3 stated that future fuel performance analyses must be done with the revised version of the code (PAD-3.3) (letter dated February 9, 1979).

The use of the PAD-3.1 code is generally acceptable because the earlier code produces more conservative thermal conditions than the revised code. This margin, however, does not exist for high burnup fission gas release and fuel rod internal pressure calculations. In a letter dated September 22, 1981, the applicant stated that the more recent code, PAD-3.3, is now used to analyze the fuel thermal performance (including fission gas release and rod internal pressure) at the Watts Bar facility. On the basis of the applicant's use of the revised model, the staff now concludes that the Watts Bar fuel performance analysis is acceptable.

In addition, revised internal fuel rod pressure criteria, as described in Westinghouse Topical Report WCAP-8963-A, are also used now in the Watts Bar safety analysis. The approved criteria are:

(1) The internal pressure is limited so that the fuel-to-cladding gap does not increase during normal operation.

(2) Extensive departure from nucleate boiling propagation does not occur during postulated transients and accidents.

The Watts Bar Final Safety Analysis Report (FSAR) has been amended (Amendment 34) to incorporate these criteria. The staff finds this acceptable. This issue, therefore, is closed.

Cladding Collapse

The staff has reviewed Westinghouse Topical Report WCAP-8377, which describes the details of a revised cladding-flattening model that, for a given fuel region, predicts initial flattening (collapse) time for pressurized rods containing relatively stable fuel. This revised analysis was based on the results of TV examinations of irradiated fuel rods, and the results indicated that the original flattening model (Westinghouse Topical Report WCAP-7982) significantly underpredicted the time of collapse. The revised model was accepted for use in plant safety analysis subject to provisions specified in the SER (memorandum dated January 14, 1975) that did not permit alterations to the specified curves used as input to the model. In a letter dated September 22, 1981, the applicant stated that the revised cladding collapse model is used in a manner consistent with conditions given in the staff's safety evaluation of WCAP-8377. The predicted cladding collapse time for the most limiting Watts Bar fuel has been calculated in this manner and has been shown to be in excess of 38,000 effective full-power hours of operation, which is greater than the expected residence time of the fuel. The staff finds this acceptable. This issue, therefore, is closed.

4.2.3 Mechanical Performance

Fuel Rod Bowing

The consideration of fuel rod bowing in the Watts Bar 17 x 17 fuel design was initially analyzed by Westinghouse in Topical Report WCAP-8346. Subsequently, Westinghouse revised that analysis in light of new information and documented the results in Topical Report WCAP-8692. A revision of this latter report has been reviewed and approved by the NRC. The Watts Bar FSAR has been amended (Amendment 31) to reflect the revised methodology. The staff concludes that the applicant has presented an acceptable means of analyzing the effects of fuel rod bowing and determining any residual rod bowing penalties on the departure from nucleate boiling ratio and total peaking factor for Watts Bar Units 1 and 2. This issue, therefore, is closed.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

The applicant referenced WCAP-7769 as part of the analytical basis for safety valve sizing. In the SER the staff requested additional information on the differences between the Watts Bar Nuclear Plant and the reference plant for WCAP-7769. The staff also requested verification that adequate protection is provided if a reactor trip is initiated by the second safety-grade signal from the reactor protection system as required by the Standard Review Plan (SRP, NUREG-0800). This information has been provided by the applicant by letter dated April 18, 1983. The only significant difference between Watts Bar and the reference plant of WCAP-7769 is that, following a reactor trip signal, the shutdown rods may enter the reactor core with a 2-sec delay at Watts Bar rather than the 1-sec delay assumed for the reference plant. The design-basis overpressure transient is a combined loss of main feedwater and turbine trip. The transient would generate reactor trip signals for each of the following conditions:

(1) turbine trip
(2) high reactor system pressure
(3) high temperature difference across the core
(4) loss of feedwater
(5) low steam generator level with low steam flow
(6) low-low steam generator level

Receipt of only one of these trip signals by the reactor protection system will trip the reactor. WCAP-7769 demonstrated that small delays in reactor trip time would not affect the peak reactor system pressure because the reactivity loss from core heatup would be adequate to limit the initial pressure peak from this event. The analyses indicated that no trip was required for the first 25 sec although reactor trip would eventually be required to compensate for the reduction in heat sink. The staff concludes that the safety valves at Watts Bar are adequately sized and that this issue is resolved.

5.4 Component and Subsystem Design

5.4.3 Residual Heat Removal System

The SER stated that the applicant had committed to test manually the atmospheric steam line dump valves. In a letter dated June 16, 1983, the applicant stated that manual actuation of the atmospheric dump valves will not be tested.

The four air-operated atmospheric dump valves at Watts Bar (one per steam generator) are seismic Category I. Electric and air power sources to the valves are safety related. The valves can be operated from the control room. The steam relief capacity of the valves is sufficient so that failure of any one valve to open will not prevent the plant from reaching cold shutdown using the remaining three valves in a reasonable period of time if required. The staff agrees with the applicant that manual actuation testing of the atmospheric relief valves is not necessary. The staff, therefore, concludes that no further action need be taken in this regard and this item is closed.

5.4.5 Reactor Coolant System Vents (II.B.1)

NUREG-0737 requires installation of reactor coolant system (RCS) and reactor vessel head high point vents that are remotely operated from the control room.

As stated in the SER, the applicant committed to install an acceptable RCS vent system before fuel loading. In addition, the applicant committed to use venting guidelines that were being developed by the Westinghouse Owners Group. In the SER the staff concluded these commitments were acceptable but stated that modification of the venting guidelines would be required if they were indicated by the staff's generic review. This review has been completed as documented in NRC Generic Letter 83-22, dated June 3, 1983. The staff concluded that the guidelines are acceptable for implementation. The staff, therefore, concludes that a license condition regarding the modification of the venting guidelines is not necessary and finds this item acceptable pending verification that the RCS vent system is installed.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Testing

To ensure that debris following a loss-of-coolant accident will not compromise the performance of the emergency core cooling system by clogging the sump, the staff asked the applicant to perform a detailed survey of insulation materials used within the containment. The applicant provided this information in a letter dated November 23, 1982. This survey confirms the staff's initial conclusion that the Watts Bar design to provide protection against sump debris is acceptable. The reactor system and main steam piping and components are encased in metal reflective insulation that, if dislocated by a major pipe rupture, would not form small debris particles that would clog the sump screens. Other materials (foam glass, Rubatex, fiberglass, polyurethane foam, urethane foam, and mineral wool) are either encapsulated in steel or located in areas of the containment where they would be unaffected by pipe rupture forces. The staff concludes that the Watts Bar design regarding protection against sump debris is acceptable and this issue, therefore, is closed.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.5 Steam Generator Water Level Trip

The water level measurement channels use differential pressure transmitters. The measurement accuracy of such a system is affected by several factors. Of primary importance is the increase in the indicated water level caused by a decrease of the water density in the reference leg. This can occur because of an increase in the ambient temperature as a result of a high-energy-line break. This issue was identified in the SER as a matter to be reviewed by the staff. In a letter dated June 21, 1982, the applicant committed to insulate the reference leg to alleviate the temperature-dependence problem. The staff finds this acceptable. The staff will review the trip setpoint for the steam generator level trip function during the Technical Specification review. The staff, therefore, considers this item closed.

7.3 Engineered Safety Features Actuation System

7.3.2 Containment Sump Level Measurement

The containment sump level is monitored by four level measurement channels using differential pressure transmitters. The staff identified a concern in the SER that debris in the sump could block the inlets to the differential pressure transmitters and result in a loss of the permissive signal to the initiation logic for the automatic switchover from the injection to the recirculation mode of the emergency core cooling system. In a letter dated September 15, 1983, the applicant stated that the sump level sensors have been moved from inside the sump wall to outside the sump wall. Sump water level is detected by the sensor by means of a 1-in. sense line routed through the sump wall. The sense line opening is protected by a 2-in.-diameter cap with ten ¼-in. holes to prevent large pieces of debris from blocking the opening in the sense line. On the basis of this action, the staff considers this item closed.

8 ELECTRIC POWER SYSTEMS

8.2 Offsite Electric Power System

8.2.2 Compliance With GDC 17

8.2.2.1 Availability of Offsite Power Circuits

In the SER the staff concluded that the offsite power system circuits at the Watts Bar Hydro Plant Switchyard meet GDC 17 and are acceptable pending documentation in the FSAR of the additional information submitted by letter dated October 9, 1981. The applicant documented the additional information in FSAR Amendment 48. This item, therefore, is closed.

8.2.2.2 Minimizing the Probability of Losing all AC Power

In the SER the staff expressed the concern that the automatic transfer of loads from the normal power source to the various other offsite sources could cause overloading and loss of both offsite circuits. The applicant, by Amendment 48 to the FSAR, proposed a system design change. The proposed design (1) retains provisions for automatic transfer of safety loads between the various offsite power sources, (2) retains provisions so that a faulted or overloaded bus will not be automatically transferred, and (3) adds two new common station transformers. The greater capacity of these new transformers (dedicated to supplying safety-related buses and their associated loads) eliminates the original overloading concern. This item, therefore, is closed.

The staff also indicated in SER Section 8.2.3 that information had not been provided describing the capability to test these transfers during normal power operation. By Amendment 48 to the FSAR, the applicant indicated that these transfers will not be tested during normal power operation because such transfers could result in transients that could cause tripping of the reactor or turbine. Therefore, the staff concludes that the design meets GDC 18 and is acceptable.

Testing requirements for the automatic transfers and the design, which prevents a faulted or overloaded bus from being automatically transferred, will be reviewed with the Technical Specifications.

8.3 Onsite Power Systems

8.3.1 Onsite AC Power System Compliance With GDC 17

8.3.1.1 Nonsafety Loads Powered From the Class 1E AC Distribution System

In regard to the capability of the offsite power system, the staff concluded in the SER that the proposed new design for the offsite power system was acceptable pending its incorporation in the FSAR. The applicant by Amendment 48 incorporated the new design in the FSAR. This item, therefore, is closed.

8.3.1.2 Low and/or Degraded Grid Voltage Condition

In the SER the staff indicated that the design for the low and/or degraded grid voltage condition was acceptable pending verification of design implementation. During a site visit on July 12-14, 1982, the staff reviewed Drawing 45W760-211-17, Revision 2. On the basis of the review of this drawing, the staff confirmed the design implementation. This item, therefore, is closed.

The staff also indicated in the SER that the voltage drop analysis and testing would be verified. During a site visit on July 12-14, 1982, the staff verified the analysis and found the results satisfactory. The test results that substantiate the analysis will be verified by the staff. If any problem areas are identified, they will be reported in a future supplement to the SER.

8.3.1.6 Diesel Generator Reliability Qualification Testing

In the SER the staff required, as a condition to the license, that the capability of the diesel generator to start at normal operating temperature be demonstrated before fuel loading. In a letter dated August 31, 1983, the applicant documented that the capability has been demonstrated by a test on a diesel generator identical to those used at Watts Bar. On the basis of this test, the staff considers this item resolved and no longer a license condition.

The staff also indicated that the diesel generator qualification testing would be verified. If any problem areas are identified as a result of the staff verification, they will be reported in a future supplement to the SER.

8.3.2 Onsite DC System Compliance With GDC 17

8.3.2.4 Diesel Generator Battery System

In the SER the staff indicated that the design analysis for demonstrating compliance of the diesel generator with regulatory requirements and guidelines was acceptable pending the incorporation of the analysis in the FSAR. By Amendment 48, the applicant incorporated the analysis in the FSAR. This item, therefore, is closed.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDC 2 and 4

8.3.3.1.2 Thermal-Overload Protection Bypass

In the SER the staff indicated that the design for bypass of thermal-overload protective devices on safety-related motor-operated valves would be verified during the electrical drawing review. During a site visit on July 12-14, 1982, the staff reviewed Drawings 45W760-62-3 (Revision 7) and 45W760-270-2 (Revision 8). On the basis of the review of these drawings, the staff verified the design. This item, therefore, is closed.

8.3.3.2 Compliance With GDC 5

8.3.3.2.3 Sharing of Raceway Systems Between Units

In the SER the staff indicated that the design for sharing of raceway systems between units would be confirmed as part of the drawing review/site visit.

During a site visit on July 12-14, 1982, the staff traced the following cable routings:

(1) 6,900-V 1B and 2B train cables between the diesel generator and the switchgear

(2) 6,900- to 480-V 2A train cables between their respective switchgear

(3) cables associated with the A and B train turbine and motor-driven auxiliary feedwater pump

On the basis of the cable tracing, the staff concludes that cables associated with the A train of Units 1 and 2 are extensively routed in close proximity and in the same raceway. Similarly, cables associated with the B train of Units 1 and 2 are routed together. A and B cables are routed in physically separate raceways. This cable routing is in accordance with the accepted Watts Bar separation criteria. This item, therefore, is closed.

8.3.3.5 Compliance With GDC 18

8.3.3.5.2 Testing of One of Two Class 1E Power Systems Versus One of Four Systems

In the SER the staff indicated that the commitment for testing one of four diesel generators at any one time was acceptable pending documentation of this commitment in the FSAR. By Amendment 48 to the FSAR, the applicant provided the required documentation. This item, therefore, is closed.

8.3.3.6 Compliance With GDC 50

In the SER the staff required, as a condition to the license, that redundant fault current protective devices be provided in series for the reactor coolant pump circuits in accordance with Position 1 of Regulatory Guide (RG) 1.63. By Amendment 48 to the FSAR, the applicant documented that the design for reactor coolant pump penetration protection would contain the required redundant circuit breakers. The proposed design meets Position 1 of RG 1.63 and is acceptable. This item, therefore, is resolved and is no longer a license condition.

17 QUALITY ASSURANCE

The staff review of the description of the Watts Bar Nuclear Plant quality assurance (QA) program for the operations phase has verified that the criteria of Appendix B to 10 CFR 50 have been adequately addressed in Section 17.2 of the FSAR through Amendment 48. This determination of acceptability included a review of the list of items to which the QA program applies.

In the SER the staff stated that this list had not been approved by the staff. Since then, the list of items has been reviewed by the staff to ensure that safety-related items within the scope of staff review are under the QA program controls. Differences between the staff and the applicant regarding the list have been resolved to the staff's satisfaction. The list has been expanded to include safety-related items reflected in NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. Therefore, the staff has no open items concerning the QA program for operations or the items to which the program applies.

On the basis of its review and evaluation of the QA program description in FSAR Section 17.2, the staff concludes:

(1) The QA organizations of TVA are provided sufficient independence from cost and schedule (when opposed to safety considerations) and have sufficient authority to carry out effectively the operations QA program and sufficient access to management at a level necessary to perform their QA functions.

(2) The QA program description contains adequate QA requirements and a comprehensive system of planned and systematic controls that address each of the criteria of Appendix B to 10 CFR 50 in an acceptable manner. This QA program description, therefore, can serve as an adequate basis for the development of specific policies and procedures to implement the QA responsibilities of TVA for the operation of the Watts Bar Nuclear Plant.

Accordingly, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations and that this item is closed.

APPENDIX A

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

September 9, 1982	Letter from applicant requesting extension of construction permit for Unit 1 to March 1984 and Unit 2 to August 1985.
September 15, 1982	Letter from applicant forwarding proposed modifications to the NRC draft version of the Technical Specifications.
September 21, 1982	Letter from applicant concerning response to NRC question Q22.70.
September 21, 1982	Letter from applicant concerning cement mortar lining.
September 21, 1982	Letter from applicant concerning single-failure criteria for a boron dilution event.
September 22, 1982	Letter from applicant concerning Radiological Effluent Technical Specifications.
September 22-23, 1982	Meeting with applicant to perform an audit of geotechnical engineering documents to confirm conclusions reached during licensing review. (Summary issued November 12, 1982).
September 23, 1982	Letter to applicant concerning seismic and dynamic qualification review of safety-related equipment.
September 29, 1982	Supplement 1 to Safety Evaluation Report (SER) issued.
September 29, 1982	Letter from applicant concerning preservice inspection program.
October 1, 1982	Letter to applicant concerning inconsistency between requirements of 10 CFR 50.54(t) and Standard Technical Specifications for performing audits of emergency preparedness programs (Generic Letter 82-17).
October 5, 1982	Letter to applicant requesting additional information regarding Item II.E.4.2 of NUREG-0737.
October 6, 1982	Letter to applicant concerning Technical Specifications for fire protection audits (Generic Letter 82-21).
October 12, 1982	Letter to applicant concerning reactor operator and senior reactor operator requalification examinations (Generic Letter 82-18).

October 19, 1982	Letter to applicant concerning review schedule.
October 18, 1982	Letter from applicant concerning program for the watchtower power block security concept.
October 20, 1982	Letter from applicant forwarding revision to Physical Security Plan.
October 25, 1982	Letter from applicant concerning the critical structures, systems, and components list.
October 26, 1982	Letter to applicant concerning guidance for implementing Standard Review Plan rule (Generic Letter 82-20).
October 28, 1982	Letter from applicant forwarding report, "Site-Specific Top-of-Ground Motions for ERCW Pipeline."
October 30, 1982	Letter to applicant concerning inconsistency between requirements of 10 CFR 73.40(d) and Standard Technical Specifications for performing audits of safeguards contingency plans (Security Plan)(Generic Letter 82-23).
November 2, 1982	Letter to applicant concerning generic security training and qualification plan.
November 2-3, 1982	Meeting with applicant to discuss proposed conceptual changes to the security program.
November 4, 1982	Letter from applicant concerning cement mortar lining of essential raw cooling water (ERCW) piping.
November 9, 1982	Letter from applicant providing schedule for responding to NRC concerns in SER.
November 10, 1982	Letter from applicant concerning sampling program to verify that buckling stresses in axial compression members used in pipe support configurations do not exceed 90% of yield strength when slenderness ratio is less than 30.
November 10, 1982	Letter from applicant concerning the loose parts monitoring system.
November 16, 1982	Letter from applicant forwarding report, "Liquefaction Evaluation of the ERCW Pipeline Route - Watts Bar Nuclear Plant."
November 18, 1982	Letter from applicant concerning buckling of Class 2 and 3 supports.
November 19, 1982	Letter to applicant extending construction completion dates. Unit 1 completion date is now extended to March 1, 1984, and that of Unit 2 to August 1, 1985.
November 23, 1982	Letter from applicant responding to NRC questions regarding an insulation survey.

November 24, 1982	Letter from applicant concerning modifications to protect the diesel generator from degraded operation resulting from a tornado-generated missile impacting the diesel generator exhaust.
November 29, 1982	Letter from applicant concerning auxiliary systems for the diesel generator units.
November 30, 1982	Letter from applicant providing schedules for milestones concerning the prompt notification system.
November 30, 1982	Letter from applicant concerning data availability for each primary meterological measurements system.
November 30, 1982	Letter from applicant concerning geotechnical audit conducted September 22-24, 1982.
December 1, 1982	Letter to applicant requesting additional information regarding Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation."
December 1, 1982	Letter from applicant concerning information requested by the NRC Seismic Qualification Review Team.
December 6, 1982	Letter from applicant concerning plans for emergency operations facility.
December 9, 1982	Letter from applicant forwarding copies of report, "Liquefaction Evaluation of the ERCW Pipeline Route - Watts Bar Nuclear Plant."
December 14, 1982	Letter from applicant providing additional information concerning diesel generators.
December 16, 1982	Letter from applicant providing Revision 3 to ASME Code, Section XI, Preservice Inspection Program Technical Instruction TI-50B.
December 17, 1982	Letter to applicant forwarding Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability (Generic Letter 82-33).
December 20, 1982	Letter from applicant concerning use of the computer card deck containing the time histories of the four artificial earthquakes used for Watts Bar.
December 20, 1982	Letter from applicant providing status report on proposed modifications to the Model D steam generators.
December 22, 1982	Letter to applicant concerning problems with the submittals of 10 CFR 73.21 safeguards information for licensing review (Generic Letter 82-39).

December 22, 1982	Letter to applicant concerning meeting to discuss recent developments for operating licensing examinations (Generic Letter 82-38).
December 28, 1982	Letter from applicant concerning Appendix R review and instrumentation available in the auxiliary control room.
December 28, 1982	Letter to applicant concerning filings relating to 10 CFR 50 production and utilization facilities (Generic Letter 82-30).
December 29, 1982	Letter from applicant concerning the process control program for solidification of radwaste.
January 4, 1983	Letter from applicant submitting Amendment 47 to Final Safety Analysis Report.
January 11, 1983	Letter to applicant concerning operator licensing examination site visit (Generic Letter 83-01).
January 11, 1983	Letter from applicant concerning compliance with Appendix R.
January 17, 1983	Letter from applicant responding to NRC staff questions.
January 17, 1983	Letter from applicant concerning compliance with 10 CFR 50, Appendix R.
January 21, 1983	Letter from applicant concerning soil amplification studies.
January 21, 1983	Presentation by the Design Review Panel of their program to review the Model D2/D3 steam generator modification.
January 24, 1983	Letter to applicant concerning final rulemaking concerning reporting of changes to quality assurance programs for nuclear power plants and fuel reprocessing plants (generic).
January 25, 1983	Letter from applicant providing comments on the SER.
January 26, 1983	Meeting with applicant to discuss Westinghouse's appeal of the staff's position regarding reactor trip breaker testing on Westinghouse plants.
January 31, 1983	Letter to applicant concerning certificates and revised format for reactor operator and senior reactor operator licenses (Generic Letter 83-06).
February 1, 1983	Letter to applicant concerning regional workshops regarding Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability" (Generic Letter 83-04).
February 1, 1983	Letter to applicant concerning safety evaluation of "Emergency Procedure Guidelines, Revision 2," NEDO-24934, June 1982 (Generic Letter 83-05).

February 1, 1983	Letter to applicant concerning resolution of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter 83-10c).
February 3, 1983	Letter from applicant forwarding revision of Physical Security Plan.
February 17, 1983	Letter from applicant forwarding Revision 8 to ASME Code, Section XI, Preservice Inspection Program Technical Instruction TI-50A.
February 22, 1983	Letter to applicant concerning reactor trip breaker test appeal meeting.
February 22, 1983	Letter from applicant forwarding the process control program for ensuring solidification and dewatering.
February 24, 1983	Letter to applicant concerning evaluation of boron dilution event response.
February 24, 1983	Letter to applicant concerning issuance of NRC Form 398 - Personal Qualifications Statement - Licensee (Generic Letter 83-12).
March 1-3, 1983	Caseload forecast meeting at Watts Bar site to update the NRC's projected fuel loading date.
March 2, 1983	Letter to applicant concerning clarification of surveillance requirements for high-efficiency particulate air filters and charcoal adsorber units in Standard Technical Specifications on engineered safety features cleanup systems (Generic Letter 83-13).
March 7, 1983	Letter to applicant concerning definition of "Key Maintenance Personnel" (clarification of Generic Letter 82-12) (Generic Letter 83-14).
March 16, 1983	Letter to applicant forwarding staff evaluation of Utility Design Review Panel Report on modification to Westinghouse D2/D3 steam generators.
March 21, 1983	Meeting with Design Review Panel to discuss information related to the forward flush transient on D2/D3 steam generators.
March 22, 1983	Letter from applicant forwarding draft Technical Specifications.
March 23, 1983	Letter to applicant concerning request for withholding information from public disclosure.
March 23, 1983	Letter to applicant concerning implementation of Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations, Revision 1" (Generic Letter 83-15).

March 24, 1983	Letter to applicant transmitting NUREG-0977 relative to the anticipated transients without scram events at Salem Generating Station, Unit No. 1 (Generic Letter 83-16).
March 25, 1983	Letter to applicant concerning alternate shutdown capabilities at the Watts Bar plant.
March 25, 1983	Letter from applicant concerning analysis of axial stresses on buried pipe.
March 28, 1983	Letter from applicant concerning initial test program.
March 29, 1983	Letter to applicant concerning evaluation of liquefaction potential of the soils beneath the ERCW pipeline and electrical conduits.
April 1, 1983	Letter to applicant concerning environmental qualification of safety-related mechanical equipment located in harsh environment areas.
April 8, 1983	Letter to applicant concerning integrity of the requalification examinations for renewal of reactor operator and senior reactor operator licenses (Generic Letter 83-17).
April 14, 1983	Letter from applicant forwarding copies of TVA Design Criteria WB-DC-10-2 pertaining to the security-power block project.
April 15, 1983	Letter from applicant responding to Generic Letter 82-33 regarding Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability.
April 18, 1983	Letter from applicant forwarding listing of welds to date for which relief is being requested and that have had examination reports reviewed.
April 18, 1983	Letter from applicant concerning applicability of WCAP-7769 to the issue of safety valve sizing.
April 20, 1983	Letter from applicant concerning initial test program.
April 22, 1983	Letter from applicant concerning modifications to the chemical feed lines at the plant.
April 22, 1983	Letter from applicant responding to Generic Letter 83-10c concerning NUREG-0737, Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps."
April 25, 1983	Letter from applicant forwarding a list of safety-related electrical components and their respective functions and a summary of data used to establish set points for balance-of-plant equipment.
April 26, 1983	Letter from applicant concerning Lawrence Livermore National Laboratory's preliminary system interaction results from

	the digraph matrix analysis of the plant's safety injection system.
April 26, 1983	Letter from applicant concerning Branch Technical Position CSB 6-4.
April 29, 1983	Letter from applicant concerning boron dilution event response.
May 9, 1983	Letter to applicant concerning integrated scheduling for implementation of plant modifications (Generic Letter 83-20).
May 10, 1983	Letter from applicant concerning research and development work performed for Sequoyah Nuclear Plant in the area of hydrogen capabilities and its applicability to Watts Bar.
May 11, 1983	Letter to applicant concerning clarification of access control procedures for law enforcement visits (Generic Letter 83-21).
May 11, 1983	Letter from applicant forwarding copies of Black and Veatch final report.
May 17, 1983	Letter from applicant concerning evaluation of potential for liquefaction at plant.
May 18, 1983	Letter to applicant concerning environmental qualification program.
May 19, 1983	Meeting with applicant to discuss proposed application of the watchtower power block concept.
May 20, 1983	Meeting with applicant to discuss measures to prevent liquefaction of the soils beneath the ERCW pipeline. (Summary issued June 8, 1983.)
May 24, 1983	Letter to applicant providing results of NRC Caseload Forecast Panel meeting.
May 24, 1983	Letter to applicant concerning the Power Block Physical Security Plan.
May 27, 1983	Letter from applicant concerning schedule for steam generator modifications.
June 1, 1983	Letter to applicant requesting additional information regarding the geotechnical concerns.
June 3, 1983	Letter to applicant forwarding safety evaluation of Emergency Response Guidelines, (Generic Letter 83-22).
June 7, 1983	Letter from applicant providing revised response to NUREG-0737, Item II.F.1.

June 9, 1983	Letter from applicant responding to Power Systems Branch concerns specified in the SER.
June 9, 1983	Letter from TVA forwarding generic control room design review program plan for Sequoyah, Watts Bar, Bellefonte, and Browns Ferry.
June 16, 1983	Letter from applicant providing updated listing showing status of open and confirmatory items in the SER.
June 22, 1983	Letter to applicant concerning peak horizontal ground acceleration for use in soil liquefaction analysis.
July 1, 1983	Letter from applicant concerning current status and revised implementation schedule for prompt notification system.
July 5, 1983	Letter to applicant concerning clarification of surveillance requirements for diesel fuel impurity level tests (Generic Letter 83-26).
July 6, 1983	Letter to applicant concerning surveillance intervals in Standard Technical Specifications (Generic Letter 83-27).
July 7, 1983	Letter from applicant documenting discussions between NRC and TVA staffs regarding preoperational test programs.
July 8, 1983	Letter to applicant concerning required actions based on generic implications of Salem ATWS events (Generic Letter 83-28).
July 21, 1983	Letter to applicant concerning deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 for diesel generator testing (Generic Letter 83-30).
July 21, 1983	Letter from applicant concerning the analysis of sheetpile walls and seismic analysis of buried pipes.
July 22, 1983	Letter from applicant responding to NUREG-0737, Item II.D.1, regarding performance testing of pressurized-water reactor relief and safety valves.
July 26, 1983	Letter from applicant responding to NUREG-0737, Item II.F.2.
July 27, 1983	Letter from applicant forwarding proposed modifications to the NRC draft version of the Technical Specifications.
July 27, 1983	Letter from applicant concerning requirements for performing a preliminary control room assessment.
August 5, 1983	Letter from applicant responding to NRC question 212.35.
August 8, 1983	Letter from applicant providing revisions to ASME Code, Section XI, Preservice Inspection Program Technical Instructions TI-50A and TI-50B.

August 11, 1983	Letter from applicant concerning liquefaction potential of the soils beneath the ERCW pipeline and Class 1E electrical conduit.
August 18, 1983	Letter from applicant forwarding report, "Reactor Building Containment Integrated Leak Rate Test, Watts Bar Nuclear Plant Unit 1."
August 22, 1983	Letter from applicant concerning the lowest service metal temperature of +40°F for the main feedwater system.
August 31, 1983	Letter to applicant concerning fire protection issues.
August 31, 1983	Letter from applicant concerning diesel generator reliability qualification testing.
September 13, 1983	Letter from applicant forwarding generic control room design review program plan.
September 14, 1983	Letter from applicant concerning geotechnical design features.
September 15, 1983	Letter from applicant concerning containment sump level.
September 20, 1983	Letter from applicant concerning final response to NUREG-0737, Item II.B.3.
September 20, 1983	Letter from applicant concerning postaccident sampling system items.
September 28, 1983	Letter from applicant concerning discontinuance of submittal of meterological data availability reports.
September 29, 1983	Letter from applicant providing results of the preliminary lighting survey for the main control room.
September 30, 1983	Letter to applicant concerning review of Black and Veatch's independent review of the auxiliary feedwater system for Unit 1.
October 4, 1983	Letter to applicant requesting additional information regarding environmental equipment qualification.
October 11, 1983	Letter from applicant concerning safety-grade manual switches in the main control room for closure of isolation valves in the upper head injection system.
October 12, 1983	Letter from applicant concerning compliance with 10 CFR 50, Appendix R.
October 14, 1983	Letter from applicant forwarding letter to Black and Veatch concerning the handling of the independent review program.
October 18, 1983	Letter from applicant forwarding drawings related to two sump level sensors.

October 18, 1983	Letter from applicant concerning meeting between TVA and Black and Veatch to discuss the independent review program.
October 19, 1983	Letter to applicant concerning NRC positions on certain requirements of Appendix R to 10 CFR 50 (Generic Letter 83-33).
October 24, 1983	Letter from applicant concerning NUREG-0737, Item II.D.3, "Direct Indication of Relief and Safety Valve Position."
October 26, 1983	Letter to applicant concerning clarification of required actions based on generic implications of Salem ATWS events.
October 27, 1983	Letter from applicant concerning revisions to Physical Security Plan.
October 28, 1983	Letter to applicant concerning compliance with General Design Criterion 51.
October 28, 1983	Letter from applicant concerning their response to NUREG-0737.
October 31, 1983	Letter to applicant transmitting NUREG-0965, "NRC Inventory of Dams," (Generic Letter 83-38).
October 31, 1983	Letter from applicant concerning NUREG-0737, ITEM II.D.3 (Direct Indication of Relief and Safety Valve Position).
October 31, 1983	Letter from applicant concerning his response to Generic Letter 82-33.
November 1, 1983	Letter from applicant concerning various TVA commitments and SER items.
November 1-2, 1983	Meeting with applicant to discuss the Technical Specifications.
November 2, 1983	Letter to applicant concerning the clarification of TMI Action Plan Item II.K.3.31 (Generic Letter 83-35).
November 7, 1983	Letter from applicant responding to Generic Letter 83-28 concerning the required actions based on generic implications of Salem ATWS events.
November 7, 1983	Letter from applicant concerning compliance with equipment qualification requirements.
November 8, 1983	Letter from applicant concerning their commitments regarding control room modifications.
November 8, 1983	Letter from applicant concerning installation of high-range noble gas monitors on the steam generator safety and PORV release lines.

November 9, 1983	Meeting with applicant at plant site to review remedial work being performed as a result of the liquefaction potential analysis for the soils beneath the ERCW pipeline.
November 21, 1983	Letter from applicant providing revisions to the ASME Section XI, "Preservice Inspection Program Technical Instruction."
November 22, 1983	Letter from applicant requesting exemption for not providing high-range noble gas monitors on the auxiliary building vent.
December 2, 1983	Generic Letter 83-32 -- NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS.
December 2, 1983	Letter from applicant concerning program plan for evaluating the environmental qualification of safety-related mechanical equipment.
December 6, 1983	Letter from applicant concerning containment purge and vent valve operability analysis report.
December 12, 1983	Letter to applicant concerning manual control of the upper head injection system.
December 13, 1983	Meeting with applicant to discuss the Technical Specifications.
December 16, 1983	Letter to applicant concerning review of emergency action levels.
December 19, 1983	Letter to applicant reporting requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications (Generic Letter 83-43).
December 19, 1983	Letter from applicant concerning postaccident sampling capability.
December 19, 1983	Letter to applicant concerning clarification to Generic Letter 81-07 regarding response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Generic Letter 83-42).
December 20, 1983	Letter to applicant concerning availability of NUREG-1021, "Operator Licensing Examiner Standards" (Generic Letter 83-44).
December 20, 1983	Meeting with applicant to discuss requests for exemption from item II.F.1 of NUREG-0737.
December 21, 1983	Letter to applicant regarding operator licensing examinations (Generic Letter 83-40).

December 22, 1983	Letter from applicant concerning Confirmatory Item 31 of the SER.
December 23, 1983	Letter to applicant concerning comments on TVA program plan for control room design reviews.

APPENDIX B

BIBLIOGRAPHY

Letter, Dec. 22, 1972 (NS-SL-518), from R. Salvatori, Westinghouse, to D. Knuth, AEC, Subject: PAD-3.1 - Thermal Performance Code for Westinghouse PWR Fuel.

---, Dec. 29, 1972 (NS-SL-521), from R. Salvatori, Westinghouse, to D. Knuth, AEC, Subject: PAD-3.1 - Thermal Performance Code for Westinghouse PWR Fuel.

---, Jan. 1, 1973 (NS-SL-527), from R. Salvatori, Westinghouse, to D. Knuth, AEC, Subject: PAD-3.1 - Thermal Performance Code for Westinghouse PWR Fuel.

---, Jan. 12, 1973 (NS-SL-543), from R. Salvatori, Westinghouse, to D. Knuth, AEC, Subject: PAD-3.1 - Thermal Performance Code for Westinghouse PWR Fuel.

---, Feb. 9, 1979, from J. Stolz, NRC, to T. Anderson, Westinghouse, Subject: Safety Evaluation of WCAP-8720 and WCAP-8785.

Memorandum, Jan. 14, 1975, from V. Stello, NRC, to R. DeYoung, Subject: Safety Evaluation of WCAP-7982.

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

U.S. General Services Administration, Office of the Federal Register National Archives and Records Service, Code of Federal Regulations, Title 10, "Energy" (including General Design Criteria), U.S. Government Printing Office, Washington, DC, Jan. 1981.

U.S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

---, NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants---LWR Edition" (includes Branch Technical Positions), July 1981; Supplement 1, Sept. 1982.

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

---, NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," University of Dayton Research Institute, Feb. 1979.

---, Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants."

Westinghouse Topical Report WCAP-9, "Topical Report for Overpressure Protection for Westinghouse Pressurized Water Reactors," Rev. 1, Oct. 8, 1971.

---, Topical Report WCAP-7982, "Final Densification Penalty Model," Oct. 1972.

---, Topical Report WCAP-8346, "Evaluation of Fuel Rod Boring," May 1974.

---, Topical Report WCAP-8377, "Revised Clad Flattening Model," July 1974.

---, Topical Report WCAP-8692, "Fuel Rod Bow Evaluation," July 1979.

---, Topical Report WCAP-8720, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations."

---, Topical Report WCAP-8963-A, "Safety Analyses for Revised Fuel Rod Internal Pressure Design Basis," Aug. 1978 (proprietary).

APPENDIX E

PRINCIPAL CONTRIBUTORS

NRC Personnel	Branch
T. Chan	Auxiliary Systems
J. Vogelwede	Core Performance
W. Jensen	Reactor Systems
H. Li	Instrumentation and Control
J. Knox	Power Systems
R. Giardina	Power Systems
J. Spraul	Quality Assurance
M. Duncan	Licensing Branch No. 4

APPENDIX G

ERRATA TO WATTS BAR SAFETY EVALUATION REPORT

Section	Page	Change
4.3.2.1	4-10	In the second paragraph from the bottom, change "peaking factors less than 2.32" to "peaking factors less than 2.31".
5.4.3	5-17	In the last paragraph, the last sentence should read, "If the alternate bypass lines are utilized, the staff requires that the malfunctioning main isolation valve be corrected and the valve in the bypass line be closed (with power removed) before the plant is repressurized".
9.5.1.2	9-29	Under "Elevation 692 ft", change "charging pump room" to "charging pump rooms."
15.2	15-2	Delete Item (9) "high steam generator water level" at top of page.
17.2	17-5	In the third paragraph, delete the words "drawings, specifications" from the sentence, "They review drawings, specifications, purchase requisitions, and plant instructions and procedures covering activities such as test, calibration, special processes, maintenance, modification, and repair for compliance with the QA program requirements."

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(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
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Supplement No. 2

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4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Safety Evaluation Report related to operation of Watts Bar Nuclear Plant, Units 1 and 2

5. DATE REPORT COMPLETED

MONTH	YEAR
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January	1984

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9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
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15. SUPPLEMENTARY NOTES
Pertains to Docket Nos. 50-390 and 50-391

16. ABSTRACT *(200 words or less)*
This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982) and Supplement No. 1 (September 1982), issued 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 and confirmatory items and license conditions identified in the Safety Evaluation Report.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
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19. SECURITY CLASS *(This report)*
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