NUREG-0847
Supplement No. 4

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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

March 1985

UNITED STATES NUCLEAR REGULATORY COMMISSION

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ABSTRACT

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), and Supplement No. 3 (January 1985) 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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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 (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was supplemented by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), and Supplement No. 3 (SSER 3, January 1985), 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 (SSER 4) provides more recent information regarding the resolution or status of some of the open and confirmatory items and license conditions identified in the SER and its supplements. 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 (SSER 4) 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 C is an update to the status of an unresolved safety issue that was discussed in the SER. Appendix E is a list of principal contributors to this supplement. Appendix H is a staff evaluation of the limiting materials of the Watts Bar containment pressure boundary within the context of GDC 51. This supplement made no changes in SER Appendices D, F, and G.

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1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. SSER 4 updates the status of some of those items. The current status of each of the 17 original issues is tabulated below and the relevant SER section is indicated. Resolution of those issues that are, to date, unresolved will be addressed in future supplements.

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*Availability 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	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Preservice and inservice pump and valve test program	Under review	3.9.6
(4) Seismic and environmental qualification of equipment	Seismic - partially resolved (SSER 3)	3.10
	Environmental - under review	3.11
(5) Preservice and inservice inspection program	Under review	5.2.4, 6.6
(6) Pressure-temperature limits for Unit 2	Under review	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) BTP CSB 6-4	Resolved (SSER 3); see License Condition 8	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18	Partially resolved (SSER 2, SSER 3)	8.2
(12) Fire protection program	Under review	9.5.1
(13) Quality classification of diesel generator auxiliary system piping and components	Under review	9.5.4-9.5.8
(14) Diesel generator auxiliary system design deficiencies	Partially resolved (SSER 3)	9.5.5, 9.5.7
(15) Physical Security Plan	Resolved (SSER 1)*	13.6

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

Issue	Status	Section
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) Q list	Resolved (SSER 2)*	17.4

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 the status 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, with the relevant SER section indicated. Resolution of issues that are outstanding, to date, will be addressed in future supplements.

Issue	Status	Section
(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5) Upgrading ERCW system to seismic Category I	Awaiting verification of installation	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety	Awaiting verification of installation	3.2.1
(7) Tornado-missile protection of diesel generator exhaust	Resolved (SSER 2)	3.5.2, 9.5.4.1, 9.5.8
(8) Steel containment building buckling research program	Resolved (SSER 3)	3.8.1
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02)	Under review	3.9.3.4
(10) Thermal performance analysis	Resolved (SSER 2)	4.2.2

*TVA has recently submitted a revised quality assurance program. However, the program approved in SSER 2 is acceptable for use pending approval of the new program.

Issue	Status	Section
(11) Cladding collapse	Resolved (SSER 2)	4.2.2
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13) Loose-parts monitoring system	Resolved (SSER 3); see License Condition 42	4.4.5
(14) Installation of residual heat removal flow alarm	Awaiting verification of installation	5.4.3
(15) Natural circulation tests	Awaiting information	5.4.3
(16) Dump valve testing	Resolved (SSER 2)	5.4.3
(17) Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines	Under review	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system set point methodology	Resolved (SSER 4)	7.1.3.1
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	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	Awaiting verification of acceptability of test results	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4

Issue	Status	Section
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2	Under review	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetrations capability to withstand failure of overcurrent protection device	Under review	8.3.3.6
(36) Missile protection for diesel generator vent line	Awaiting verification of modifications	9.5.4.2
(37) Component booster pump relocation	Awaiting verification of modifications	9.2.2
(38) Electrical penetrations documentation	Under review	9.5.1.3
(39) Compliance with NUREG/CR-0660	See License Condition 22	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator	Awaiting verification of procedure changes	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14
(42) Submergence of electrical equipment as result of a LOCA	Under review	8.3.3.1.1

1.9 License Conditions

In Section 1.9 of the SER and Supplement Nos. 1 and 3 to the SER, the staff identified 42 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 (as appropriate) and the relevant SER section indicated.

Condition	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2
(2) Preservice/inservice testing of pumps and valves	Under review	3.9.6

Condition	Status	Section
(3) Detectors for inadequate core cooling (II.F.2)	Under review	4.4.8
(4) Inservice Inspection Program	Unchanged	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Awaiting verification of installation	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) noble gas monitor	Under review	11.7.1
(b) iodine particulate sampling	Under review	11.7.1
(c) high range incontainment radiation monitor	Awaiting verification of installation	12.7.2
(d) containment pressure	Awaiting verification of installation	6.2.1
(e) containment water level	Awaiting verification of installation	6.2.1
(f) containment hydrogen	Awaiting verification of installation	6.2.5
(7) Modification to chemical feedlines	Under review	6.2.4
(8) Containment isolation dependability (II.E.4.2)	Under review	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7)	Under review	6.2.5, App. C
(10) Status monitoring system	Unchanged (SER)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Awaiting verification of installation	7.8.1
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6
(13) DC monitoring and annunciation	Partially resolved (SSER 3)	8.3.2.2
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3

Condition	Status	Section
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1)	Under review	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (II.B.3)	Partially resolved (SSER 3)	9.3.2
(20) Fire protection program	Under review	9.5.1
(21) Performance testing for communications systems	Under review	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660)	Awaiting verification of modifications	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Unchanged (SER)	10.3.4
(24) Primary coolant outside containment (III.D.1.1)	Under review	11.7.2
(25) Independent safety engineering group (I.B.1.2)	Under review	13.4
(26) Use of experienced personnel during startup	Unchanged (SER)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2)	Under review	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7)	Under review	13.5.2
(29) Modifications to emergency operating instructions (I.C.8)	Awaiting verification of modifications	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3
(31) Initial test program	Partially resolved (SSER 3)	14
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1

Condition	Status	Section
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2)	Under review	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31)	Resolved (SSER 4)	15.5.5
(37) Control room design review (I.D.1)	Awaiting verification of modifications	18
(38) Physical Security Plan	Unchanged (SER)	13.6
(39) Control of heavy loads (NUREG-0612)	Unchanged (SSER 3)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28)	Unchanged (SSER 3)	15.3.6
(41) Steam generator tube rupture	Unchanged (SSER 3)	15.4.3
(42) Loose-parts monitoring system	Unchanged (SSER 3)	4.4.5

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

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Structures, and Core Support Structures

3.9.3.4 Component Supports

In the SER the staff stated that the applicant provided results of analyses for pipe supports to show that short columns (L/r < 100) do not tend to buckle and that the design concepts at the Watts Bar facility preclude the use of columns with L/r >100 (L is the effective length of the column and r is the radius of gyrations). Although the staff concurred with the applicant's conclusions about columns with L/r >100, the staff disagreed with the applicant's criteria for columns with L/r <100. The staff's position relative to the margin to be maintained for critical buckling was outlined in an NRC letter to the applicant dated April 13, 1984. In response to this position, the applicant conducted a sampling program (letter from the applicant to NRC, dated May 14, 1984) and ascertained that the compressive stress for its pipe supports does not exceed the acceptance criteria established by the NRC. On the basis of a review of the sampling program, the staff concludes that Class 2 and 3 pipe supports at Watts Bar comply with the applicable NRC design criteria. Thus, the staff considers Outstanding Issue 2 to be resolved.

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

In 1978 several design and manufacturing deficiencies in deep draft pumps were found at several nuclear power plant facilities both operating and under construction. These deficiencies caused excessive operational vibration and bearing wear, which resulted in reduced flow rates. As a result, IE Bulletin 79-15 was issued to address these deficiencies as they relate to the long-term operability of deep draft pumps in safety-related applications. The staff has reviewed the applicant's responses to this issue in the applicant's submittals through October 30, 1984.

The Watts Bar Nuclear Plant has 16 deep draft pumps located in the intake pumping station. Eight of these pumps supply the essential raw cooling water

(ERCW) system; four pumps are for the screen wash on the ERCW intake station; and four pumps supply the high-pressure fire protection (HPFP) system. The HPFP pumps and the ERCW screen wash pumps do not run continuously during normal plant operation. During normal plant operation, there will be four ERCW pumps running continuously. Service among the eight ERCW pumps is alternated to distribute wear and to satisfy surveillance requirements.

The bearings on the ERCW pumps and ERCW screen wash pumps are cooled by filtered water. Additionally, the bearings incorporate fluted passages on their surfaces to flush out suspended particles small enough to pass the traveling screen filter.

The HPFP pumps do not use water-cooled bearings, consequently, for these pumps, bearing wear caused by particles suspended in water is not a concern.

All 16 pump casings are supported laterally (along their length). Consequently, pump operability will not be affected by lateral displacement during seismic events.

Baseline, warning, and alarm vibration levels have been established for the ERCW and HPFP pumps. Periodic inservice tests of these pumps include vibration monitoring. If measured vibrations exceed the warning level, the frequency of measurements will be increased. Should the vibrations exceed the alarm level, the pumps will be shut down for repair or maintenance. The applicant has documented that the natural frequencies for all deep draft pumps are sufficiently removed from the operational speeds to maintain acceptable vibration levels.

The ERCW screen wash pumps are visually monitored weekly for excessive vibration levels. Weekly visual monitoring of these pumps is acceptable because of their brief intermittent operational requirements. The safety function of these pumps is to prevent excessive head loss across the traveling screens by washing accumulated debris into a trash sluice.

During installation and maintenance of deep draft pumps, proper shaft alignment is assured by adhering to the manufacturer's installation, operation, and maintenance procedures. These procedures include vibration checks to be made on pumps reassembled following maintenance.

On the basis of a review of the applicant's responses to IE Bulletin 79-15, the staff concludes that the applicant has adequately addressed the issue of long-term operability of deep draft pumps. The staff further concludes that the applicant has provided an acceptable means of meeting GDC 1, 2, and 4, as well as Appendix B of 10 CFR 50, as applicable to the operability of safety-related deep draft pumps.

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

In the SER, the staff concluded that because of the generic problem of tube degradation caused by flow-induced vibration in Model D steam generators, operation of the Watts Bar facility would be limited to 50% power when the original steam generators are used. Operating experience on Westinghouse Model D2/D3 pre-heat steam generators from October 1981 to early 1982 indicated that the pre-heat section was subject to severe flow-induced vibrational problems which resulted in accelerated tube wear and leaks. The Watts Bar plant has Westinghouse Model D3 steam generators in both of its units. After extensive testing and analysis, Westinghouse proposed a design modification to resolve this problem. Several utilities, with Westinghouse and NRC concurrence, established a design review panel to examine all aspects of the final Westinghouse design for this pre-heater modification. As a result of this program, Westinghouse design changes to the steam generator inlet were accepted by the utilities and the NRC and incorporated into the Watts Bar steam generators. Additional details of these design changes and the above programs are described in NUREG-0966, "Safety Evaluation Report Related to the Model D2/D3 Steam Generator Design Modification."

By letter dated May 27, 1983, the applicant indicated that it would implement those modifications to the steam generators in the Watts Bar Nuclear Plant, Units 1 and 2, as proposed by the Model D-2/D-3 Steam Generator Design Review Panel in their report of January 1983 (Blackley, January 19, 1983), before fuel loading. The applicant, in a letter dated February 21, 1984, indicated that the steam generators have been modified. The principal elements of the modification are to

(1) remove the existing four-hole backflow resistor from the feedwater nozzle
(2) remove the impingement plate assembly in the steam generator
(3) install a manifold assembly in the steam generator
(4) install a new backflow restrictor (19 holes) in the feedwater nozzle

The staff required the implementation of a vibration monitoring program on the first two plants that installed these modifications. These plants were the William B. McGuire and Virgil C. Summer Nuclear Stations. The results of these vibration monitoring programs are contained in the Westinghouse report, "Modified D2 and D3 Steam Generator Performance Summary Repprt," SG 84-07-008, dated July 5, 1984. The data presented in this report indicate that the modification is functioning satisfactorily. The staff has reviewed this report and concludes that the modification of the Model D3 steam generator at Watts Bar is acceptable and the plant can be operated at 100% power. The applicant need not perform an early steam generator inspection (i.e., after 6 months of power operation) at the Watts Bar facility as was required at the first two modified plants, Summer and McGuire Unit 1 (as discussed in NUREG-0966).

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.5 Combustible Gas Control Systems

In the SER, the staff stated that additional information was required concerning the analysis of the production and accumulation of hydrogen within the containment during a design-basis LOCA. The necessary information has been provided in Amendments 49 and 53 to the Watts Bar FSAR. The following is the staff's evaluation of these amendments.

After a design-basis loss-of-coolant accident, hydrogen may accumulate within the containment as a result of: (1) metal-water reaction between the fuel cladding and the reactor coolant; (2) radiolytic decomposition of the post-accident emergency cooling water; and (3) corrosion of metals (zinc and aluminum) by emergency core coolant and containment spray solutions. The applicant has analyzed the production and accumulation of hydrogen within containment from the above sources using the guidelines of Regulatory Guide (RG) 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Rev. 2, and 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors." The applicant has used the same assumptions as recommended in RG 1.7 to calculate the rate of hydrogen released by radiolysis and corrosion of metals, and a 1.5% zirconium-water reaction in the reactor core. The 1.5% zirconium-water reaction was determined by assuming a maximum total reaction of 0.3% of the Zircaloy cladding in the reactor core, and multiplying by a factor of 5 as required by 10 CFR 50.44. Emergency core cooling system analyses have indicated a maximum total reaction of less than 0.3%.

With the foregoing assumptions, and considering that the return air fan recirculation system is operated at 10 minutes following onset of the accident, mixing the upper and lower compartment volumes, RG 1.7 hydrogen flammability limits (4%) would not be reached in the containment volume until about 7 days following the accident. The applicant proposes to operate the hydrogen recombiners (previously described in the SER) well in advance of this time; i.e., after 24 hours, which, the analysis shows, would maintain hydrogen concentration below this limit. The staff finds the applicant's analysis to be acceptable.

The staff concludes, therefore, that the design of the combustible gas control system is acceptable and meets the requirements of 10 CFR 50.44 and 50.46, and GDC 5, 41, 42, and 43. Therefore, the staff considers Outstanding Issue 9 to be resolved.

6.2.6 Containment Leakage Testing

Containment Air Lock Surveillance

By letter dated December 3, 1984, the applicant requested an exemption from certain requirements of Appendix J to 10 CFR 50. The staff's evaluation of this request for exemption follows.

Paragraph III.D.2(b)(ii) of Appendix J states: "Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a."

Whenever the plant is in cold shutdown (Mode 5) or refueling (Mode 6), containment integrity is not required. However, if an air lock is opened during Modes 5 and 6, paragraph III.D.2(b)(ii) of Appendix J requires than an overall air lock leakage test at not less than P_a be conducted before plant heatup and startup (i.e., entering Mode 4). The existing air lock doors are so designed that a full pressure, i.e., P_a (15 psig) test of an entire air lock can only be performed after strong backs (structural bracing) have been installed on the inner door. Strong backs are needed since the pressure exerted on the inner door during the test is in a direction opposite to the direction of the accident pressure. Installing strong backs, performing the test, and removing the strong backs require at least 8 hours per air lock (there are 2 air locks), during which access through the air lock is prohibited.

If the periodic 6-month test of paragraph III.D.2(b)(i) of Appendix J and the test required by paragraph III.D.2(b)(iii) of Appendix J are current, no maintenance has been performed on the air lock that could affect its sealing capability, and the air lock is properly sealed, there should be no reason to expect the air lock to leak excessively just because it has been opened in Mode 5 or Mode 6.

Accordingly, the staff concludes that the applicant's proposed approach of substituting the seal leakage test of paragraph III.D.2(b)(iii) for the full-pressure test of paragraph III.D.2(b)(ii) of Appendix J is acceptable when no maintenance that could affect sealing capability has been performed on an air lock. Whenever maintenance that could affect sealing capability has been performed on an air lock, the requirements of paragraph III.D.2(b)(ii) of Appendix J must still be met by the applicant.

Therefore, an exemption from this requirement [10 CFR 50, Appendix J, paragraph III.D.2(b)(ii)] is justified and acceptable for Watts Bar, Units 1 and 2; and the applicant's proposal to adopt surveillance requirement 4.5.1.3.b.2 of Revision 4 of NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," is acceptable.

Thus, on the basis of the foregoing and in accordance with 10 CFR 50.12(a), the staff has concluded that the partial exemption from paragraph III.D.2(b)(ii) of Appendix J to 10 CFR 50, as discussed above, is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff has reviewed confirmatory information submitted by the applicant and concludes that reasonable assurance has been provided that the Watts Bar Units 1 and 2 reactor containment pressure boundary materials will behave in a non-brittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are satisfied. Therefore, the staff considers Confirmatory Issue 19 closed.

The technical basis for the staff's conclusion is presented in Appendix H to this SER.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.3 Design Criteria

7.1.3.1 Safety System Setpoint Methodology

In the SER, the staff indicated that an audit review of the setpoint methodology would be performed during review of the plant's Technical Specifications. The applicant's letters, dated April 25, 1983; September 4, 1984; and October 16, 1984, provided information related to this issue.

The staff has reviewed the information provided and finds that the methodology used to determine the setpoints for Watts Bar Units 1 and 2 is consistent with the methodology approved previously by the staff (see NUREG-0717, Supplement 4) and is, therefore, acceptable.

During the staff's review, several inconsistencies were encountered between values used in Chapter 15 of the Watts Bar FSAR, values used in the plant's current draft Technical Specifications, and values shown in the setpoint methodology. The applicant has committed to revise the appropriate documentation and, during review of the final draft of the Technical Specifications, the staff will confirm that the inconsistencies are eliminated. Therefore, the staff considers Confirmatory Issue 21 closed.

7.6 All Other Systems Required for Safety

7.6.5 Overpressure Protection During Low-Temperature Operation

By Amendment 52 of the FSAR, the applicant provided updated information on the reactor coolant system (RCS) pressure control system used during low-temperature operation. Two power-operated relief valves (PORVs) are used to provide overpressure protection of the RCS during low-temperature operation. The PORVs are automatically opened when RCS pressure exceeds a programmed setpoint based on RCS temperature. During normal operation, this system is manually blocked to preclude a single failure resulting in inadvertent operation of a PORV. The auctioneered lowest wide-range RCS temperature measurements are used to provide the programmed overpressure setpoint. The wide-range RCS pressure measurements are used to activate the system. For the PORV "A" control circuitry, the temperature inputs come from protection set I, and the pressure input comes from protection set III. For PORV "B" control circuitry, the temperature and pressure inputs come from protection set II. All the signals from protection sets are properly isolated. Each of the two PORVs is supplied with an independent Class 1E power supply. Three alarms are provided. The first alarm indicates that the auctioneered RCS temperature is approaching the low-temperature mode of operation, the second alarm indicates that the measured pressure is approaching the overpressure setpoint, and the third alarm indicates that an actuation signal has been provided to the PORV. The Technical Specifications Sections 3.4.9.3 and 4.4.9.3.1 satisfactorily defined the PORV programmed trip setpoint and the surveillance requirements for the low-temperature overpressure protection system. The staff finds this design acceptable, and therefore, considers Confirmatory Issue 25 closed.

7.8 NUREG-0737 Items

7.8.4 Proposed Anticipatory Trip Modification (II.K.3.10)

TMI Action Plan Item II.K.3.10 of NUREG-0737 contains the following position:

> The anticipatory trip modification proposed by some licensees to confine the range of use to high power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by this modification.

By letters dated April 6, 1984, and October 23, 1984, the applicant requested that the requirement for the anticipatory reactor trip on turbine trip be removed for reactor power levels at or below 50% for the Watts Bar facility. The request was supported by analyses demonstrating that the lifting of a pressurizer PORV would be unlikely following a turbine trip at 52% power. The rod control system was determined to be capable of reducing power to within the 40% capacity of the steam dump system before the power imbalance could cause the PORV to lift.

This was concluded even for the most adverse moderator reactivity feedback conditions that would exist at the beginning of plant life. If additional failures such as failure of the steam dump or loss of offsite power occurred, the PORV would be expected to open. The applicant demonstrated however, that the reactor system pressure would remain within acceptable levels and departure from nucleate boiling would not occur even under adverse failure conditions. Supplement 2 of the staff's SER concluded that the Watts Bar plants are adequately protected from overpressure following turbine trip from 100% of full power without credit for the anticipatory trip. This is more limiting than the 50% case.

Following a turbine trip from 50% power, reactor power would normally be reduced by the rod control system. If the rod control system were in manual mode the reactor would remain at power until the operator took action or until the reactor tripped from the loss of secondary coolant in about 10 minutes. The applicant evaluated the offsite dose consequences for the case of full steam release to the atmosphere through the secondary safety valves. The offsite dose consequences were estimated to be very small fractions of 10 CFR 100 dose guidelines.

The staff concluded that the applicant has adequately addressed the requirements of NUREG-0737 Item II.K.3.10 for removal of the anticipatory reactor trip on turbine trip at or below 50% power and that the design of Watts Bar may be modified accordingly. Therefore, the staff finds the applicant's proposal acceptable.

11 RADIOACTIVE WASTE MANAGEMENT

11.2 Liquid Waste Management

In FSAR Revisions 49 and 52, the applicant revised its description of the liquid radwaste treatment system. The system described in Revisions 49 and 52 differs from the system previously described and evaluated by the NRC staff in that the auxiliary waste evaporator has been deleted and the condensate demineralizer waste evaporator is used to process both tritiated water, non-tritiated water, and condensate demineralizer waste. A mobile demineralizer is also used to process both tritiated and non-tritiated water. Design parameters of the principal components of the liquid waste management system are given in the revised Table 11.1.

The staff has evaluated this revised description and has determined that the conclusions reached in the SER are not affected by the revisions.

Table 11.1 Design parameters of principal components considered in the evaluation of liquid and gaseous radioactive waste treatment systems

Component	Number	Capacity each
LIQUID SYSTEM*		
<u>Tritiated water processing</u>		
Tritiated drain collector tank	1	24,700 gal
Mobile demineralizer	1**	30 gpm
Condensate demineralizer waste evaporator	1***	30 gpm
<u>Non-tritiated water processing</u>		
Floor drain collector tank	1	23,500 gal
Mobile demineralizer	1**	30 gpm
Condensate demineralizer waste evaporator	1***	30 gpm
<u>Laundry, hot shower, chemical waste, and decontamination waste processing</u>		
Laundry and hot shower drain tanks	2	600 gal
Chemical drain tank	1	600 gal
Cask decontamination tank	1	15,000 gal
<u>Condensate demineralizer waste processing</u>		
Condensate demineralizer waste processing equipment high crud tanks	2	19,000 gal
Condensate demineralizer waste evaporator	1**	30 gpm
GASEOUS SYSTEM*		
<u>Gaseous waste processing system</u>		
Compressors	2	40 scfm
Decay tanks	9	600 ft^3

*Quality group and seismic design in accordance with RG 1.143.
**One mobile demineralizer is used to process both tritiated and non-triated water.
***One condensate demineralizer waste evaporator is used to process tritiated water, non-tritiated water, and condensate demineralizer waste.

15 ACCIDENT ANALYSIS

15.2 Normal Operation and Anticipated Transients

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.4 Inadvertent Boron Dilution

At the time the Watts Bar SER was written, the staff did not have sufficient information to conclude that the boron-dilution alarm met the single-failure criteria of Appendix A to 10 CFR 50. The alarm is designed to warn the operator of impending criticality so that appropriate action may be taken. By letter dated November 2, 1984, the applicant stated that the boron-dilution alarm system receives signals from two independent source range neutron channels. Each channel is powered by an independent Class 1E power source. Excessive increase in count rate from either channel will actuate the control room annunciator system. The entire circuit will be tested every 18 months during the source range neutron flux-channel calibration. The bistables will be tested monthly in the source range neutron flux channel functional test. The annunciator circuits will be tested each shift. The staff concludes that the system is adequately protected from single failure and that Outstanding Issue 16 is closed.

15.4 Radiological Consequences of Accidents

15.4.5 Fuel-Handling Accident

In Amendment 54 to the FSAR, the applicant proposed operating the reactor building purge ventilation system charcoal filters at a reduced efficiency from that used in the staff's previous evaluation. As a result, the staff has reevaluated the consequences of a fuel-handling accident inside primary containment. The applicant states that at all times during refueling operations the containment will either be isolated or ventilated to the atmosphere through the reactor building purge ventilation system (RBPVS).

The assumptions regarding the reactor shutdown time, the number of fuel rods damaged, the iodine decontamination efficiency of the water in the refueling cavity, and the atmospheric dispersion factors for the fuel-handling accident inside containment are the same as those assumed for the fuel-handling accident in the fuel pool area in the auxiliary building as reported in the SER. The resultant radiological consequences following a postulated fuel-handling accident inside containment also are in revised Table 15.1 and the appropriate assumptions are in revised Table 15.6.

The staff finds that the applicant has provided an adequate system to mitigate the radiological consequences of a postulated fuel-handling accident inside the containment and in the spent fuel pool area. The staff concludes that the fuel-handling area ventilation system meets the relevant requirements of GDC 61. The staff further concludes that the distance to the exclusion area and to the low population zone boundaries for Watts Bar, in conjunction with the operation of dose-mitigating engineered safety features (ESFs) and implementation of plant procedures, are sufficient to provide reasonable assurance that the calculated offsite radiological consequences of a postulated fuel-handling accident are well within the 10 CFR 100 exposure guidelines.

The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at Watts Bar meet the requirements of GDC 61 with respect to radioactivity control; (2) the staff review of the applicant's assumptions and analyses of the radiological consequences from the fuel-handling accident, (3) the staff's independent analyses using the assumptions in RG 1.25, Sections C.1.a through C.1.k, and (4) the Watts Bar Technical Specifications relating to fuel-handling and ventilation system operations.

15.5 NUREG-0737 Items

15.5.1 Thermal-Mechanical Report (II.K.2.13)

NUREG-0737 requires the performance of a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

The SER stated that the applicant was committed to the Westinghouse Owner's Group (WOG) generic resolution of this issue. The staff has completed its review of the WOG submitted for this item, and has concluded that there is reasonable assurance that vessel integrity will be maintained for this type of event. Review of this item will continue under Unresolved Safety Issue (USI) A-49, "Pressurized Thermal Shock."

The staff has determined that the USI A-49 review need not be completed to support the full-power license. Therefore, the staff considers License Condition 32 resolved.

15.5.2 Voiding in the Reactor Coolant System During Transients (II.K.2.17)

NUREG-0737 requires the applicant to analyze the potential for voiding in the reactor coolant system during anticipated transients.

The SER stated that Westinghouse was performing a generic study to address the issue of voiding in the reactor coolant system. Westinghouse has submitted that study which addresses the potential for void formation in the Westinghouse-designed nuclear steam supply system (NSSS) during natural circulation cooldown/depressurization transients. The staff has reviewed and approved the study and has determined that no further action needs to be taken by the applicant. Therefore, the staff considers License Condition 33 resolved.

15.5.4 Automatic Trip of Reactor Coolant Pumps (II.K.3.5)

NUREG-0737 requires that the reactor coolant pumps be tripped automatically in case of a small-break LOCA. The applicant was asked to consider other solutions to the small-break LOCA problem.

The SER stated that Westinghouse was performing a generic study to address this issue. Generic Letter 83-10c was sent to the applicant which (1) reaffirmed the conformance of small-break LOCA evaluation models with Appendix K to 10 CFR 50 for the case of limited reactor coolant pump operation after a reactor trip, and (2) approved the use of these models for the determination of the preferred reactor coolant pump trip strategy (automatic trip, manual trip, or no trip). By letter dated April 22, 1983, the applicant responded. The staff is currently reviewing this letter and the WOG submittals regarding this matter and anticipates resolution in 1985.

The staff has determined that this review need not be completed to support the full-power license. If the results of the generic review reveal that modifications to the preferred reactor coolant pump trip strategy are required to ensure the safe shutdown of the facility during a transient, the staff will require that the applicant make the appropriate modifications. Therefore, the staff considers License Condition 35 resolved.

15.5.5 Small-Break LOCA Methods (II.K.3.30) and Plant-Specific Calculations (II.K.3.31)

NUREG-0737 requires the NSSS vendor to revise its analysis methods for small-break LOCA analysis to show compliance with Appendix K of 10 CFR 50. The applicant was then required to submit plant-specific calculations using the above NRC-approved models.

The SER stated that Westinghouse was modifying its small-break LOCA model to address this issue. The WOG submitted WCAP-10079 to address the revisions made to its small-break LOCA model. Staff review of the WOG submittal is expected to be completed in 1985.

The staff has determined that this review need not be completed to support the full-power license. If the results of the generic review reveal that plant-specific analyses are required to resolve this issue, the staff will require the applicant to make the appropriate revisions. Therefore, the staff considers License Condition 36 resolved.

Table 15.1 Radiological consequences of design-basis accidents

Postulated accident	Exclusion area boundary, rems		Low population zone, rems	
	Thyroid	Whole body	Thyroid	Whole body
LOSS OF COOLANT				
Containment leakage				
0-2 hr	3.1	1.8	0.43	0.25
2-8 hr			0.56	0.22
8-24 hr			0.41	0.18
24-96 hr			1.92	0.15
96-720 hr			1.44	0.06
Total containment leakage	3.1	1.8	4.8	0.86
ECCS component leakage	5.4	0.01	9.1	0.01
Total LOCA	8.5	1.8	13.9	0.9
STEAMLINE BREAK OUTSIDE SECONDARY CONTAINMENT				
Long-term operation case (Case 2)	7.1	<0.1	5.6	<0.1
Short-term operation case (Case 3)	8.9	<0.1	3.8	<0.1
CONTROL ROD EJECTION				
Containment leakage pathway	35.0	<0.9	42.0	0.2
Secondary system release pathway	12.0	<0.1	3.0	<0.1
FUEL-HANDLING ACCIDENT				
In fuel-handling area	1.0	0.3	0.1	<0.1
Inside primary containment	26	0.4	3.6	<0.1
SMALL LINE BREAK OUTSIDE CONTAINMENT	17.0	<0.1	2.3	<0.1
STEAM GENERATOR TUBE RUPTURE				
Case 1 (DEI*-131 at 60 μCi/gm)	73	<0.1	12.0	<0.1
Case 2 (DEI*-131 at 1μCi/gm)	13	<0.1	3.0	<0.1

*DEI - dose equivalent iodine.

Table 15.6 Assumptions used for estimating the radiological consequences following a postulated fuel-handling accident

Parameter	Assumption
Power level, MWt	3592
Fuel rods damaged, no.	264
Fuel rods in core, no.	50,952
Radial peaking factor of damaged rods	1.65
Shutdown time, hr	100
Inventory released from damaged rods (iodines and noble gases), %	10
Pool decontamination factors	
Iodines	100
Noble gases	1
Iodine fractions released from pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies for auxiliary building gas treatment system (spent fuel pool area), %	
Elemental	99
Organic	99
Particulate	99
Iodine removal efficiencies for reactor building purge system, %	
Elemental	90
Organic	30
Particulate	90
0-2 hr χ/Q value at 1200 m, sec/m^3	3.6×10^{-4}
0-8 hr χ/Q value at 4828 m, sec/m^3	5.0×10^{-5}

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(3) Surveillance Requirements for ERCW Cement Mortar Lining

In its report on the Watts Bar Nuclear Plant, Units 1 and 2, dated August 16, 1982, the ACRS expressed concern about the long-term survivability of the cement mortar lining of the essential raw cooling water (ERCW) piping and the effect a sudden failure of the lining would have on essential systems. A 3/8-inch-thick layer of cement mortar was applied to the large-diameter carbon steel piping to prevent the buildup of a layer of corrosion products which could in time restrict cooling water flow. The ACRS recommended periodic inspections of the cement mortar lining. The ACRS also stated a concern that the lining might fail by "heaving" of the type that causes concrete to crack away from steel reinforcing bars in bridge decks. By letters dated September 21, 1982, and June 15, 1984, the applicant provided additional information on the cement mortar lining and on a program of inservice inspection of the lining.

Evaluation

Steel piping lined with cement mortar has been in use for more than 60 years in city water distribution systems, and more recently to carry cooling water for power-generating plants. The steel was protected from internal corrosion by the alkalinity of the coating. The linings remained intact for many years. Lining failures were observed in cases of plastic deformation of the steel pipe (Uhlig, 1971).

The applicant performed tests to demonstrate the adherence and flexibility of freshly installed cement mortar lining. These tests are discussed in Section 3.7.3.12 of the FSAR. The results of these tests indicate that significant short-term failure of the lining is unlikely.

As cement mortar ages, it normally gains in strength, and carbonation of the free lime content by dissolved carbon dioxide in the river water does not materially affect its strength (Lea, 1977; Taylor, 1977). However, leaching of more than approximately 50% of the free lime and calcium carbonate content of the mortar would significantly weaken it (Lea, 1971).

The rates of leaching and carbonation depend upon the hardness and acidity of the leaching water. On the basis of experience with cement mortar linings exposed to similar water, the staff concludes that it is unlikely that the linings will be significantly weakened by leaching during plant life.

With respect to the concerns about "heaving," this mode of cracking is related to the use of chlorides on bridge decks for snow removal (Tonini and Dean, 1976). The concentrated chloride solution percolates through the concrete to the steel reinforcing bars and catalyzes rapid corrosion of the metal, resulting in the growth of an iron oxide layer which finally cracks the surrounding concrete.

The average concentration of chloride in Tennessee River water is approximately 10 ppm, far below the concentration required to overcome the corrosion-inhibiting effect of the alkaline cement mortar lining. Therefore, the staff does not expect a "heaving" effect in cement mortar lining exposed to the Watts Bar cooling water.

In the event of cement mortar lining failure, the staff considered the fate of lining debris particles. According to Spell's correlation for the water velocity required to maintain slurry particles in suspension (Perry et al., 1969), only small particles of cement mortar debris, much smaller than 1 mm in diameter, would be entrained by the highest expected flow velocity in the ERCW piping. Larger particles of debris would settle out along the bottom of the piping. The presence of fine suspended solids from lining failure in the ERCW is not expected to have any adverse safety consequences. Solid particles approximately 1 mm in diameter can pass through the apertures in the traveling screens in the intake structure during normal operation. Therefore, the staff concludes that failure of the cement mortar lining of the ERCW piping would not pose a significant safety problem.

By letter dated June 15, 1984, the applicant proposed a program to periodically inspect the condition and calcium content of a section of cement-mortar-lined pipe exposed to flowing Tennessee River water. The applicant stated that it would take further action if the loss of calcium content exceeded 40%. Greater calcium loss would result in a weakened structure. The annual inspection would ensure that the cement mortar lining maintained its integrity and continued to provide corrosion protection to the steel pipe. The staff finds this program acceptable, and concludes that the cement mortar lining in the ERCW piping is acceptable.

APPENDIX A

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

August 7, 1984	Letter from applicant forwarding report, "Reactor Building Containment Integrated Leak Rate Test Supplement, Watts Bar Nuclear Plant Unit 1."
August 14,1984	Letter to applicant forwarding results of the fire protection audit.
August 20, 1984	Generic Letter 84-20, "Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions," issued.
August 22, 1984	Letter to applicant concerning review of deep draft pumps.
August 22, 1984	Letter to applicant concerning review of draft Technical Specifications.
August 23, 1984	Letter to applicant concerning containment purge and vent valve operability.
August 23, 1984	Letter to applicant concerning compliance with GDC 51.
August 23, 1984	Letter from applicant concerning deletion of reactor building purge ventilation system from the Unit 1 Technical Specifications.
August 23, 1984	Letter from applicant concerning pressure isolation valve in-service test program.
August 29, 1984	Letter to applicant concerning review of Appendix R submittal.
August 29, 1984	Meeting with applicant to discuss Appendix R requirements. (Summary issued October 3, 1984.)
August 30, 1984	Letter from applicant concerning Technical Specification limits for reactor coolant system flow rate and flow measurement uncertainty.
September 4, 1984	Letter from applicant concerning safety system set point methodology.
September 4, 1984	Letter from applicant concerning shift operating crews, including hot participation experience.

September 6, 1984	Letter from applicant providing current status of control room modifications.
September 13, 1984	Meeting with applicant to discuss modifications to fire protection system. (Summary issued September 26, 1984.)
September 14, 1984	Letter to applicant requesting additional information concerning the safety parameter display system.
September 14, 1984	Letter from applicant providing comments/proposed modifications to the proof and review version of the Technical Specifications.
September 17, 1984	Letter from applicant providing supplemental response to Generic Letter 83-28.
September 24, 1984	Letter from applicant concerning NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
October 3, 1984	Letter from applicant concerning control room modifications.
October 4, 1984	Letter from applicant concerning integrity of systems outside containment.
October 5, 1984	Letter from applicant advising that fuel load dates have been changed to March 1985 and March 1987 for Units 1 and 2, respectively.
October 9, 1984	Letter from applicant concerning techniques used in the seismic analysis of rigorously analyzed piping.
October 10, 1984	Letter to applicant concerning report on implementation of inadequate core cooling instrumentation.
October 15, 1984	Letter from applicant concerning fire protection.
October 16, 1984	Generic Letter 84-21, "Long-Term Low Power Operation in Pressurized Water Reactors," issued.
October 16 1984	Letter from applicant providing revised data for auxiliary feedwater pumps set points.
October 16, 1984	Letter from applicant concerning fire protection.
October 17, 1984	Letter from applicant forwarding Revision 9 to Physical Security/Contingency Plan.
October 18, 1984	Letter from applicant concerning use of fire-retardant coatings on all grouped electrical cables in designated safety-related areas.
October 19, 1984	Letter from applicant concerning requirement for temperature monitor in diesel generator room.

October 23, 1984	Letter from applicant concerning proposed anticipatory trip modification.
October 24, 1984	Letter to applicant concerning PORV and block valve emergency power.
October 25, 1984	Letter from applicant concerning containment purge and vent valve operability assurance.
October 25, 1984	Letter from applicant concerning manual reset of the safety injection signal during the injection phase following a loss of offsite power.
October 30, 1984	Letter from applicant concerning use of deep draft pumps.
November 2, 1984	Letter from applicant concerning potential boron-dilution events.
November 6, 1984	Letter to applicant concerning fire protection criteria.
November 7, 1984	Letter from applicant concerning relief from preservice inspection program.
November 9, 1984	Letter to applicant concerning shift crew experience.
November 9, 1984	Letter from applicant providing comments on proof and review version of Technical Specifications.
November 14, 1984	Letter to applicant concerning use of ASME Code N-401, Recording Data.
November 14, 1984	Meeting with applicant to discuss fire protection. (Summary issued November 27, 1984.)
November 19, 1984	Letter to applicant requesting meeting to discuss control room design review for TVA nuclear facilities.
November 20, 1984	Letter from applicant concerning startup test program.
November 27, 1984	Letter from applicant concerning the permanent hydrogen mitigation systems.
December 3, 1984	Letter from applicant requesting exemption from requirements of paragraph III.D.2(b)(ii) of Appendix J to 10 CFR 50.
December 4, 1984	Meeting with applicant to discuss control room design reviews for TVA nuclear facilities. (Summary issued December 28, 1984.)
December 7, 1984	Letter from applicant concerning engineered safety features actuation system slave relays.

December 10, 1984	Letter from applicant concerning compliance with GDC 51.
December 12, 1984	Letter from applicant concerning engineered safety features actuation system slave relays.
December 19, 1984	Letter to applicant requesting additional information concerning the initial test program.
December 27, 1984	Generic Letter 84-24, "Certification of Compliance to 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," issued.
January 3, 1985	Letter from applicant concerning possible discrepancies between draft Technical Specifications, FSAR, and SER.
January 9, 1985	Letter to applicant requesting additional information concerning the fire protection program.
January 9, 1985	Letter from applicant concerning photographs of the main control room.
January 9, 1985	Generic Letter 85-01, "Fire Protection Policy Steering Committee Report," issued.
January 14, 1985	Letter to applicant concerning review of responses to power systems concerns.
January 14, 1985	Letter from applicant concerning initial test program.
January 15, 1985	Letter to applicant requesting additional information regarding main steamline break accident analysis.
January 16, 1985	Letter from applicant concerning the as-built configuration of the underground barrier at Watts Bar Nuclear Plant.
January 16, 1985	Letter from applicant verifying that auxiliary feedwater pumps could survive the transition to the backup water source in the event the preferred source is unavailable.
January 16, 1985	Letter from applicant concerning compliance with NUREG-0612.
January 16, 1985	Letter from applicant concerning power systems concerns.
January 17, 1985	Letter to applicant concerning program entitled, Effectiveness of LWR Regulatory Requirements in Limiting Risk.
January 22, 1985	Meeting with applicant to discuss Technical Specification issues. (Summary issued January 30, 1985.)

January 28, 1985	Letter to applicant concerning Federal Emergency Management Agency exercise report.
January 28, 1985	Letter to applicant concerning piping design criteria.
January 28, 1985	Letter to applicant requesting additional information regarding associated circuits.
January 29, 1985	Letter to applicant transmitting Supplement No 3 to SER.

APPENDIX B

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---, NUREG-0694, "TMI-Related Requirements for New Operating Licenses," June 1980.

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APPENDIX C

NUCLEAR REGULATORY COMMISSION
UNRESOLVED SAFETY ISSUES

This appendix provides an update to the NRC staff's evaluation of one unresolved safety issue (USI) that is applicable to the Watts Bar facility.

A-3 Westinghouse Steam Generator Tube Integrity

The staff noted in the Watts Bar SER that it was reviewing additional information on this item and would report its findings in a supplement to the SER. The staff has supplemented its review of the Model D3 steam generator design as presented in Section 5.4.2.2 of this supplement. On the basis of this review, the staff finds the steam generators used in the Watts Bar facility acceptable. Therefore, the staff concludes that the Watts Bar facility can be operated before final resolution of this generic issue without endangering the health and safety of the public.

APPENDIX E

PRINCIPAL CONTRIBUTORS

NRC Personnel	Branch
F. Akstulewitz	Accident Evaluation
M. Duncan	Licensing
J. Halapatz	Materials Engineering
W. Jensen	Reactor Systems
S. Kirslis	Chemical Engineering
H. Li	Instrumentation and Control
C. Nichols	Meteorology and Effluent Treatment
J. Pulsipher	Containment Systems
J. Rajan	Mechanical Engineering
N. Romney	Seismic Qualification

APPENDIX H

TECHNICAL BASIS REGARDING
FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

GDC 51, "Fracture Prevention of Containment Pressure Boundary," requires that the reactor containment boundary shall be designed with sufficient margin to ensure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Below is the staff's technical basis regarding the limiting materials of the Watts Bar reactor containment pressure boundary within the context of GDC 51. The staff has determined that the following information provided by TVA in its submittals dated December 23, 1981; December 17, 1982; August 22, 1983; and January 9, March 30, June 7, and December 10, 1984, demonstrates that the requirements of GDC 51 have been satisfied.

1. CONTAINMENT VESSEL EQUIPMENT HATCH LOCKS PENETRATION SLEEVES

(a) Equipment Hatch Tension Ring

ASME Code steel specification SA 516, Grade 70, quenched and tempered, 3.00 in. thick applied. ASME Code Summer 1977 Addenda, Class 2 rules assign a T_{NDT} of -10°F and a permissible lowest service metal temperature (PLSMT) of 30°F. The postulated lowest service metal temperature (LSMT) is identified as 30°F (FSAR 3.8.2.2.2).

(b) Welded Spare Penetration Head (Chicago Bridge and Iron Co. Dwg 72-4333, Piece Mark 310-4)

ASME Code steel specification SA 516, Grade 70, normalized and tempered, 2.5 in. thick applied. ASME Code Summer 1977 Addenda, Class 2 rules assign a T_{NDT} of 0°F and a PLSMT of 30°F.

(c) Rolled Pipe, Penetration Nozzle (Chicago Bridge and Iron Co. Dwg 72-4333, Piece Mark 313-1)

ASME Code steel specification SA 516, Grade 70, normalized and tempered, 1.25 in. thick applied. ASME Code Summer 1977 Addenda, Class 2 rules assign a T_{NDT} of 0°F and a PLSMT of 30°F.

(d) Seamless Pipe, Penetration Nozzle (Chicago Bridge and Iron Co. Dwg 72-4333, Piece Mark 310-1)

ASME Code steel specification SA 333, Grade 6, quenched and tempered, 24-in. Schedule 80 pipe (1.2-in. wall) applied. NUREG-0577 would categorize this material as C-Mn, to which Table 4.4 would assign a

T_{NDT} at or below the -28°F average NDT for normalized material. Assuming a T_{NDT} of -28°F, ASME Code Summer 1977 Addenda, Class 2 rules would assign a PLSMT of 2°F.

(e) Main Steam Penetration Sleeve (TVA Dwg 47W700-4R6: Item 25a)

ASME Code steel specification SA 516, Grade 70, 1.75 in. thick, is applied via SA 155, KCF 70, Class 1. SA 516, Grade 70, for 1.75-in. material, demands normalization. Fabricator stress relieved fabricated sleeve. ASME Code Summer 1977 Addenda, Class 2 rules would assign a 0°F T_{NDT} and a PLSMT of 30°F.

2. MAIN STEAM SYSTEM CONTAINMENT PRESSURE BOUNDARY MATERIALS

(a) Lowest Service Metal Temperature

TVA Design Division Piping Bill of Material for Watts Bar Dwg 47W400 series identifies 70°F as the LSMT.

(b) Main Steam Penetration Process Pipe

ASME Code steel specification SA 516, Grade 70, 1.175 in. thick, is applied via SA 155, KCF 70, Class 1. Mill practice for this pipe fabricates normalized pipe. ASME Code Summer 1977 Addenda, Class 2 rules would assign a 0°F T_{NDT} and a PLSMT of 30°F.

(c) Safety Valve Manifold (Piece Mark 01A-MS-4: Serial No. 6836)

ASME Code steel specification SA 516, Grade 70, 2.75 in. thick, is applied via SA 155, KCF 70, Class 1. ASME Code Summer 1977 Addenda, Class 2 rules would assign a 0°F T_{NDT} and a PLSMT of 35°F.

(1) Pipe (on Piece Mark 01A-MS-4)

ASME Code steel specification SA 106, Grade B, 6 in. XXS (extra strong) (0.864-in. wall), by U.S. Steel (Lorain Works), whose pipe mill practice would deliver pipe to the cooling bed above the Ar_3 temperature. NUREG-0577, Fig. B.7 data for normalized material would assign a T_{NDT} at or below the Table 4.4 average NDT of 40°F. Assuming a T_{NDT} of 40°F, ASME Code Summer 1977 Addenda, Class 2 rules would assign a PLSMT of 70°F.

(2) Long weld neck flanges (on Piece Mark 01A-MS-4)

ASME Code steel specification SA 105, normalized, 6 in. (ID) x 9 in. (OD), is applied. NUREG-0577, Table 4.4 would assign an (average NDT + 1.3σ) NDT of -5°F. ASME Code Summer 1977 Addenda, Class 2 rules would assign a PLSMT of 25°F.

(3) Relief valve manifold weld cap (on Piece Mark 01A-MS-4)

ASME Code steel specification SA 234 WPB from SA 516, Grade 70, 1.109-in. minimum wall, is applied. Mill practice via SA 234 WPB fabricates a normalized and tempered cap. ASME Code Summer 1977 Addenda, Class 2 rules assign a T_{NDT} of 0°F and a PLSMT of 30°F.

(d) Main Steam System Fittings (Typical: Piece Mark 01A-MS-5)

(1) Ell (32-in. x 1.175-in. main wall S/R 90°)

ASME Code steel specification SA 234 WPB from SA 516, Grade 70 is applied. Analysis for item 2(c)(3) (relief valve manifold weld cap) applies.

(2) Tee (32-in. x 1.175-in. main wall)

ASME Code steel specification SA 234 WPB from SA 516, Grade 70 is applied. Analysis for item 2(c)(3) (relief valve manifold weld cap) applies.

(3) Weldolet (32-in. x 1.175-in. main wall)

ASME Code steel specification SA 234 WPB from SA 105, normalized, is applied. NUREG-0577, Table 4.4 assigns an (average NDT + 1.3σ) NDT of -5°F. ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 25°F.

(e) Flued Head (44 in. x 32 in.)

ASME Code steel specification SA 105, quenched and tempered via TVA specification 1521-B, 5-1/4-in. web thickness is applied. NUREG-0577, Table 4.4 would assign the quenched and tempered material a T_{NDT} in the population below the -28°F NDT cited for normalized material. Assuming a -28°F T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 27°F.

(f) Main Steam Isolation Valve (32 in. x 32 in.)

Body

ASME Code steel specification SA 216, Grade WCB, normalized and tempered, 2-3/16-in. minimum wall, is applied. NUREG-0577, Table 4.4 would assign a T_{NDT} in the population below the 35°F average NDT cited for 2-1/2-in. to 5-in. thick material. Assuming a 35°F T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 65°F.

Cover

ASME Code steel specification SA 105, quenched and tempered, 6.3-in. minimum thickness, is applied. Assuming a -28°F T_{NDT} as for item 2(e) (flued head), ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 36°F.

Poppet

ASME Code steel specification SA 105, quenched and tempered, 9.875-in. minimum thickness, is applied. Assuming a -28°F T_{NDT} as for item 2(e) (flued head), ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 50°F.

Pilot Poppet

ASME Code steel specification SA 182, Grade F6, oil quenched and tempered (1275°F), 2-in. minimum thickness, is applied. Although the concept of T_{NDT} does not apply to this material, the inference of suitable toughness at its LSMT is made from empirical toughness test data for 1275°F temper material available from steel producers.

Bolting

ASME Code steel specification SA 540, Grade B23 is applied for bolts and nuts. NUREG-0577, Table 4.6 categorizes this material as having the least susceptibility to brittle failure.

3. MAIN FEEDWATER SYSTEM CONTAINMENT PRESSURE BOUNDARY MATERIALS

(a) Lowest Service Metal Temperature

TVA analysis (letter from J. W. Hufham, TVA, to E. Adensam, NRC, dated December 10, 1984), identifies 100°F as the LSMT.

(b) Pipe (Piece Mark 03A-FW-1)

ASME Code steel specification SA 333, Grade 6, 18-in. Schedule 80 (0.937-in. wall) and 16-in. Schedule 80 (0.843-in. wall), is applied. SA 333 demands at least normalization. NUREG-0577, Table 4.4, in a worst-case characterization as "mild steel," would assign a 40°F T_{NDT}, the average NDT for "mild steels." Assuming a 40°F T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 70°F.

(c) Fittings (On Piece Mark 03A-FW-1-Con. Red.)

ASME Code steel specification SA 333, Grade 6, 18 in. (0.937-in. wall) x 16 in. (0.843-in. wall) is applied via SA 420, Grade WPL 6. The analysis for item 3(b) (pipe) applies.

(d) Flued Head: 30 in. x 18 in. (On Piece Mark 03A-FW-1)

ASME Code steel specification SA 105, quenched and tempered via TVA specification 1521-B, 4-1/2-in. web thickness, is applied. NUREG-0577, Table 4.4 would assign the quenched and tempered material a T_{NDT} in the population below the -28°F average NDT cited for normalized material. Assuming a -28°F T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of -24°F.

(e) Main Feedwater Isolation Valve

Body

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 1.093-in. minimum wall, is applied. NUREG-0577, Table 4.4 would assign an (average NDT, 1.3σ) NDT of 10°F. ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 40°F.

Bonnet

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 1.562-in. minimum thickness, is applied. NUREG-0577, Table 4.4 would assign a T_{NDT} in the population below the 35°F NDT cited for 2-1/2-in. to 5-in. thick material. Assuming a 35° T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 65°F.

Wedge

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 1.75-in. minimum thickness (letter from L. Mills, TVA, to E. Adensam, NRC, dated August 22, 1983), is applied. NUREG-0577, Table 4.4 would assign a T_{NDT} in the population below the 35°F average NDT cited for 2-1/2-in. to 5-in. material. Assuming a 35°F T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 65°F.

Bolting

Bolts: ASME Code steel specification SA 564, Grade 630, H 1100, 1-1/2-in. diameter, is applied. The concept of T_{NDT} does not apply to this material. However, load distribution under the limiting environmental conditions will assign stresses to individual bolting at a level unlikely to induce brittle fracture.

(f) Main Feedwater Swing Check Valve (Serial No. D66295 Typical)

Body

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 1.56-in. minimum wall, is applied. NUREG-0577, Table 4.4 would assign a T_{NDT} in the population below the 35°F average NDT cited for 2-1/2-in. to 5-in. material. Assuming a 35°F

T_{NDT}, ASME Code Summer 1977 Addenda, Class 2 rules assign a PLSMT of 65°F.

Cover

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 1.56-in. minimum thickness, is applied. Using analysis for body (above) assigns a PLSMT of 65°F.

Disc

ASME Code steel specification SA 352, Grade LCB, normalized and tempered (letter from L. Mills, TVA, to E. Adensam, NRC, dated December 23, 1981), 2.375-in. minimum thickness. Same analysis as for body (above) assigns a PLSMT of 65°F.

Bolting

Studs: ASME Code steel specification SA 320, Grade L7 is applied. Nuts: ASME Code steam specification SA 194, Grade 7 is applied. NUREG-0577, Table 4.6 would categorize these materials as having the least susceptibility to brittle fracture.

SUMMARY

On the basis of the above material, the staff concludes that reasonable assurance has been provided that the Watts Bar Units 1 and 2 reactor containment pressure boundary materials will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are satisfied.

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16. ABSTRACT (200 words or less)
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