NUREG-0847
Supplement No. 3

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 1985

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ABSTRACT

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982), Supplement No. 1 (September 1982), and Supplement No. 2 (January 1984) 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) and Supplement No. 2 (SSER 2, January 1984), 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 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 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 the unresolved safety issues that were discussed in the SER. Appendix E is a list of principal contributors to this supplement. No changes in SER Appendices D, F, and G 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 five 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.

*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	Under review	--
(3) Preservice and inservice pump and valve test program	Under review	--
(4) Seismic and environmental qualification of equipment	Seismic - partially resolved (SSER 3) Environmental - under review	3.10 --
(5) Preservice and inservice inspection program	Under review	--
(6) Pressure-temperature limits for Unit 2	Under review	--
(7) Model D-3 steam generator preheater tube degradation	Under review	--
(8) BTP CSB 6-4	Resolved (SSER 3); see License Condition (8)	6.2.4
(9) H_2 analysis review	Under review	--
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 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	--
(13) Quality classification of diesel generator auxiliary system piping and components	Under review	--
(14) Diesel generator auxiliary system design deficiencies	Partially resolved (SSER 3)	9.5.7
(15) Physical Security Plan	Resolved (SSER 1)*	--

*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.

Issue	Status	Section
(16) Boron dilution event	Under review	--
(17) Q List	Resolved (SSER 2)*	--

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 14 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	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)	--
(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	--
(10) Thermal performance analysis	Resolved (SSER 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)	--
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	--
(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	--
(15) Natural circulation tests	Awaiting information	--
(16) Dump valve testing	Resolved (SSER 2)	--
(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	--
(19) Compliance with GDC 51	Under review	--
(20) Insulation survey (sump debris)	Resolved (SSER 2)	--
(21) Safety system set point methodology	Under review	--
(22) Steam generator water level reference leg	Resolved (SSER 2)	--
(23) Containment sump level measurement	Resolved (SSER 2)	--
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Under review	--
(26) Availability of offsite circuits	Resolved (SSER 2)	--
(27) Nonsafety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	--
(28) Low and/or degraded grid voltage condition	Awaiting verification of test results	--
(29) Diesel generator reliability qualification testing	Awaiting verification of acceptability of test results	--
(30) Diesel generator battery system	Resolved (SSER 2)	--

Issue	Status	Section
(31) Thermal overload protective bypass	Resolved (SSER 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)	--
(34) Testing Class 1E power systems	Resolved (SSER 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	--
(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	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 No. 1 to the SER, 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	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3
(2) Preservice/inservice testing of pumps and valves	Under review	--

Condition	Status	Section
(3) Detectors for inadequate core cooling (II.F.2)	Awaiting information	--
(4) Inservice Inspection Program	Under review	--
(5) Installation of reactor coolant vents (II.B.1)	Awaiting verification of installation	--
(6) Accident monitoring instrumentation (II.F.1)		
(a) noble gas monitor	Awaiting information	--
(b) iodine particulate sampling	Awaiting information	--
(c) high range incontainment radiation monitor	Awaiting verification of installation	---
(d) containment pressure	Awaiting verification of installation	--
(e) containment water level	Awaiting verification of installation	--
(f) containment hydrogen	Awaiting verification of installation	--
(7) Modification to chemical feedlines	Under review	--
(8) Containment isolation dependability (II.E.4.2)	Under review	6.2.4
(9) Hydrogen control measures (II.B.7)	Under review	--
(10) Status monitoring system	Unchanged (SER)	--
(11) Installation of acoustic monitoring system (II.D.3)	Awaiting verification of installation	--
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	--
(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)	--
(19) Postaccident sampling system (II.B.3)	Partially resolved (SSER 3)	9.3.2
(20) Fire Protection Program	Awaiting information	--
(21) Performance testing for communications systems	Under review	--
(22) Diesel generator reliability (NUREG/CR-0660)	Awaiting verification of modifications	--
(23) Secondary water chemistry monitoring and control program	Unchanged (SER)	--
(24) Primary coolant outside containment (III.D.1.1)	Awaiting information	--
(25) Independent safety engineering group (I.B.1.2)	Awaiting information	--
(26) Use of experienced personnel during startup	Unchanged (SER)	--
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2)	Under review	--
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7)	Under review	--
(29) Modifications to emergency operating instructions (I.C.8)	Awaiting verification of modifications	--
(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)	Under review	--

Condition	Status	Section
(33) Voiding in the reactor coolant system (II.K.2.17)	Under review	--
(34) PORV isolation system (II.K.3.1, II.K.3.2)	Under review	--
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Under review	--
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31)	Under review	--
(37) Control room design review (I.D.1)	Awaiting verification of modifications	--
(38) Physical Security Plan	Unchanged (SER)	--

There are additional issues for which license conditions have been determined to be desirable to ensure that staff requirements are met during plant operation. These license conditions are tabulated below with the relevant section of this report.

(39) Control of heavy loads (NUREG-0612)	Added in SSER 3	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28)	Added in SSER 3	15.3.6
(41) Steam Generator Tube Rupture	Added in SSER 3	15.4.3
(42) Loose parts monitoring system	Added in SSER 3	4.4.5

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.8 Design Basis for Subsurface Hydrostatic Loading

The SER lists the design-basis groundwater level for the essential raw cooling water (ERCW) pipeline as a confirmatory issue. The applicant had not provided any groundwater information for the ERCW pipeline and in lieu of factual data, the staff conservatively assumed ground elevation as the design groundwater level for the ERCW pipeline.

The applicant subsequently installed eight piezometers along the ERCW pipeline to establish groundwater levels. These data were also used in conjunction with other long-term groundwater and rainfall records to project an approximate 25-year (probability of 0.04 per year) groundwater level for use with combined seismic events for design purposes. The 25-year groundwater level was chosen on the basis of the combined events criteria in American Nuclear Society (ANS) Standard 2.8, "Standards for Determining Design Basis Flooding at Power Reactor Sites." Although that standard does not specifically address the combination of a seismic event and a high groundwater level, it does consider a safe shutdown earthquake (SSE) combined with a 25-year flood to be an acceptable combination when considering dam failure floods.

Amendment 50 to the Watts Bar Final Safety Analysis Report (FSAR), Section 2.5.4.6, contains a complete description of the analyses used to determine the projected 25-year groundwater level for the ERCW pipeline. FSAR Table 2.5-73 shows the estimated 25-year groundwater elevation for each of the eight ERCW piezometers. The staff has reviewed the applicant's analyses and estimated 25-year groundwater elevations (listed in Table 2.1) and finds them to be acceptable and to meet the requirements of General Design Criterion (GDC) 2. Therefore, Confirmatory Item (1) is resolved.

2.5 Geology and Seismology

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.2 Subsurface Conditions and Geotechnical Properties

In the SER, the staff stated that it would perform an audit of Tennessee Valley Authority's (TVA's) calculations to confirm the adequacy of the procedures used for accounting for the material and geometric damping in the soil-structure interaction (SSI) analysis of the soil- and pile-supported structures. The staff performed this audit on September 24, 1982, and, on the basis of its review of the information during the audit, the staff finds that the applicant used the following analysis techniques to analyze the soil-supported and pile-supported Category I structures.

Structure	Analysis technique
Diesel generator building	Lumped mass modal analysis with equivalent soil springs
Waste-packaging area	Lumped mass modal analysis with equivalent soil springs
Condensate demineralizer waste evaporator (CDWE) building (pile supported)	Lumped mass modal analysis with equivalent soil springs and finite element analysis (LUSH (Lysmer et al., 1974))
Refueling water storage tank	Lumped mass modal analysis with equivalent soil springs and finite elements analysis (FLUSH (Lysmer et al., 1975))

The calculated soil geometrical and material damping and the design damping values used in the analyses of various structures are shown in Table 2.2. In the SSI analyses, the applicant varied the shear wave velocity of the soil by a minimum of ±30% to account for uncertainties in the dynamic properties of the soil. On the basis of its review of the damping values used in the analyses (Table 2.2) and in view of the applicant's use of ±30% variation in the values of the soil shear wave velocities, the staff finds that the applicant has used reasonable assumptions for dynamic modulus and damping values in his soil-structure interaction analyses and, therefore, considers Confirmatory Item (2) closed.

2.5.4.3 Foundation Construction and Evaluation

Foundations on Rock

In March 1982, the applicant informed the staff that the intended design differential settlement of 1 in. between rock-supported structures was not actually used in the design of piping and electrical components passing between adjacent rock-supported structures. The affected components include heating, ventilation, and air conditioning (HVAC) duct, cable trays, Category I piping, instrument lines, and conduits that pass between the reactor building of Unit 1 and the auxiliary building and between the reactor building of Unit 2 and the auxiliary building.

By letter dated September 14, 1983, the applicant notified the staff that the capability of individual safety-related components located at the interfaces of rock-supported structures has been evaluated. On the basis of this evaluation, the applicant concluded that all affected safety-related HVAC duct, cable trays, and their supports can safely withstand a 1-in. differential settlement. In addition, the applicant had previously indicated in the Watts Bar FSAR that the expected differential settlement of adjacent rock-supported structures, based on settlement computations, is less than 0.3 in. On the basis of this information and the applicant's analysis presented to the staff on September 16, 1983, the staff agrees with the applicant that he has adequately addressed settlement effects on the Category I piping, conduit, and instrumentation lines. The staff finds the applicant's analysis and the results acceptable and considers Confirmatory Item (4) closed.

Concrete Wingwalls and Sheetpile Retaining Walls

As discussed in the SER, the applicant informed the staff by letter dated April 27, 1982, that he had discovered errors in the final design calculations for the two soil-anchored sheetpile walls that extend from each end of the intake pumping station toward the main plant and retain backfill surrounding Category I pipes. On the basis of revised calculations, the applicant concluded that the sheet-pile walls will be overstressed in bending during an SSE. The applicant committed to institute corrective field measures to reinforce these walls to eliminate such overstressing. The staff found this approach reasonable and acceptable.

During an audit of the applicant's calculations on September 24, 1982, the staff reviewed the design procedures, assumptions, and input soil parameters used by the applicant for the tied sheetpile wall design. At this meeting, the applicant informed the staff that he has investigated the stability of sheetpile walls using saturated soil to an elevation of 700 ft on one side, with no water on the opposite side. The design is based on Coulomb's equation for calculating earth pressures using angle of internal friction, d, for the soil of 32°. The staff informed the applicant that the design procedures, assumptions, and input soil parameters used in the analyses are generally acceptable. However, the staff requested the applicant to investigate further the effects of the following conditions on the stability of sheetpile walls:

(1) reduction in the available soil passive pressure when the water level rises above the normal water level as a result of flooding

(2) blocked weep holes

The applicant provided responses to this staff request by letters dated November 30, 1982, and July 21, 1983. On the basis of its review of this response and previously submitted information, the staff finds that the applicant has adequately analyzed the effect of potential reduction in the available soil passive pressures when the water level rises above the normal water level as a result of flooding. The applicant has determined that under these conditions, the sheetpile wall has a factor of safety of 1.00, which the staff considers adequate for these loading conditions. In the letter dated July 21, 1983, the applicant also stated that he has investigated the stability of sheetpile walls for the case where the weep holes were not properly functioning and has determined that the sheetpile walls would have a factor of safety of 1.68 for this case. The staff finds these design loading assumptions and analyses results reasonable and acceptable. On the basis of the results of its FSAR review, the confirmatory audit of the applicant's calculations on September 24, 1982, and a review of the information submitted by the applicant since the audit meeting, the staff concurs with the applicant's conclusion that the sheetpile walls have an adequate margin of safety under the design static and dynamic loading conditions and, therefore, considers Confirmatory Item (3) closed.

2.5.4.4 Liquefaction Potential

ERCW Pipeline and Class 1E Electrical Conduit Support

During a meeting on May 20, 1983, the applicant informed the staff that he had reassessed the liquefaction potential of soils below the water table in all the borings along the ERCW pipeline and electrical conduit duct banks. For these analyses, the applicant used the Seed and Idriss (1971) procedure and assumed a maximum horizontal peak acceleration of 0.4g at the ground surface. The applicant found, in using this analysis criterion, that soils may liquefy in several areas along the ERCW pipelines. One of the susceptible areas was found to be continuous beneath sloping ground close to the ERCW pumping station. In other areas, the liquefaction-susceptible soils are either isolated or under very flat ground. The applicant indicated that he would provide field remedial measures (an underground barrier) to stabilize the area between the pumping station and boring #141. The applicant also agreed to address the effects of postearthquake settlement of liquefied soils.

The staff completed its review of the applicant's use of a maximum ground acceleration of 0.4g at the soil surface for evaluating liquefaction potential. By letter dated June 22, 1983, the staff informed the applicant that his use of a peak horizontal acceleration of 0.4g for liquefaction analysis is appropriate and acceptable.

The staff has reviewed the applicant's use of the Seed and Idriss (1971) method for liquefaction analysis and concluded that the use of this method for evaluating liquefaction potential at the Watts Bar facility and the conclusions based on this analysis are acceptable.

The field remedial measures described in Amendment 50 of the Watts Bar FSAR show the construction of underground barriers along the ERCW pipelines. The purpose of these measures is to contain the soil beneath the pipelines in case of any liquefaction in this area. The applicant designed the barriers to withstand a maximum ground acceleration of 0.4g. The compacted soil embankment is about 100 ft wide at the base and about 25 ft deep from the ground surface to underlying shale deposit. The distance of the barriers from the ERCW pipelines varies from 50 ft to 500 ft. The compacted barriers are formed by compacting native borrow material from the location of the barriers. The compaction criterion is 100% standard Proctor density. The applicant has filled ground surface depressions in the area between the location of the pipelines and the top of the barriers to preclude potential ground movement in this area. The applicant also has provided appropriate modification to accommodate anticipated settlement effects.

The staff and its consultant, the U.S. Army Corps of Engineers, Tulsa District, have reviewed the applicant's as-built drawings submitted with Amendment 50 of the FSAR. On the basis of these reviews, the staff agrees with the applicant that the underground barriers will provide sufficient confinement to any liquefied soil in the affected area. Therefore, the staff concludes that Open Item (1) is closed.

Table 2.1 Summary of groundwater level estimates for essential raw cooling water (ERCW) pipeline

ERCW piezometer number	25-year groundwater estimate
P1	702.9
P2	717.6
P3	716.8
P4	714.4
P5	712.5
P6	710.2
P7	718.4
P8	723.4

Table 2.2 Foundation damping for lumped mass modal analysis

Structure	Mode no.	Predominant motion	Damping (% of critical)		
			Geometrical	Material	Design
Diesel generator building	1	Horizontal translation of foundation	60	10	10
	2	Rocking and horizontal translation of foundations	30	10	10
	3	Primary building deformation	--	5	10
Waste packaging building	1	Rocking and horizontal translation of foundation	20	10	10
	2	Horizontal translation of foundation	20	10	10
	3	Primary building deformation	--	5	10
Condensate demineralizer waste evaporator building	1	Rocking and horizontal translation of foundation	--	5	5
	2	Rocking and horizontal translation of foundation	--	5	5
	3	Vertical	--	5	5
Refueling water storage tank	1	Horizontal translation of foundation	--	10	10

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

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

As was noted in the SER, staff acceptance of the emergency raw cooling water system (ERCWS) was contingent on a satisfactory upgrading or replacement of portions of the ERCWS that service areas containing equipment required for plant safety. These portions of the ERCWS were incorrectly classified and installed to nonseismic Category I design requirements. To meet the guidance of Position C.1.g of Regulatory Guide (RG) 1.29, the portions of the ERCWS required for plant safety should be designed to withstand the effect of a safe shutdown earthquake (SSE) and remain functional, that is, seismic Category I. These portions of the ERCWS include heating, ventilation, and air conditioning (HVAC) air cooling units and chiller packages that provide cooling to areas that contain safety-related equipment such as safety injection pumps, containment spray pumps, residual heat removal pumps, and auxiliary feedwater pumps.

The applicant has reviewed the documentation for HVAC equipment that cools areas where equipment required for plant safety is located and has verified that this HVAC equipment meets the seismic Category I requirements of the applicable design criteria and is in accordance with a quality assurance program that is in compliance with 10 CFR 50, Appendix B.

The applicant has also reviewed the documentation for HVAC equipment that cools areas where equipment not required for plant safety is located and has verified that although this HVAC equipment is not designed to seismic Category I requirements, it is seismically designed to maintain the pressure boundary integrity of the ERCWS.

In Amendment 49 to the FSAR, the applicant revised Tables 3.2-2 and 3.2-2a to reflect the current status of the ERCWS. The staff has reviewed FSAR Section 3.2, flow diagram Figures 9.2-1 through 9.2-4a, and the series of progress reports on the improper classification of the ERCWS that were issued by the applicant between April 24, 1981, and September 28, 1983 (letters dated Apr. 24, June 8, July 14, Sept. 2, and Dec. 9, 1981, and Feb. 17, Aug. 11, Oct. 19, Apr. 27, and Sept. 28, 1983), and finds the seismic classification of the ERCWS acceptable. The staff will also verify that these actions have been implemented for the ERCWS. Pending verification of these actions, the staff concludes that those portions of the ERCWS that are required to withstand the effects of an SSE and remain functional are properly classified as seismic Category I in accordance with RG 1.29. This constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2 and is, therefore, acceptable.

3.2.2 System Quality Group Classification

As was noted in the SER, staff acceptance of the ERCWS was contingent on a satisfactory upgrading or replacement of portions of the ERCWS that service areas containing equipment required for plant safety. These portions of the ERCWS were incorrectly classified and installed to Quality Group D standards. To meet the guidance of Position C.2.1 of RG 1.26, the portions of the ERCWS required for plant safety should be constructed to Quality Group C standards, that is, American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code" (ASME Code), Section III, Class 3.

The portion of the ERCWS that was installed incorrectly was piping between the first two isolation valves of the chillers/coolers. Approximately 1,200 ft of ERCWS piping was installed incorrectly as TVA Class M (Quality Group D) instead of to Quality Group C standards, and about 400 of 900 welds were made during this installation without welder identification. The applicant has replaced all of the affected piping with new piping that is constructed to Quality Group C standards. As was noted in Section 3.2.1, the staff has reviewed FSAR Section 3.2, the appropriate flow diagrams, and the series of progress reports on the improper classification of the ERCWS and finds the Quality Group C classification of the replacement piping acceptable. The staff will also verify that these actions have been implemented for the ERCWS. Pending verification of these actions, the staff concludes that those portions of the ERCWS that are required to be constructed to Quality Group C standards are properly classified in accordance with RG 1.26. This constitutes an acceptable basis for satisfying the requirements of GDC 1 and is, therefore, acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Steel Containment

In the SER, the staff requested the applicant to provide an additional verification of his containment buckling analytical methodology. This request was made because new information, discussed below, had been obtained through NRC-sponsored research programs (NUREG/CR-2165). By letter dated May 16, 1984, the applicant stated that he has reviewed the NRC research results and concluded that the new information from the research had been already adequately accounted for in his calculation.

The most important new information resulting from the research was that reinforcement of an opening according to the area replacement method of ASME Code, Section III, might not restore original buckling strength. However, the applicant stated that the design by the Watts Bar steel containment contractor, Chicago Bridge and Iron, compensated for the weakness of the area replacement method by providing extra reinforcing members around the opening. This extra reinforcing consisted of stiffening above and around the opening so as to avoid stress concentration. Also, the vertical stiffening was placed in two spans between ring stiffeners. In view of the fact that the applicant provided an additional safety margin for loading to account for potential shortcomings in the calculational methodology for asymmetrical shell buckling, the staff has concluded that the applicant's evaluation is acceptable and, therefore, considers Confirmatory Item (8) to be closed.

3.9 Mechanical Systems and Components

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

3.9.3.3 Design and Installation of Pressure Relief Devices

Relief and Safety Valve Test Requirements

As required by NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.D.1, all pressurized-water-reactor (PWR) plant licensees and applicants are required to demonstrate that their pressurizer safety valves, power-operated relief valves (PORVs), PORV block valves, and all associated discharge piping will function adequately under conditions predicted for design-basis transients and accidents. In response to this requirement, the Electric Power Research Institute (EPRI), on behalf of the PWR Owners Group, has completed a full-scale valve testing program, and the PWR Owners Group submitted these test results to the NRC by letter dated April 1, 1982. Additionally, each PWR plant applicant for an operating license (OL) was required to submit a report by fuel loading that would demonstrate the operability of these valves and the associated piping.

On July 22, 1983, the applicant responded to this requirement with a submittal that contains information from the EPRI valve test program results that applies to Watts Bar Units 1 and 2. The submittal also states that the safety and relief valve discharge piping and supports are being modified to ensure functionability.

The staff has not completed a detailed review of the applicant's submittal; however, on the basis of a preliminary review, the staff finds that the general approach of using the EPRI test results to demonstrate operability of the safety valves, PORVs, and PORV block valves is acceptable. In the submittal the applicant notes that at Watts Bar Units 1 and 2, safety valves, PORVs, and PORV block valves are used that are the same size and model as those that performed satisfactorily for test sequences considered representative or that bound conditions to which the valves at Watts Bar Units 1 and 2 could be exposed.

In summary, on the basis of a preliminary review, the staff has concluded that the applicant's general approach to responding to this TMI item is acceptable and provides adequate assurance that the Watts Bar reactor coolant system overpressure protection systems can adequately perform their intended functions until the staff completes its detailed review. If this detailed review reveals that modification or adjustments to safety valves, PORVs, PORV block valves, or associated piping are needed to ensure that the overpressure protection systems can perform their intended functions, the staff will require that the applicant make appropriate modifications. Therefore, the staff considers License Condition (1) to be resolved.

3.10 Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment

In Section 3.10 of Supplement No. 1 and during the site audit of safety-related electrical and mechanical equipment, several generic and specific concerns regarding seismic and dynamic qualification were identified. These generic and specific concerns are discussed in Supplement No. 1 and a letter to the applicant dated September 23, 1982. In an effort to resolve the generic and specific concerns, the applicant provided post-audit information to the NRC in submittals dated December 1, 1982, June 10, 1983, and February 9, May 17, May 25, and June 19, 1984. The current status of each concern, along with the disposition, when applicable, is discussed in the following sections.

3.10.1 Generic Concerns

The status of the generic concerns is:

(1) Single-frequency and single-axis testing was frequently used to qualify electrical equipment. The applicant was requested to provide justification for using this form of testing for all equipment qualified in this manner. The applicant responded by referring to qualification reports (Westinghouse Topical Reports WCAP-8673, -8694, and -8624) that were previously submitted to the NRC in regard to this issue. The applicant also referred to an NRC memorandum from J. P. Knight to R. C. DeYoung, dated August 26, 1976, in support of a supplemental seismic demonstration program to ensure that adequate margin exists for electrical equipment tested by Westinghouse before May 1974. The staff received the necessary documents and further discussed the issue during a meeting with Westinghouse representatives in Bethesda, Maryland, on April 28, 1983. During the meeting, the applicant was requested to provide verification of analysis, including the following:

    (a) The effect of directional coupling should be considered if applicable.

    (b) Where applicable, verification should be provided that acceleration at each device location is less than 0.95g because relay chatter at higher acceleration levels is expected.

    (c) The test response spectrum (TRS) envelopes the required response spectrum (RRS) for all directions because the RRS for different directions can be different for buildings or cabinets.

    The applicant was requested to verify this in writing for equipment at Watts Bar procured from Westinghouse. By letter dated May 17, 1984, the applicant responded to this request. The nature of the response was acceptable, but the scope was not. The response to the concern about directional coupling was limited to audit items. The response to the concern about acceleration level was limited to devices mounted in the solid-state protection system. The response to the envelopment concern did not indicate that Seismic Qualification Review Team forms for all safety-related equipment supplied by Westinghouse were provided. To be acceptable, the responses must address the concerns for all safety-related equipment supplied by Westinghouse. This issue remains open.

(2) In numerous cases the RRS were not broadened at the peaks to account for uncertainty in the prediction of natural frequencies of the supporting structures. In addition, the TRS should include sufficient margin to

account for uncertainty in the manufacturing process and testing apparatus. The applicant's December 1, 1982, submittal stated that requirements were imposed to ensure ±10% broadening for balance-of-plant (BOP) equipment, and it also addressed RRS broadening for certain specific nuclear steam supply system (NSSS) items as discussed in Section 3.10.2. In that submittal the applicant did not adequately demonstrate that the broadening requirement had actually been applied or considered in qualifying all safety-related equipment. To better ensure that the broadening requirement would pose no difficulties for any equipment, the applicant was requested to evaluate the effects of peak broadening for a sampling of equipment where RRS were not broadened. The applicant stated in the June 10, 1983, submittal that the qualifications for all Watts Bar safety-related equipment (BOP and NSSS) included consideration of spectra peak broadening. In all cases the spectra were broadened by ±10% throughout the critical frequency range. Several examples of qualifications where such broadening was applied, which were evidently representative of all safety-related equipment, were provided with the submittal. On the basis of the applicant's assurance that this policy was used throughout the qualification program, this issue is closed.

(3) In numerous cases, particularly for electrical cabinets, equipment is field mounted by welding but test mounted by bolting. The applicant was asked to identify and justify all such cases. In the December 1, 1982, submittal, the applicant maintained that the welding used in all configurations is stronger than the corresponding bolting used in the tests. The applicant stated that more metal area is used in the welded connections and that, if anything, the welded connections will increase rigidity of the cabinet response. The staff requested that the applicant further evaluate whether an increase in cabinet rigidity may shift the cabinet frequency into a more critical range of the response spectrum curve, and also whether using welding could adversely affect damping in the system. Additionally, instead of relying on the results of purely analytical predictions, the staff required that the applicant select a cabinet and verify its natural frequencies by in situ testing. In the June 10, 1983, submittal, the applicant asserted that the base steel assemblies used to mount the cabinets are rigid and that the welded-versus-bolted configuration would result in a very small difference in equipment seismic response. The applicant indicated that results of Westinghouse tests show that welding can be substituted for bolting in these base steel assemblies with a 6% or smaller change in equipment resonance frequencies. The applicant was requested, however, to demonstrate by in situ tests on a Watts Bar cabinet that the response of the cabinet is essentially unaffected by the difference in mounting. By letter dated May 17, 1984, the applicant reiterated his position. Since no new information has been presented, the request for an in situ test is still felt appropriate. Discussion under equipment-specific Item 5, "Main Control Board," in Section 3.10.2 is pertinent for the resolution of this issue. This issue is open.

(4) Many safety-related equipment items, such as the insulation of motors, transformers, and other electric devices, are age sensitive with respect to their seismic performance. To ensure that safety-related equipment is seismic resistant throughout the plant life, a detailed program of surveillance and maintenance should be provided for staff review and approval.

The applicant's submittal of May 17, 1984, did not address the equipment located in a mild environment. Therefore, this issue remains open.

(5) The applicant was requested to provide a statement describing how damping values were used in qualifications for seismic input as obtained from TVA floor response spectra. The applicant stated that Westinghouse uses a 5% critical damping value in the development of test response spectra. The TRS is then compared to an RRS of equal or less damping to show acceptability. This approach complies with the recommendations of Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975 and is therefore satisfactory. This concern is closed.

(6) Because nozzle loads exerted on safety-related equipment were frequently not mentioned in the qualification documentation, the applicant was requested to show how nozzle loads were considered in qualification. The applicant responded that the nozzle loads criteria imposed on equipment suppliers ensure that the equipment can sustain specified loads. These same criteria are imposed on the piping analysis in the form of load limits at the equipment interface. Problems that arise in the imposition of these criteria are resolved on a case-by-case basis. This approach ensures that acceptable values for nozzle loads are not exceeded. This concern is closed.

(7) The status of safety-related equipment that is not yet seismically qualified should be periodically updated and fully qualified before fuel loading. The applicant has confirmed his completion of seismic qualification of safety-related equipment by letter dated May 17, 1984. This issue is closed.

3.10.2 Specific Concerns

The status of the specific concerns is:

(1) <u>Reactor Trip Switchgear</u>

(a) The applicant was requested to demonstrate that the welded field mounting is structurally as sound as the bolted lab mounting. Resolution of this item relies on the response to Item 3 in Section 3.10.1 and will be evaluated under that generic issue.

(b) A box-shaped cable support beam that extends downward from the floor above and contacts the cabinet near its top should be addressed in the cabinet qualification. By letter dated February 9, 1984, the applicant stated that adequate clearance between the beam and cabinet has been provided. In addition, the staff requested that a walkdown audit on a sampling basis be conducted before fuel loading to confirm that field modifications of this type have been made. This walkdown audit (including all corrective actions) should be completed before fuel loading, and the results should be maintained by the applicant in an auditable manner and be available for NRC review. The applicant was requested to provide written confirmation that this has been performed before fuel loading. The applicant committed to provide the results of the walkdown in the May 17, 1984, submittal. Written confirmation of the completion of these actions was provided in the applicant's May 25, 1984, submittal. This issue is closed.

(c) The RRS should be peak broadened to adequately qualify the equipment. In his submittal, the applicant showed that the TRS sufficiently envelopes an RRS that has been broadened 10%. This issue is closed.

(d) The applicant was requested to provide an explanation as to how damping was used to qualify for operational basis earthquake (OBE) and SSE seismic input. The applicant's response was adequately encompassed by his response to Item 5 in Section 3.10.1. This issue is closed.

(2) Reactor Protection System Cabinet

(a) The applicant was asked to demonstrate that field mounting is as adequate as lab mounting. Resolution relies on the response to Item 3 in Section 3.10.1 and will be evaluated under that generic issue.

(b) The applicant was requested to evaluate the effects of mounting the cabinets close together. The applicant responded that a minimum of 1/2-in. clearance between cabinets is required. The cabinets have been inspected and adjusted accordingly. By letter dated February 9, 1984, the applicant provided written confirmation that this work has been completed. This issue is closed.

(c) The applicant was requested to evaluate the degree of amplification that occurred in the cabinet response motion during tests to clearly justify single-frequency testing. The applicant replied by referring to his response to Item 1 in Section 3.10.1. The applicant also stated that the RRS, when peak broadened by 10%, is still enveloped by the TRS. Resolution of this item relies on a satisfactory response to Item 1 in Section 3.10.1 and will be evaluated under that generic issue.

(3) Charging/Safety Injection Pumps

(a) The applicant was asked to provide a detailed analysis of the suction nozzle connection using an approach appropriate for the nozzle geometry. The applicant responded in the December 1, 1982, submittal that the suction nozzle was analyzed according to the requirements of ASME Code, Section III, Appendix A2212. This approach does not, however, sufficiently address effects from externally applied nozzle loads or consider the nozzle geometry. The applicant was therefore requested to further evaluate the nozzle. The June 10, 1983, submittal identified external loads that were applied to the nozzle, which included seismic, deadweight, and operating loads. The submittal did not, however, clarify the acceptability of using the Bijlaard method to perform a stress analysis at the connection when the nozzle geometry does not conform to guidelines established for the use of this technique. Thus, the applicant was requested to justify the use of the Bijlaard technique for this case. In the May 17, 1984, submittal, the applicant responded that the Bijlaard technique was not used. This being the case, justification for the technique is not necessary. Furthermore, the concern about the shortcomings of the reference to ASME Code, Section III, Appendix A-2212, was alleviated by a description of the nozzle analysis performed that was included in the June 10, 1983, submittal. This issue is closed.

(b) The applicant was requested to provide a comparison of the nozzle loads used in the pump analysis and those obtained from piping analysis. The applicant's December 1, 1982, submittal stated that piping analysis results showed that piping loads imposed on the pumps exceeded the nozzle load limit criteria provided to the pumps' vendor. An evaluation by the vendor of the revised nozzle loads showed the pumps to be acceptable. Therefore, this issue is closed.

(4) Control Rod Drive Mechanism

(a) The qualification documentation did not identify the load combinations used. The applicant replied that the faulted condition used included loss-of-coolant-accident (LOCA), safe shutdown earthquake (SSE), and deadweight loads. The LOCA and SSE loads were combined by the square-root-of-the-sum-of-the-squares method, and the resultant was added absolutely to the deadweight loads. Because this is an acceptable methodology for determining load confirmation, this issue is closed.

(b) The applicant was requested to verify that Westinghouse response spectra envelope the Watts Bar floor-level response spectra. The applicant's December 1, 1982, submittal stated that an analysis of the control rod drive mechanism (CRDM) using Watts Bar response spectra was performed in the same manner as the generic analysis. The resulting stresses in the CRDM from the plant-specific analysis are enveloped by those from the generic analysis. Thus, the CRDM qualifies according to the Watts Bar seismic criteria. Therefore, this issue is closed.

(c) The applicant was requested to provide a comparison of natural frequencies identified during pluck tests on the CRDM with calculated natural frequencies. The applicant's December 1, 1982, submittal stated that pluck tests were performed on a Westinghouse Model 1-105 CRDM. The natural frequencies identified by tests were thus somewhat higher than those calculated for the longer L-106A model. The difference in frequencies was accounted for by amplifying the response of the longest CRDM by the ratio between the peak acceleration value and the acceleration value used in the original analysis. The staff requested that the applicant supply further information on the qualification safety margin so that the significance of any error introduced by this approach could be assessed. In the June 10, 1983, submittal, the applicant explained that the responses at all frequencies in the 4.26- to 10.8-Hz range were amplified by the ratio between the peak spectrum acceleration in this range and the spectrum acceleration at 4.26 Hz (the lowest calculated natural frequency). This was done to account for uncertainties in the determination of natural frequencies in this range.

Although this approach is likely to be conservative, the applicant was requested to provide the safety margin for this piece of equipment. The May 17, 1984, submittal indicated the margin is 1.10. The staff finds this acceptable and considers this issue closed.

(5) Main Control Board

(a) For the qualification performed the panel was assumed to be fixed at its base. However, the panel is attached to the floor with spot welds along only the inside edge of an angle-shaped member at the base of the panel. The applicant was requested to consider potential flexibility that could be introduced by freedom of the outside edge of this member. The December 1, 1982, submittal stated that the critical natural frequency of the panel was calculated to change from 21.1 Hz to 19.7 Hz as a result of this flexibility, which does not affect the panel qualification. The applicant was requested to furnish these calculations for review. In the June 10, 1983, submittal, the applicant furnished these calculations for review. In these calculations, the stiffness of an equivalent rotational spring at the cabinet base was calculated under the assumption that the angle-shaped member was fixed along the inside edge and guided along an outside edge. Were this member more conservatively assumed to be fixed on the inside edge and free on the outside edge, the frequency calculation would yield 14.7 Hz rather than 19.7 Hz. A difference of this magnitude could result in a significant effect on cabinet response. This possible nonconservatism in the applicant's calculations is, however, countered to some degree by his assumption that the angle-shaped member receives no support from the floor. To clear the concern about flexibility at the base of this panel, the applicant was requested to justify the assumption that the outside edge of the angle-shaped member is guided. A justification was provided in the May 17, 1984, submittal. However, the question discussed in the submittal of which value is correct (TVA has stated the conservative calculation would yield 16.7 Hz rather than 14.7 Hz) is not particularly pertinent. If the applicant's value of 16.7 Hz is used, the shift in frequency from 21.1 Hz represents a decrease in frequency of 21%. This represents the bounding decrease, with smaller decreases being based on engineering judgment. If this decrease percentage is applied to the minimum frequency for the full control board (14.3 Hz), a value of 11.32 Hz is obtained. Further, if a broadened RRS is compared with the response spectra (RS) of the analysis time history, the broadened RRS is not enveloped by the time history RS under 12 Hz. Therefore, the qualification of the control board cannot be ensured without the exercise of engineering judgment. This being the case, an in situ test of the board is requested to gain a measure of confidence for the minimum horizontal natural frequency of the board. Note that such a test could also close Generic Concern No. 3. This issue remains open.

(b) The applicant was requested to provide justification for not peak broadening the RRS. The applicant's submittal maintained that peak broadening of the RRS for this piece of equipment was unnecessary. The natural frequencies of the equipment are sufficiently removed from response spectrum peaks so that equipment response is not affected by the shape of the spectrum near the peaks. The staff agrees with the applicant's justification and considers this issue closed.

(6) Electrical Penetrations

The applicant was requested to supply a comparison of site-specific RRS and the TRS that shows adequate envelopment of the RRS by the TRS. The applicant responded that the equipment was qualified to the generic RRS. The generic RRS envelopes the Watts Bar floor-level RRS. Thus, qualification to the generic criteria ensures qualification to site-specific criteria. This issue is closed.

(7) 125-V dc Vital Batteries

(a) The applicant was requested to supply a description of surveillance programs that will be instituted to maintain seismic capability of the vital batteries throughout plant life. As part of the applicant's December 1, 1982, submittal, an annual field inspection procedure for maintaining the batteries was provided. This procedure details a maintenance program that, if followed, should ensure continued seismic capability of the batteries. Therefore, this issue is closed.

(b) The applicant was requested to provide verification that batteries will have spacers installed, as was done during qualification tests. By letter dated June 19, 1984, the applicant stated that spacers will be installed by December 1, 1984. Spacers are in place in the emergency diesel generator battery assembly. As with Item (1)(b) in this section, this item will be closed on confirmation of the field modification. Because this confirmation has not yet been submitted, this issue is open.

(c) The applicant was requested to explain whether the vendor had been given release from the requirement that the battery cells have positive anchorage. The December 1, 1982, submittal stated that the Watts Bar battery specification does not require holddown hardware for anchorage. Side and end rails located above the center of gravity of the batteries, as used at Watts Bar, have been demonstrated by qualification tests to sufficiently restrain the batteries. Therefore, this issue is closed.

(8) Diesel Generator Control and Protection Relay Panel

(a) The applicant was requested to provide a comparison of an RRS and the TRS and justification for the adequacy of single-frequency input motion. In the December 1, 1982, submittal, the applicant provided TRS curves for the single-frequency tests that envelope the RRS with a large margin at the natural frequencies of the equipment. These test results along with results from tests conducted at RRS peak frequencies indicate that the anticipated equipment response to SSE conditions was well exceeded during tests, even if multimodal excitation effects are considered. For this specific item, this issue is closed.

(b) The applicant was requested to address the consequence of observed relay chatter. The applicant responded that the relay exhibiting chatter is used only during the testing phases of the diesel generator system and will not prevent proper operation of the system during a seismic event. Therefore, this issue is closed.

(c) The applicant was requested to demonstrate that all relays (including differential relays) will be functional in both the open and closed positions. As part of the December 1, 1982, submittal, the applicant furnished a copy of the test procedure that was followed during seismic testing of the relays. The procedure indicates that the lockout and auxiliary relays were tested in both the energized and deenergized modes. The protective relays that were tested in only the deenergized mode because they do not perform any safety-related function in the energized mode were reviewed carefully to determine that no safety concerns arose as a result of a change of state from energized to deenergized condition while the diesel generator is providing emergency power. By letter dated May 17, 1984, the applicant indicated that these relays could prevent safe operation of the diesel generators only if other, fully qualified relays tripped improperly. This issue is closed.

(9) <u>Metal Clad Switchgear</u>

The applicant was requested to describe the mounting method used during testing and to show that the field mounting is as adequate as the test mounting. The December 1, 1982, submittal stated that the test specimen was attached to the test table with 10-1/2-in. bolts and that the cabinet was field mounted in the same manner. Thus, the testing configuration was representative. For this specific piece of equipment, this issue is closed.

(10) <u>Main Steam Isolation Valve</u>

(a) The applicant was requested to provide a comparison of nozzle loads with allowable values. In the December 1, 1982, submittal, the applicant supplied allowable values for bending moment and axial loads at the nozzle that well exceed the values obtained from the piping analysis. The applicant was requested to furnish similar information on torsion and shear loads. In the June 10, 1983, submittal, the applicant supplied a value for the torsional moment that was applied during qualification tests on this valve. By letter dated May 17, 1984, the applicant quoted values for the torsional moment and shear force obtained from the piping analysis to ensure that the shear stresses were enveloped by the test. The staff has reviewed these values, finds them acceptable, and considers this issue closed.

(b) The applicant was requested to supply the maximum acceleration levels at the valve location as obtained from the piping analysis results. The applicant's submittal stated that the seismic response acceleration calculated for the main steam isolation valves (MSIVs) has a minimum resultant magnitude of 2.68g, which lies well within the test acceleration levels. This concern is closed.

(c) The applicant was requested to describe tests performed on the 32-in. MSIV and explain how these ensure operability of the valve under Watts Bar seismic conditions. In the December 1, 1982, submittal, the applicant indicated that the valve was subjected to internal pressure, the nozzle loads described in Item (10)(a) above, and static seismic loads in each direction. The seismic loads were applied at the operator center of gravity. The valve was previously shown to behave rigidly, and the applied loads exceed loads incurred in the Watts Bar installation. Provided that the torsion and shear loads mentioned in Item (10)(a) above are properly addressed, these tests adequately demonstrate qualification of the valve. This issue is closed.

(11) Essential Raw Cooling Water Pump

The applicant was requested to perform an evaluation of the effects of impact loads resulting from nonlinear supports along the pipe column. The applicant responded by performing a conservative simplified calculation that showed the "downhole" portion of the pump and its gapped supports would impact the impeller bearings, but the impact load was well within allowable loading on the impeller bearings. Operability of the pump in a seismic event is, therefore, not impaired by impact loads. Therefore, this concern is closed.

(12) Diesel Combustion Air Intake Filter

Because the qualification report for this item was not available during the plant audit, the applicant was requested to submit the qualification documents for review. The applicant's December 1, 1982, submittal contained qualification documents that indicate that the filter was qualified by stress analysis for seismic and deadweight loads. Calculations showed that the filter, together with six vertical stiffeners, responds rigidly to seismic loads. Seismic loads derived from the Watts Bar response spectra were applied statically to the filter, and resulting stresses were found to lie well within acceptable levels. The documentation verified that this item was adequately qualified. The vertical stiffeners were, however, considered to be 4 x 4 x 3/8 in. in size in one document and 2 x 3 x 3/8 in. in another. The applicant was requested to resolve this inconsistency. In the June 10, 1983, submittal, the applicant indicated that the angle stiffeners were verified by field inspection to be 2 x 3 x 3/8 in. in size. He also identified the material (American Society for Testing and Materials (ASTM) A36 steel) from which they are constructed. The staff finds this acceptable and considers this issue closed.

(13) Barksdale Pressure Switch/Square D Relay

(a) The applicant was requested to supply details of biaxial tests and items involved therein. The December 1, 1982, submittal indicated that biaxial sine beat tests at SSE input levels were performed on the Barksdale switch at all resonance frequencies. After initial biaxial tests, the specimen was rotated 90° horizontally and retested.

No biaxial tests, however, were performed on the square D relays. This was justified on the basis that electrical relays inherently reflect a single axis failure mode, since contact chatter is due to excitation along the axis parallel to that of contact motion. In addition, a wide safety margin in seismic input to the panel during the relay tests alleviates concern that directional coupling in panel response could increase loading on the relays to unacceptable levels. Therefore, the concern on single and biaxial testing is closed.

(b) The applicant was requested to supply a justification for single-frequency tests. The December 1, 1982, submittal presented rationale for this type of testing. The square D relays and the Barksdale pressure switch were qualified in separate test programs. The relays were contained in a control panel that was subjected to high-level sine beat tests over a range of frequencies and at equipment natural frequencies. Single-frequency tests were justified by noting that one frequency predominates in the panel RRS. Because equipment natural frequencies are well removed from this critical frequency, the potential for multimodal excitation is minimal. Additionally, the TRS accelerations far exceed RRS accelerations at equipment natural frequencies. This justification for single-frequency tests on the relays is reasonable.

The pressure switch was tested in a similar fashion. Again, single-frequency tests were justified by noting that the RRS exhibits one predominant frequency. In this case, however, at least two of the equipment natural frequencies lie at or near the RRS peak. This situation creates the potential for multimodal excitation and the possibility that the single-frequency TRS do not adequately envelope a broadened RRS. The applicant was requested to further evaluate this issue. The June 10, 1983, submittal did not supply any new information relative to this concern about multimodal excitation for the pressure switch. The May 17, 1984, submittal would be acceptable if the applicant had demonstrated, using the unbroadened spectra, that multimodal response is not possible. The fact that the response spectra had been broadened does not, by itself, ensure that the unbroadened spectra will not generate multimodal response. This issue remains open.

3.10.3 Summary

On the basis of the Seismic Qualification Review Team site audit and the submittals from the applicant, the staff concludes that an appropriate seismic and dynamic qualification program has been defined and substantially implemented, with the exception of the above open issues. The open issues for both generic and equipment-specific items must be resolved before fuel loading.

Completion of the applicant's seismic and dynamic qualification program for safety-related equipment is required to satisfy applicable portions of GDC 2, 4, 14, and 30 of Appendix A to 10 CFR 50 and Paragraphs XI of Appendix B to 10 CFR 50 and VI(a)(1) and (2) of Appendix A to 10 CFR 100 as they relate to qualification of equipment.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.5 Loose Parts Monitoring System

In the SER, the staff stated that before an operating license is issued, the applicant should provide a commitment to supply a report on the loose parts monitoring system (LPMS) that contains:

(1) an evaluation of the LPMS for conformance to RG 1.133, Revision 1

(2) a description of the operator training program

(3) a description of the system hardware operation and implementation of the loose parts detection program after startup testing

In a letter dated February 25, 1982, the applicant provided a brief description of the operator training program and the LPMS hardware and an evaluation of conformance to RG 1.133. By letter dated November 10, 1982, the applicant submitted additional information. The Watts Bar FSAR was also revised in Amendment 49 to provide a more detailed description of the LPMS. The applicant's evaluation has shown that the LPMS meets Positions 1 through 4 in Section C of RG 1.133 regarding the LPMS characteristics, alert level establishment, data acquisition, and safety analysis report content. With regard to Position 5, the limiting conditions for operation and surveillance requirement with respect to the LPMS have been incorporated into Technical Specifications 3.3.3.11 and 4.3.3.11. Position 6, "Notification of a Loose Part," has not been addressed by the applicant. However, 10 CFR 50.73 requires that the applicant shall submit a licensee event report (LER) within 30 days after the discovery of "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being: (a) in an unanalyzed condition that significantly compromised plant safety and (b) in a condition that was outside the design basis of the plant." Because the presence of loose parts in the primary system belongs to this category, an LER would be required according to 10 CFR 50.73. Because 10 CFR 50.73 became effective January 1, 1984, and replaces all existing requirements for applicants to report reportable occurrences as defined in individual plant Technical Specifications, Position 6 is covered by this regulation.

On the basis of its review of the applicant's submittals, the staff has found that the LPMS is acceptable. However, to ensure compliance with RG 1.133, the license will be conditioned to require the applicant to submit to the NRC within 90 days following completion of the startup test program the results of the preoperational test and the alert level setting of the LPMS.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

Protection Against Damage From External Pressure

The containment vessel is designed for an external pressure of 2.0 psig. Inadvertent operation of the spray system and/or air return fan system during normal plant operation would cause a reduction in the containment pressure. The applicant has stated that such transients would not cause containment external pressure to exceed the design value of 2.0 psig. The staff requested the applicant to furnish the analysis that supports this statement. A simplified analysis by the staff showed that the containment external design pressure value would not be exceeded by the transients mentioned above. As a result, the staff considered this matter to be a confirmatory item.

By letter dated June 4, 1982, the applicant provided the requested analysis, with which the staff concurs. The analysis shows that inadvertent actuation of one or both trains of the containment spray system during normal plant operation would reduce containment pressure to -1.3 psig, which is within the design capacity of -2.0 psig.

The applicant has further calculated that inadvertent operation of one air return fan during normal plant operation results in a containment internal pressure of -2.0 psig after about 790 sec (13 min). Because fan operation is alarmed in the control room, operators would be alerted to the need to take corrective action.

The staff considers 13 min to be sufficient time for the plant operators to determine that inadvertent fan actuation has occurred and to terminate fan operation.

Inadvertent operation of both air return fans is not considered likely because of the redundancy in the electrical power supplies. This could only occur as a result of a spurious containment high-high pressure signal or phase B isolation signal. Nevertheless, a 10-min delay in fan operation is designed into the signal, after which it would take another 6.5 min for containment pressure to drop to -2.0 psig. Again, operators would be alerted and would have sufficient time to terminate the transient.

The staff, therefore, concludes that the Watts Bar design provides adequate protection against damage from external pressure transients and considers Confirmatory Item (17) resolved.

6.2.4 Containment Isolation System

The SER stated that the Watts Bar plant conforms with all the provisions of Item II.E.4.2, "Containment Isolation Dependability," of NUREG-0737, "Clarification of TMI Action Plan Requirements," except for Section (6) concerning containment purging and venting during normal operation. It was further stated that the applicant had not fully addressed the provisions of Branch Technical Position (BTP) CSB 6-4, "Containment Purging During Normal Plant Operations" (NUREG-0800). The applicant has since provided the necessary information, addressing point-by-point the provisions of BTP CSB 6-4. The applicant has shown conformance with the provisions of BTP CSB 6-4, contingent on a favorable finding by the staff as to the operability of the containment purge/vent isolation valves, that is, that they are qualified to close against LOCA-induced pressure transients inside containment if they are open at the onset of an accident. Therefore, subject to resolution of the contingency stated above, the staff finds that the purge/vent system design conforms with the provisions of Item II.E.4.2 of NUREG-0737 and BTP CSB 6-4 and, therefore, considers Open Item (8) resolved. Operability of the containment purge/vent isolation valves is currently under review and will be addressed in a future supplement to the SER under License Condition (8).

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Features Actuation System

7.3.5 IE Bulletin 80-06

Office of Inspection and Enforcement (IE) Bulletin 80-06 calls for the review of all safety equipment to determine which, if any, safety functions might be unavailable after reset and what changes could be implemented to correct any problems. In the SER, the staff concluded that the applicant's proposed design modifications are acceptable subject to review of the electrical schematics, which were not available at that time. In FSAR Amendment 48, the applicant submitted the electrical schematics that implement the following changes:

(1) For feedwater isolation valves (FCV-3-33, FCV-3-47, FCV-3-87, and FCV-3-100), feedwater check valve bypass valves (FCV-3-185, FCV-3-186, FCV-3-187, and FCV-3-188), and upper tap main feedwater isolation valves (FCV-3-236, FCV-3-239, FCV-3-242, and FCV-3-245), a new reset switch and a relay have been added for each steam generator loop. When the engineered safety feature (ESF) signal is reset, the individual valve will not change state until both the loop and the ESF train reset switches have been reset.

(2) For steam generator blowdown isolation valves (FCV-43-54D, FCV-43-56D, FCV-43-59D, FCV-43-63D, FCV-43-55, FCV-43-58, FCV-43-61, and FCV-43-64), the ESF signal is sealed in by means of a valve-mounted limit switch. The individual valve will not change state until a hand switch in the sample room is used to reopen the individual valve.

(3) For residual heat removal heat exchanger outlet flow control valves (FCV-74-16 and FCV-74-28), the ESF signal is sealed in by the limit switch. A new reset switch has been added at the control room control board. When the ESF signal is reset, the individual valve will not change state until the individual reset switch has been reset.

The staff has reviewed the electrical schematics and finds the design modification acceptable.

The applicant has also identified all the safety-related equipment that does not remain in its emergency mode after ESF reset. The applicant evaluated this equipment and determined that it does not impact the safety of the plant or the ability to achieve and maintain safe shutdown. The staff has reviewed the applicant's response and finds that the applicant's justification is acceptable. Therefore, the staff considers Confirmatory Item (24) resolved.

8 ELECTRICAL POWER SYSTEMS

8.2 Offsite Electric Power System

As noted in Section 8.2.2.2 of the SER, by letters dated October 9, 1981, and January 7, 1982, the applicant proposed a system design change that adds two service transformers with related distribution system components to the offsite power system. In Amendments 48 and 49 to the FSAR, the applicant presented additional description and analysis relating to the proposed new design for the offsite power system.

The following items address concerns raised during the staff review and their resolution.

(1) Section 8.2.2 of the FSAR states that, with degraded voltage conditions on the 6.9-kV shutdown board (Class 1E bus), offsite power is automatically transferred from the normal to the first alternate to the second alternate to the onsite diesel generator source. The same section of the FSAR also states in contradiction that, on degraded voltage, offsite power is only transferred from the normal to the diesel generator source. This contradiction was discussed with the applicant during telephone conference calls on October 27 and November 1, 1983. On the basis of these telephone conference calls, the staff concluded that the Watts Bar design provides immediate automatic transfer to the preferred offsite circuits on degraded grid voltage. However, the Watts Bar design shown in FSAR Figure 8.3-5a is not consistent with the applicant's verbal description of the design provided during the above referenced telephone conference calls. This item will continue to be pursued with the applicant, and the results of the staff review will be reported in a future supplement.

(2) The routing of the two offsite circuits from the service transformers to the Class 1E distribution system is through separate conduit and cable trays of a single commonly supported raceway system. With a single commonly supported raceway system being used for both offsite circuits, the circuits are not located, in accordance with GDC 17, so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident conditions. The Watts Bar design, however, has other offsite circuits that are physically separated and that can be made electrically independent and available to supply power on a delayed basis. With the delayed availability of these other circuits, the staff concludes that this part of the Watts Bar design meets GDC 17 and is acceptable.

(3) Section 8.2.1.6 of the FSAR indicated only that control power associated with the two offsite power circuits is provided by a 125-V dc supply for the station service transformer and switchgears C and D. Information on the physical and electrical separation of control power and independence between control power associated with onsite and offsite sources had not been described or analyzed in the FSAR in accordance with the requirements of GDC 17. By Amendment 49 to the FSAR, the applicant provided the required description and analysis, as discussed below:

The 125-V dc control power for the offsite power circuits is provided from the onsite Class 1E power system. The control power cables are treated as associated circuits and are routed in separate raceways. The staff finds that this routing meets the independence requirements of GDC 17 for offsite circuits and is acceptable.

In regard to supplying control power from the onsite Class 1E power system, it was the staff's concern that design provisions were not provided, in accordance with the requirements of GDC 17, to minimize the probability of losing electrical power from the offsite system as a result of or concurrently with loss of the onsite power system supplies. By Amendment 49 to the FSAR, the applicant indicated that loss of control power will cause failure of the automatic transfer system used to transfer offsite circuits from the normal to the preferred offsite power sources and may necessitate reestablishment of offsite circuits by manually closing breakers. Although it may be preferable for control power to be from a supply independent of the Class 1E system, the staff concludes that the Watts Bar design, which requires loss of two independent Class 1E dc power supplies in order to cause loss of both offsite circuits, minimizes the probability of losing offsite power in accordance with the requirements of GDC 17 and is, therefore, acceptable.

8.3 Onsite Power Systems

8.3.2 Onsite DC System Compliance With GDC 17

8.3.2.2 DC System Monitoring and Annunciation

In the SER, the staff required, as a condition to the license, that the same indications and alarms that are provided for the 125-V station battery system be provided for the 125-V diesel generator battery system or, as a minimum, the following abnormal conditions shall be alarmed in the control room:

(1) battery circuit open
(2) battery charger circuit open
(3) dc bus ground fault
(4) dc bus undervoltage
(5) dc overvoltage
(6) battery charger failure
(7) battery discharge

In addition, the staff required that the following parameters also be monitored:

(1) dc bus voltage
(2) battery circuit input current
(3) battery circuit output current
(4) battery charger output current

By letter dated January 15, 1984, the applicant indicated that the Watts Bar design fully satisfies the above requirements. However, on the basis of information presented in Amendment 48 to the FSAR, the staff finds that the following items have not been included in the Watts Bar design.

(1) Battery circuit input current is not monitored.
(2) DC bus undervoltage is not alarmed in the control room.

Justification for not including these items will be pursued with the applicant, and the results of the staff review will be reported in a future supplement.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDC 2 and 4

8.3.3.1.1 Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

In the SER, the staff indicated that the design for the automatic deenergization of circuit loads as a result of a loss-of-coolant accident would be verified as part of the staff's site visit/drawing review. This item will be discussed with the applicant, and the results of the staff review will be reported in a future supplement.

8.3.3.2 Compliance With GDC 5

8.3.3.2.2 Sharing of AC Distribution Systems and Standby Power Supplies Between Units 1 and 2

In the SER, the staff indicated that sharing of onsite ac and dc systems had not been adequately described or analyzed in Section 8.3 of the FSAR. By letter dated January 7, 1982, the applicant provided the subject description and analyses. On the basis of this letter, the staff concluded in the SER that the design meets the guidelines of RG 1.81 and was found acceptable pending revision of the FSAR that reflects requirements of the shared safety systems. By Amendment 48 to the FSAR, the applicant partially documented the description and analysis presented in the January 7, 1982, letter. By letter dated January 17, 1984, the applicant submitted preliminary copies of an amendment to the FSAR in order to provide the remaining description and analysis.

The staff has determined that information presented in Amendment 48 and the January 17, 1984, letter is consistent with information presented in the applicant's letter of January 7, 1982. This item is, therefore, acceptable pending confirmation that the information in the January 17, 1984, letter is incorporated into the FSAR.

8.3.3.2.4 Possible Sharing of DC Control Power to AC Switchgear

In the SER, the staff required, as a condition in the license, that all possible interconnections between redundant divisions through normal and alternate power sources to various loads be identified in the FSAR regardless of the source of power and that these interconnections meet the following positions:

(1) The circuit breaker for the alternate feed shall be kept open and be alarmed in the control room when closed.

(2) The manual transfer switch shall be alarmed in the control room when it is in the alternate supply position.

By letter dated January 17, 1984, the applicant documented that all interconnections are identified in Tables 8.3-9 and 8.3-10 of the FSAR and meet the staff position, except as noted and accepted in Section 8.3.3.2.4 of the SER. This item is acceptable, and, therefore, License Condition (14) is no longer required.

8.3.3.3 Physical Independence (Compliance With GDC 17)

(2) Associated Circuits

The staff's evaluation of this item is included in Section 8.3.3.3(3) of this report.

(3) Separation Criteria Between Class 1E and Non-Class 1E Circuits

In the SER, the staff required, as a condition to the license, that protective devices (used to isolate non-Class 1E from Class 1E circuits) be of a high quality commensurate with their importance to safety and be periodically tested. By Amendment 48 to the FSAR, the applicant provided the results of a reliability study to demonstrate that a single circuit breaker that is periodically tested has reliability equivalent to each of the following protective device configurations:

(a) two series connected circuits breakers that are not tested
(b) a circuit breaker in series with a fuse that is not tested
(c) a single fuse that is not tested

On the basis of the results of this reliability study, the staff concludes that each of the above configurations provides equivalent protection and is acceptable. The applicant by letter dated January 17, 1984, has stated:

(a) The protective devices of the 160 associated circuits either have or are being modified to have one of the above protective device configurations.

(b) The protective devices of non-Class 1E circuits that are routed closer to Class 1E circuits than allowed by RG 1.75 have been identified and will be periodically tested, except those that have one of the above protective device configurations.

The staff concludes that this item is acceptable, and, therefore, License Conditions (15) and (16) are no longer required.

8.3.3.4 Compliance With NUREG-0737 Items

<u>Emergency Power for Pressurizer Equipment (II.G.1)</u>

To meet the guidelines of both TMI Item II.G.1 and Branch Technical Position RSB 5-2, the staff in the SER required as a condition to the license that the power supplies for the power-operated relief valves (PORVs) and block valves be from the same power train but from different buses. By letters dated June 9, 1983, and January 17, 1984, the applicant stated that the Watts Bar design does not comply with the above requirements and indicated his disagreement with the requirement because of safety reasons. The basis presented in the applicant's June 9, 1983, letter is inconsistent with the current design being implemented at the Watts Bar facility. By letter dated October 24, 1984, the staff informed TVA of the basis of its position. This item will be pursued with the applicant, and the results of the staff review will be reported in a future supplement.

8.3.3.6 Compliance With GDC 50

In the SER, the staff required a reevaluation of the penetrations' capability to withstand, without seal failure, the total range of available time-current characteristics assuming a single failure of any overcurrent protective device. By Amendment 48 to the FSAR, the applicant documented the results of the reevaluation. On the basis of these results, discussions with the applicant, and information submitted by letter dated January 17, 1984, the staff concludes that all circuits (Class 1E, non-Class 1E, normally energized, and normally deenergized) that pass through containment electric penetrations have been designed with the required capability, contain the required primary and backup protective devices (except low-energy instrument circuits), meet the guidelines of Position 1 of RG 1.63, and are, therefore, acceptable pending confirmation that information in the January 17, 1984, letter is incorporated into the FSAR.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.4 Fuel-Handling System

In the SER, the staff stated that the applicant would submit the results of the review against the guidelines of NUREG-0612, "Control of Heavy Load at Nuclear Power Plants," by August 1982. The staff further stated that the applicant's commitment to implement the interim actions of NUREG-0612 before issuance of an operating license provides reasonable assurance of safe handling of heavy loads until NUREG-0612 can be fully implemented. By letters dated February 6, and March 20, 1984, the applicant submitted the required information regarding Sections 5.1.1 and 5.3 (Phase I) and Sections 5.1.2 through 5.1.6 (Phase II) of NUREG-0612. This information is currently under review and will be the subject of a future safety evaluation. However, to ensure completion of Phase I in a timely manner, the staff will require that the following license condition be placed in the license:

> The applicant will meet the guidelines of Sections 5.1.1 and 5.3 of NUREG-0612 (Phase I) before the first refueling outage.

9.3 Process Auxiliaries

9.3.2 Process Sampling System

Postaccident Sampling Capability (II.B.3)

In the SER, the staff determined that the postaccident sampling system (PASS) met 5 of the 11 criteria for Item II.B.3 in NUREG-0737. The following six criteria were unresolved.

(1) Criterion (2) - Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

(2) Criterion (5) - Verify that chloride analysis can be completed within 4 days following an accident which requires postaccident sampling.

(3) Criterion (8) - Demonstrate the capability of analyzing the grab samples and verify that equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

(4) Criterion (9)(a) - Verify that the sensitivity of onsite liquid sample analysis capability is such as to permit measurement of nuclide concentration of the range from approximately 1 μCi/g to 10 Ci/g.

and

Criterion (9)(b) - Verify that provisions are available to restrict background radiation levels such that the sample analyses will provide results with a range of accuracy within a factor of 2.

(5) Criterion (10) - Describe the procedures for onsite radiological and chemical analyses and provide the accuracy, range, and sensitivity of these analyses in an accident chemistry and radiation environment (that is, the presence of large amounts of fission products and a high radiation field in the samples).

Provide information on testing frequency and type of testing to ensure long-term operability of the postaccident sampling system and on operator training requirements for postaccident sampling.

(6) Criterion (11)(a) - Verify that the residues of sample collection will be returned to containment or to a closed system.

and

Criterion (11)(b) - Verify that the ventilation exhaust from the sample station will be filtered with charcoal adsorbers and high-efficiency particulate air filters.

By letters dated October 29, 1981, September 20, 1983, December 19, 1983, and July 13, 1984, the applicant provided additional information to address these items, as discussed below.

Criterion (2)

NUREG-0737 requires that the applicant establish an onsite radiological and chemical analysis capability to provide, within a 3-hour time frame, quantification of the following:

(1) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes)

(2) hydrogen levels in the containment atmosphere

(3) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids

or have inline monitoring capabilities to perform all or part of the above analyses.

The applicant has the capability to obtain and analyze samples (within 3 hours of the time a decision is made to sample) from the reactor coolant system, containment sump, and containment atmosphere under accident conditions, and to obtain grab samples for offsite analyses. The applicant also has onsite radiological and chemical analysis capabilities to quantify hydrogen levels in containment atmosphere; radioactive isotopes (noble gases, iodine and cesium isotopes, and nonvolatile isotopes); and dissolved gases, chloride, and boron concentrations in liquid samples.

In a letter dated December 19, 1983, the applicant provided an interim procedure for estimating the degree of reactor core damage from measured and predicted postaccident radionuclide concentrations from failed fuels. The applicant indicated that, as a member of the Westinghouse Owners Group, the generic methodology for estimating core damage that is being prepared by the Westinghouse Owners Group will be adopted for the Watts Bar facility.

The staff has determined that these provisions meet Criterion (2); therefore, the procedure for estimating core damage is acceptable on an interim basis. The license will require the applicant to provide a final plant-specific procedure to estimate the extent of core damage on the basis of radionuclide concentrations and taking into consideration other physical parameters such as core temperature data, sample location, and containment radiation levels and hydrogen concentrations before restart following the first refueling outage.

Criterion (5)

NUREG-0737 states that the applicant shall provide for a chloride analysis within 24 hours of the taking of a sample (1) if the plant's coolant water is seawater or brackish water and (2) if there is only a single barrier between primary containment systems and the cooling water. In all other cases, the applicant shall provide that the analysis be completed within 4 days. The chloride analysis does not have to be done on site.

The applicant has stated that the chloride analysis will be performed within 4 days after an accident that requires postaccident sampling. This analysis will be done in line, with undiluted samples, using an ion chromatograph. The staff finds these provisions meet Criterion (5) and are, therefore, acceptable.

Criterion (8)

NUREG-0737 requires that the applicant shall provide backup sampling through grab samples if in-line monitoring is used for any sampling and analytical capability as specified, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per week until the accident condition no longer exists.

The applicant has stated that a diluted and undiluted reactor coolant grab sample and undiluted containment atmosphere grab sample will be obtained for analyses of boron, dissolved hydrogen, pH, chloride, and radioisotopes in the reactor coolant and hydrogen, oxygen, and radioisotopes in the containment atmosphere. The staff finds that these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion (9)

NUREG-0737 requires that the applicant's radiological and chemical sample analysis capability shall include provisions to:

(1) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source term given in RGs 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure

should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

(2) Restrict background levels of radiation in the radiological and chemical analysis facility from sources so that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design that will control the presence of airborne radioactivity.

The applicant has stated that the radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source terms given in RGs 1.4, Revision 2, and 1.7. These analyses will be accurate within a factor of 2. The staff finds that these provisions meet Criterion (9) and are, therefore, acceptable.

Criterion (10)

NUREG-0737 requires that accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.

The applicant has stated that the accuracy, range, and sensitivity of the Watts Bar PASS instruments and analytical procedures are consistent with the recommendations of RG 1.97, Revision 3, and the clarifications of NUREG-0737, Item II.B.3. Therefore, they are adequate for describing the radiological and chemical status of the reactor coolant. The analytical methods and instrumentation were selected for their ability to operate in the postaccident sampling environment. The applicant proposes that equipment used in postaccident sampling and analyses be calibrated or tested annually and retraining of operators for postaccident sampling be performed as needed. Every 6 months, one-half of the chemistry technicians will both operate the PASS and actually take samples of the fluids in pertinent systems. At the same time, identical samples will be taken in the hot sample room. This will verify that the PASS is functioning properly. By using this timetable, the operator will be retrained on a yearly basis, and the PASS will be tested every 6 months.

The staff has concluded that these provisions meet Criterion (10) and are, therefore, acceptable.

Criterion (11)

In the design of the postaccident sampling and analysis capability, NUREG-0737 states that consideration should be given to the following items:

(1) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing the sample loss or distortion, for preventing blockage of sampling lines by loose material in the reactor coolant system or containment, for appropriate disposal of the samples, and for flow restrictions to

limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(2) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

The applicant has addressed provisions for purging to ensure samples are representative, size of sample line to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from the PASS filtered through charcoal adsorbers and HEPA filters. Excess samples are flushed to a radwaste system. Heat tracing of the containment atmosphere sample line is provided to aid in obtaining representative samples. The staff has determined that these provisions meet Criterion (11) of Item II.B.3 of NUREG-0737, and are, therefore, acceptable.

Conclusion

The staff has concluded that the postaccident sampling system now meets all of the 11 criteria of Item II.B.3 of NUREG-0737, and, therefore, the postaccident sampling system is acceptable. The proposed procedure for estimating the degree of reactor core damage is acceptable on an interim basis. Before restart following the first refueling outage, the applicant will be required to provide a final procedure for estimating the degree of core damage.

9.5 Other Auxiliary Systems

9.5.7 Emergency Diesel Engine Lubricating Oil System

The SER stated:

> The auxiliary keep-warm lubrication system for the diesel engine is composed of a continuously operating oil circulating ac motor-driven pump which prelubricates the turbocharger bearings only and circulates oil to the lube oil cooler for preheating. The other wearing parts of the engine do not receive any lubrication until after the engine starts, and the engine driven lube oil pumps reach full speed. This is not acceptable....

In letters dated October 9, 1981, and December 14, 1982, the applicant proposed using the manufacturer's modification to alleviate the staff's concern regarding dry diesel engine starting. The staff reviewed the manufacturer's modification, EMD MI-9644, and determined that the modification would provide continuous prelubrication to the lower portions of the engine, but not the upper portions, that is, rocker arms, camshaft, and associated wearing parts, before an emergency diesel engine start. The staff informed the applicant that adequate justification for the nonautomatic prelubrication of the upper rocker arm assembly wearing parts be provided or prelubrication be provided periodically.

By letter dated January 12, 1984, the applicant submitted the manufacturer's justification for not prelubricating the upper rocker arm assembly of the emergency diesel engines at the Watts Bar facility. The staff in its review of this information found:

(1) The bearings at the camshaft and rocker arms are lightly loaded during startup, are not subjected to marginal conditions that require prelubrication, and will not experience damage or distress even if they are not lubricated during startup.

(2) The unfilled portion (volume) of the camshaft and upper rocker arm assembly lube oil system, after implementation of EMD MI-9644, is small (1/2 to 1 gal depending on engine size) compared with the total volume of the engine lube system. Because of the small volume involved, this portion of the system is rapidly filled during the first few revolutions of the engine by the engine-driven lube oil pump. The pump delivers 1/10 to 1/4 gal per engine revolution, depending on the engine size.

(3) Lube oil system operating pressure is rapidly established following startup because most of the oil system is fully charged.

On the basis of this information, the staff finds the implementation of EMD MI-9644 at the Watts Bar facility will alleviate the concern regarding dry engine starting and is an acceptable alternative to continuous or intermittent prelubrication of the upper bearing surfaces on the diesel generators. The modification will be installed before fuel loading.

On the basis of its review, the staff concludes that the emergency diesel engine lubricating oil system meets the requirements of GDC 2, 4, 5, and 17; meets the guidance of the cited regulatory guides and SRP Section 9.5.7; meets the recommendations of NUREG/CR-0660 and industry codes and standards; can perform its safety function; and is, therefore, acceptable.

13 CONDUCT OF OPERATIONS

13.5 Plant Procedures

13.5.3 NUREG-0737 Items

Report on Outages of Emergency Core Cooling System (II.K.3.17)

In the SER, the staff stated that the applicant has committed to develop and implement a plan for collecting emergency core cooling system (ECCS) outage information. Since publication of the SER, the staff has received sufficient information from operating facilities to perform its review of the TMI Action Plan item. In a letter dated October 28, 1983, the applicant revised his commitment to comply with this item. The staff concurs with the applicant's revised position that active participation in the nuclear power reliability data system and compliance with the requirements of 10 CFR 50.73 satisfy the current requirements of Item II.K.3.17 of NUREG-0737. Therefore, the staff will not impose a license condition on the Watts Bar facility in this matter and considers License Condition (30) closed.

14 INITIAL TEST PROGRAM

In the SER, the staff identified open and confirmatory items resulting from its review of FSAR Chapter 14 (as amended through Amendment 46). The applicant has since provided additional information and FSAR amendments addressing the staff's concerns. The following is a discussion of these items.

(1) Test Procedures

The staff reviews selected test procedures as part of the preoperational phase inspection plan. The SER stated that the applicant plans to provide the staff with copies of test procedures 1 month before testing. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Appendix B, states that preoperational test procedures should be available 60 days before testing and startup test procedures should be available 60 days before fuel loading. In a letter dated December 22, 1983, the applicant indicated that he had discussed this issue with the NRC staff who had indicated that 1 month was sufficient. This issue is considered resolved.

(2) Conformance to RG 1.20, Revision 2; RG 1.52, Revision 2; and RG 1.79, Revision 1

The SER stated that FSAR Table 14.2-3 does not reflect conformance of preoperational tests with these regulatory guides. In FSAR Amendment 49 the table was appropriately revised. This item, therefore, is resolved.

(3) Preoperational Tests

The SER stated that the applicant's FSAR does not include preoperational tests of the following systems:

(a) condensate system

(b) condenser circulating water system

(c) hotwell level control system

(d) condensate storage tank auxiliaries

(e) 48-V dc system

(f) failed fuel detection system

(g) chemical addition system for the secondary plant

(h) turbine gland sealing system and gland seal water system

(i) standby lighting system

(j) ventilation systems for the intake pumping station

(k) emergency core cooling (ECC) systems (leakage tests) and leak detection and pumping systems provided to control leakage from ECC systems

(l) turbine building area ventilation system

(m) raw cooling water system

In a letter dated April 21, 1982, the applicant indicated that the above systems would be tested as part of the Non-Critical Systems (NCS) Test Program. NCS tests have a lower level of quality assurance (QA) overview than the preoperational tests covered by FSAR Chapter 14. NCS test procedures are not subject to review by the Plant Operations Review Committee, and the NCS test program is not subject to QA audit or review by the Nuclear Safety Review Board. This graded approach to testing is consistent with GDC 1. The staff has reviewed the NCS Test Program described in Procedure No. 1105.01 of the applicant's Test Staff Program Manual and has concluded that NCS testing is acceptable for Items (a), (b), (c), (g), (h), (l), and (m) above. Test descriptions for NCS are not provided in the FSAR; however, these test procedures will be subject to review by the staff in accordance with the preoperational phase inspection plan. The staff requires that the status of NCS tests be tracked and considered in determining readiness for licensing in the same way as the preoperational tests described in the FSAR.

The SER stated that the systems in Items (d), (e), (f), (i), (j), and (k) should be preoperationally tested and that the staff would review test abstracts, when they are received, and report the results of its review. In letters dated April 20, 1983, March 28, 1983, and May 2, 1984, and in FSAR Amendments 48 and 49, the applicant provided additional information or test descriptions for these items. A discussion of each test follows.

Item (d): It is the applicant's position that the condensate storage tank auxiliaries should not have to be preoperationally tested because the condensate storage tank is not an engineered safety feature (ESF) and is backed up by the essential raw cooling water system, which is an ESF and is preoperationally tested. The staff accepts this position and considers testing as part of the NCS Test Program adequate. Therefore, this item is resolved.

Item (e): The applicant stated that the 48-V dc system should not be preoperationally tested because it serves only as a backup power supply to a nonessential communications system. The 48-V dc system is a backup power supply to the Southern Bell DIMENSION system. The DIMENSION system is a primary system for offsite and interplant communications. However, it is not required for emergency shutdown, emergency response, or emergency notifications. Other communications systems that are tested will be available for emergency use. The staff therefore concludes that the 48-V dc system need not be preoperationally tested in accordance with FSAR Chapter 14 or as part of the NCS Test Program.

Items (f), (i), and (j): These items are now described in the FSAR preoperational test abstracts. The staff has reviewed the abstracts and found them acceptable. These items are considered resolved.

Item (k): The applicant has provided a test description for the station drainage water system, but it was not clear that this preoperational test included testing of the leakage detection systems of concern. However, a commitment was made in FSAR Amendment 35, Chapter 6, page 6.3-23a, that the individual detectors located in each emergency core cooling system (ECCS) pump compartment, in the ECCS heat exchanger rooms, in the pipe gallery for each unit, and in the pipe chase are to be preoperationally tested to verify initial operability. These leakage detection systems monitor the residual heat removal (RHR) pump compartments as part of the ECCS. The RHR pumps also provide flow to the containment spray system. The staff concludes that acceptable testing will be performed by fulfillment of this commitment, and, therefore, Item (k) is resolved.

Although all of the staff concerns discussed in the SER have been resolved, new concerns arose because of applicant changes to the previously reviewed test program (post-SER FSAR amendments). The natural circulation test, power coefficient test, and control rod drop and plant trip tests had been deleted without justification. These concerns were addressed by the applicant in the letter dated May 2, 1984, as discussed below:

Natural Circulation Test

The natural circulation test, which satisfies TMI Task Action Plan Item I.G.1, had been deleted. By letter dated May 2, 1984, the applicant stated that this test is to be reinstated. Therefore, this concern is resolved.

Power Coefficient Test

The applicant proposes performing the power coefficient measurement only at full power. The justifications offered for the modified power coefficient measurement are:

(1) The power coefficient is not measured directly.

(2) The measurement is time consuming compared with the value of the data.

(3) The measurement also was previously deleted in another plant (McGuire Unit 2) because the measurement had been made in an identical sister plant.

Watts Bar Nuclear Plant Units 1 and 2 are nearly identical to the Sequoyah Nuclear Plant Units 1 and 2 and the McGuire Operating Station Units 1 and 2. The power coefficient or Doppler coefficient (the major component of the power coefficient) was measured and compared with the design values in three of these plants. The comparison between the measurements and design values was within the acceptance criteria in all cases, demonstrating the ability to analytically predict this design parameter. In addition, this modification to the power coefficient measurement is consistent with the deletion of the measurement in the McGuire Unit 2 test program. Therefore, the staff concludes that the proposed reduced power coefficient measurements are acceptable.

Control Rod Drop and Plant Performance Test

The applicant requests deleting this test entirely from the test program on the basis that the trip function and other test objectives are accomplished in the

Technical Specification surveillance testing and other startup tests. The original primary intent of this test was to demonstrate that control rods of low reactivity worth and/or distant from external neutron detectors were capable of activating the negative rate two-out-of-four power level detector trips. The safety concern was that, at full power, if a control rod(s) was dropped while in automatic control and a reactor scram did not occur, the automatic control would try to reestablish the predrop power level and in the attempt would overshoot the steady-state power level. The dynamics of this power overshoot might produce the conditions leading to violation of the departure from nucleate boiling (DNB) limit.

The nuclear steam supply system (NSSS) vendor (Westinghouse) has presented analytical studies that define detection limits for control rod reactivity worths and/or locations and have shown that the DNB limit is not violated for rods below the detection limit. In addition, the control rod drop tests performed at several similar plants have demonstrated that rods well below the analytically defined detection limits are capable of scramming the plant.

The staff has reviewed the information presented by the applicant. On the basis of the NSSS vendor's analyses, generic testing at similar plants, and the accomplishment of other test objectives elsewhere in the test program, the staff concludes that it is acceptable to delete the control rod drop and plant performance test from the Watts Bar Nuclear Plant, Units 1 and 2, test program.

Sequence and Administrative Changes

In addition to the above items, the applicant proposed making the following administrative and test sequence changes:

(1) Modify test sequence documents by removing Tests SU-3.8C, SU-4.10A, and SU-4.11, and make these tests individual test documents without changing the objectives or method of performance.

(2) Incorporate Test SU-2.5, "Inverse Count Rate Monitoring for Core Loading," into Test SU-2.1, "Initial Core Loading," and Test SU-3.1, "Inverse Count Rate Monitoring for Approach to Critical," into Test SU-3.2, "Initial Criticality for Ease of Test Performance."

(3) Incorporate Tests SU-3.4, SU-3.7, and SU-4.2 into plant technical instructions (TIs) that have been reviewed by the Plant Operations Review Committee. Test SU-3.4, "Reactivity Computer," will be incorporated into TI-7, "Reactivity Computer Checkout," and Tests SU-3.7, "Incore Moveable Detectors," and SU-4.2, "Thermocouple Mapping," will be incorporated into TI-41, "Incore Flux Mapping."

The staff finds the sequence and administrative changes acceptable since the originally reviewed test objectives and methods are not changed.

Additional Concerns

The applicant has made other modifications to the Watts Bar initial test program in additional amendments to the FSAR (through Amendment 53). By letter dated December 19, 1984, the staff requested additional information regarding these changes. These items will be pursued with the applicant, and the results of the staff review will be reported in a future supplement.

Conclusions

The staff concludes that the Watts Bar initial test program described in the application, with the exception of the above open items, will meet the acceptance criteria of SRP Section 14.2 (NUREG-0800) and that the successful completion of the program will demonstrate the functional adequacy of plant structures, systems, and components. The results of the staff's review of the remaining open items will be reported in a future supplement.

15 ACCIDENT ANALYSIS

15.3 Limiting Accidents

15.3.2 Steamline Break

In a letter dated May 2, 1984, the applicant proposed removal of the boron concentration requirement of 20,000 ppm in the boron injection tank (BIT) and proposed changes to the Technical Specifications that would delete those Technical Specifications that address the boron concentration and heat tracing for the BIT.

The BIT was incorporated in the plant design to mitigate the consequences of postulated steamline break (SLB) events. During these events, the high head safety injection pumps automatically align to discharge through the BIT, which contains a highly concentrated boric acid solution (20,000 ppm). This solution is then flushed into the primary system to ensure adequate shutdown reactivity. The high head safety injection pumps take suction from the refueling storage tank, which contains borated water at a concentration of 2,000 ppm.

In the revised analyses to support the proposed design change, the BIT is assumed to contain a boron concentration of zero ppm. The applicant reanalyzed the SLB accident using the same computer codes and assumptions as for the previous analysis. The results indicate that the reactor returns to power with a maximum heat flux of approximately 18% of the design and a corresponding reactor coolant system pressure of approximately 800 psia. The minimum departure from nucleate boiling ratio remains above the 1.30 limit.

Although limited cladding perforation following an SLB event is permitted by the Standard Review Plan (NUREG-0800), the applicant has demonstrated that no cladding perforation is predicted to occur. On the basis of its review of the applicant's evaluation, the staff concludes that there is no significant change in the safety margin.

With regard to the proposed changes to the Technical Specifications and heat tracing for the BIT, the applicant stated that the current requirement was due to high boron concentration in the BIT and associated piping. Reduction of the boron concentration requirement to zero ppm would eliminate all Technical Specifications concerning BIT boron concentration, temperatures, and associated surveillance, including heat tracing, since heat tracing would only be required for boron concentrations above 4 weight percent corresponding to approximately 7,000 ppm.

On the basis of its review of the applicant's evaluation, the staff concludes that there is no significant reduction in the safety margin. The supporting analysis also demonstrates compliance with SRP Section 15.1.5 (NUREG-0800). Therefore, the staff finds the applicant's proposal acceptable.

15.3.6 Anticipated Transients Without Scram

Generic Letter 83-28 was issued by the NRC on July 8, 1983, indicating actions to be taken by applicants based on the generic implication of the Salem anticipated transients without scram (ATWS) events. Item 4.3 of the generic letter requires that modifications be made to improve the reliability of the reactor trip system by implementation of an automatic actuation of the shunt attachment on the reactor trip breakers. By letter dated June 4, 1984, the applicant provided responses to the plant-specific questions identified by the staff in its SER on the generic Westinghouse design submitted by letter dated August 10, 1983. The staff has reviewed the applicant's proposed design for the automatic actuation of reactor trip breaker shunt trip attachments and finds it acceptable.

Evaluation

The staff requested the applicant to submit information on the following required plant-specific items on the basis of its review of the Westinghouse Owners Group (WOG)-proposed generic design for this modification. Below is a discussion of the applicant's responses.

(1) Provide the electrical schematic/elementary diagrams for the reactor trip and bypass breakers showing the undervoltage and shunt coil actuation circuits as well as the breaker control (e.g., closing) circuits, and circuits providing breaker status information/alarms to the control room.

The design of the electrical circuits for the shunt trip modification has been reviewed and found consistent with the WOG generic proposed design, which was previously reviewed and approved by the staff. However, the applicant's design includes test jacks to facilitate response time testing during plant operation. This addition to the WOG generic design consists of test jacks wired directly to an auxiliary switch "a" contact and test jacks wired in series with 1-kilohm (2.5-W) resistors across the undervoltage coil. Thus, test connections for an undervoltage trip signal and breaker-tripped condition are available to perform the response time test. The resistors in series with the test connections to the undervoltage coil provide protection against potential accidental shorts or grounds during response time testing to ensure that such events would not result in an inadvertent breaker trip or overload on the protection system power source for the undervoltage trip attachment. On the basis of its review of these plant-specific aspects of the applicant's design, the staff concludes that they do not introduce a safety-significant consideration, will facilitate on-line response time testing, and are, therefore, acceptable.

(2) Identify the power sources for the shunt trip coils. Verify that they are Class 1E and that all components providing power to the shunt trip circuitry are Class 1E and that any faults within non-Class 1E circuitry will not degrade the shunt trip function. Describe the annunciation/indication provided in the control room upon loss of power to the shunt trip circuits. Also describe the overvoltage protection and/or alarms provided to prevent or alert the operator(s) to an overvoltage condition that could affect both the undervoltage (UV) coil and the parallel shunt trip actuation relay.

Redundant Class 1E power sources are used for the shunt trip actuation of the reactor trip breakers and for the shunt trip of the bypass breakers. Class 1E circuitry is separated from non-Class 1E circuitry in accordance with RG 1.75 and is, therefore, acceptable.

The breaker position status lights are used to supervise the availability of power to the shunt trip circuits. The red light, which is connected in series with the shunt coil and the "a" auxiliary contact, indicates that the breaker is closed and that the power is available to the shunt trip device and, therefore, provides detectability of power failure to the shunt trip coil. Also, normally open contacts of an auxiliary relay that is energized when the breaker is closed provide breaker-status information to the plant computer. These contacts would change state if power for the shunt trip was lost. Main control room annunciation is provided upon trip of any 125-V dc breaker in the battery board. Also, undervoltage and overvoltage annunciation is provided for the shunt trip circuit's 125-V dc power source. Annunciation is provided for the loss of any one of the two 48-V dc power supplies in each train of the solid-state protection system (SSPS).

Normally, the shunt trip coils in the reactor trip breakers are in the de-energized condition when the trip breakers are closed. The red lamp current (approximately 50 ma) flows through the trip coil to monitor the circuit continuity, which is not large enough to actuate the trip coil armature. Since the current through the shunt trip coils is interrupted when the breaker trips, energization of the shunt trip coil is only momentary. The maximum available voltage occurs during a battery equalizing charge at a maximum voltage of 115% of the nominal voltage. Because of the short-duty cycle of the shunt trip coil, it can operate at this overvoltage condition without harmful effects.

The added shunt trip circuitry is powered from the reactor protection logic power supply. The reactor protection logic power supply consists of two redundant 48-V dc power supplies for each train of the SSPS. Components in the added shunt trip circuitry have been selected on the basis of their ability to perform their intended function up to 115% of nominal voltage. The overvoltage protection in each redundant reactor protection logic power supply is set at 115% of nominal voltage.

On the basis of its review, the staff concludes that appropriate consideration has been given to the aspects of the design described above, and the design is, therefore, acceptable.

(3) Verify that the relays added for the automatic shunt trip function are within the capacity of their associated power supplies and that the relay contacts are adequately sized to accomplish the shunt trip function. If the added relays are other than Potter & Brumfield MDR series relays (P/N 2383A38 or P/N 955655) recommended by Westinghouse, provide a description of the relays and their design specifications.

The design at Watts Bar includes the Potter & Brumfield MDR series P/N 955655 relays as specified in the WOG generic design for the automatic shunt trip function. The relay contacts are adequately sized to accomplish the shunt trip function. The staff finds this aspect of the design acceptable.

(4) State whether the test procedure/sequence used to independently verify operability of the undervoltage and shunt trip devices in response to an automatic reactor trip signal is identical to the test procedure proposed by the Westinghouse Owners Group (WOG). Identify any differences between the WOG test procedure and the test procedure to be used and provide the rationale/justification for these differences.

The applicant notes that the steps used to independently confirm the operability of the undervoltage trip and shunt trip devices in response to an automatic reactor trip signal are identical to the test procedure proposed by the WOG to NRC by letter No. OG-101 dated June 14, 1983. This procedure will be incorporated into Surveillance Instruction (SI) 3.1.26 before Unit 1 fuel load. The staff finds this acceptable.

(5) Verify that the circuitry used to implement the automatic shunt trip function is Class 1E (safety related), and that the procurement, installation, operation, testing, and maintenance of this circuitry will be in accordance with the quality assurance criteria set forth in Appendix B to 10 CFR 50.

The applicant confirmed that the circuitry used to implement the automatic shunt trip function is Class 1E (safety related) and the procurement, installation, operation, testing, and maintenance of this circuitry will be in accordance with the TVA nuclear power station quality assurance program and Watts Bar-specific quality assurance procedures which satisfy Appendix B to 10 CFR 50. The staff finds this acceptable.

(6) Verify that the shunt trip attachments and associated circuitry are/will be seismically qualified (i.e., be demonstrated to be operable during and after a seismic event) in accordance with the provisions of Regulatory Guide 1.100, Revision 1, which endorses IEEE Standard 344, and that all nonsafety-related circuitry/components in physical proximity to or associated with the automatic shunt trip function will not degrade this function during or after a seismic event.

The applicant notes that the shunt trip attachments and associated circuitry will be seismically qualified in accordance with IEEE Standard 344. The WOG is working with Westinghouse to obtain seismic qualification of the shunt trip attachments. Nonsafety-related circuits are isolated from safety-related circuits by qualified isolators. The shunt trip circuitry is located within seismically qualified Class 1E reactor trip switchgear. The staff finds this acceptable pending verification of the seismic qualification of the shunt trip attachments.

(7) Verify that the components used to accomplish the automatic shunt trip function are designed for the environment where they are located.

The applicant has verified that the plant-specific environmental conditions defined in Table 1 of the WOG generic design package envelope Watts Bar Units 1 and 2. The staff finds this acceptable.

(8) Describe the physical separation provided between the circuits used to manually initiate the shunt trip attachments of the redundant reactor trip breakers. If physical separation is not maintained between these

circuits, demonstrate that faults within these circuits cannot degrade both redundant trains.

Physical separation between the circuits used to manually initiate the shunt trip attachments of the redundant trip breakers is maintained by routing the field cabling from the main control board and reactor protection logic to redundant train A and train B reactor trip switchgear as train A and train B circuits. Coil-to-contact isolation is provided within reactor trip switchgear, and metal-braid-enclosed cabling is used for train A and train B wiring where a 6-in. air gap is not maintained. Dual section manual reactor trip switches with metal barriers are used between redundant train switch decks. The staff finds this meets the requirements of RG 1.75 and is, therefore, acceptable.

(9) Verify that the operability of the control room manual reactor trip switch contacts and wiring will be adequately tested prior to startup after each refueling outage. Verify that the test procedure used will not involve installing jumpers, lifting leads, or pulling fuses and identify any deviations from the WOG procedure. Permanently installed test connections (i.e., to allow connection of a voltmeter) are acceptable.

The applicant notes that all control room manual reactor trip switch contacts and wiring will be tested before startup after each refueling outage by the performance of SI-3.1.1. The test procedure used does not involve installing jumpers, lifting leads, or pulling fuses. The UV coil voltage will be monitored by the voltmeter permanently installed on the SSPS test panel. The shunt coil voltage will be monitored by temporarily connecting a voltmeter across the combination of shunt trip coil and the series connected "a" auxiliary contact. This procedure will be incorporated into SI-3.1.1 before Unit 1 fuel loading. The staff finds this acceptable.

(10) Verify that each bypass breaker will be tested to demonstrate its operability prior to placing it into service for reactor trip breaker testing.

The applicant notes that before the reactor trip breaker test during power operation is performed, the operability of the bypass breaker required for testing will be verified by racking the bypass breaker into test position and tripping the breaker open. This procedure will be incorporated into SI-3.1.26 before Unit 1 fuel loading. The staff finds this acceptable. However, the staff will require that the bypass breaker undervoltage trip attachment be demonstrated operable at a refueling outage frequency.

(11) Verify that test procedures used to determine reactor trip breaker operability will also demonstrate proper operation of the associated control room indication/annunciation.

The applicant notes that the reactor trip breaker tests will verify the proper operation of the main control room reactor trip breaker indication. The staff finds this acceptable.

(12) Verify that the response time of the automatic shunt trip feature will be tested periodically and shown to be less than or equal to that assumed in the FSAR analyses or that specified in the Technical Specifications.

The applicant notes that the response time of the automatic shunt trip will be tested and used in demonstrating that the reactor trip system instrumentation response times specified in the Technical Specifications are within their limits. The staff finds this acceptable.

(13) Propose Technical Specification changes to require periodic testing of the undervoltage and shunt trip functions and the manual reactor trip switch contacts and wiring.

The applicant notes that Technical Specifications require a manual reactor trip actuating device operational test be performed at least once every 18 months and a reactor trip breaker trip actuating device operational test be performed on each train at least every 62 days on a staggered test basis. The applicant further notes that the Technical Specifications require that at least once every 18 months and following maintenance or adjustment of the reactor trip breakers, the trip actuating device operational test shall include independent verification of the UV and shunt trips. The staff will review the plant Technical Specifications to confirm that they include surveillance requirements consistent with the applicant's commitments. This review will be completed before the operating license for Unit 1 is issued.

Conclusion

On the basis of the review of the applicant's response to the plant-specific questions identified in the staff's evaluation of the WOG generic design modifications, the staff finds that these modifications are acceptable.

However, the staff will require that the applicant submit confirmation that the seismic qualification of the shunt trip attachment has been successfully completed as noted in Item (6). Further, the staff requires that proposed Technical Specifications be submitted, which are responsive to the staff requirements noted in Items (10) and (13), following implementation of this modification.

15.4 Radiological Consequences of Accidents

15.4.3 Steam Generator Tube Rupture

In the Watts Bar FSAR, the applicant made general, unverified assumptions concerning system performance following a complete severance of a single steam generator tube. In addition, the FSAR assumed that the break flow was terminated within 30 min of the event by operator actions to equalize the primary and secondary pressures. In the SER, the staff addressed the accident, including the sequence of events and the radiological consequences, and found them acceptable. However, the actual steam generator tube rupture (SGTR) event that occurred at Ginna indicated that more than 30 min could be required for pressure equalization, implying that the Watts Bar analysis was nonconservative with respect to assumed operator actions.

As a result, by letter dated May 3, 1984, the staff requested additional information, including an evaluation of operator action times, as to whether liquid can enter the steamlines and the effects on the integrity of steam piping and supports. The staff also requested that a reactor systems analysis be performed for natural circulation cooldown with an SGTR, including the effect of the worst single failure of a system that is either required or expected to operate during the event.

By letter dated June 11, 1984, TVA stated that it has joined the Westinghouse Owners Group, which is investigating the SGTR issue on a generic basis. The owners group plans to issue a report late in 1984. The applicant has committed to implement all corrective actions recommended by the owners group as approved by the NRC staff before startup following the first refueling outage for Unit 1.

As to whether there is adequate assurance that the Watts Bar plant can operate safely for one cycle of operation until the SGTR issue is satisfactorily resolved, the staff notes the following:

(1) All components necessary for mitigation of the design-basis SGTR are safety related.

(2) The plant emergency procedures for an SGTR have been reviewed and approved by the NRC staff.

(3) There is a low probability of a design-basis SGTR in the first cycle of operation.

Therefore, the staff concludes that there is sufficient assurance that the Watts Bar plant can operate safely for one cycle until this issue is resolved. The staff will condition the license to require satisfactory resolution of this issue before the startup following the first refueling outage of Unit 1.

APPENDIX A

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

December 22, 1980	Letter to applicant regarding control of heavy loads (NUREG-0612).
February 18, 1981	Letter to applicant concerning post-TMI requirements for the emergency operations facility (Generic Letter 81-10).
January 27, 1982	Letter from applicant transmitting Revision 0 to Radiological Emergency Plan.
February 25, 1982	Letter from applicant regarding loose parts monitoring program.
August 12, 1982	Letter from applicant regarding NUREG-0737, Items II.B.1, II.B.2, II.F.1, and II.F.2.
June 3, 1983	Letter from applicant concerning responses to Seismic Qualification Review Team (SQRT) reviewer concerns.
June 10, 1983	Letter from applicant concerning responses to SQRT reviewer concerns.
June 29, 1983	Letter from applicant regarding reactor vessel level indication system installation and operation.
August 10, 1983	Letter to Westinghouse transmitting SER on generic Westinghouse design of the shunt attachment on reactor trip breakers.
January 4, 1984	Letter from applicant providing additional information concerning NUREG-0737, Item II.F.2.
January 5, 1984	Letter from applicant concerning automatic trip of reactor coolant pumps.
January 5, 1984	Letter to applicant concerning NRC use of the terms, "important to safety" and "safety related" (Generic Letter 84-01).
January 5, 1984	Letter from applicant forwarding safety parameter display system safety analysis.
January 6, 1984	Letter to applicant concerning notice of meeting regarding facility staffing (Generic Letter 84-02).

January 6, 1984	Letter from applicant forwarding Amendment 49 to FSAR.
January 9, 1984	Letter from applicant concerning compliance with General Design Criterion (GDC) 51.
January 9, 1984	Letter from applicant forwarding drawings showing vital area designations.
January 10, 1984	Letter from applicant updating status report for items listed in Appendix D of SER.
January 11, 1984	Letter from applicant concerning equipment qualification program.
January 12, 1984	Letter to applicant concerning facility staffing survey.
January 12, 1984	Letter to applicant concerning diesel generator auxiliary systems design deficiencies.
January 12, 1984	Letter from applicant concerning diesel engines.
January 12, 1984	Letter from applicant concerning milestone schedules for the prompt notification system.
January 12, 1984	Letter from applicant concerning the fire protection program.
January 12, 1984	Letter from applicant concerning the issue of liquefaction potential.
January 13, 1984	Letter to applicant concerning Physical Security Plan - designation of vital equipment.
January 13, 1984	Letter to applicant concerning availability of NUREG-0933, "A Prioritization of Generic Safety Issues" (Generic Letter 84-03).
January 17, 1984	Letter from applicant concerning NRC Power Systems Branch status list of open and confirmatory items and license conditions.
January 17, 1984	Letter from applicant concerning equipment qualification program.
January 19, 1984	Letter from applicant concerning supervision of fire alarm and detection circuits.
January 23, 1984	Letter from applicant informing that fuel loading is now anticipated to be June 1984 for Unit 1 and December 1985 for Unit 2.
January 24, 1984	Letter from applicant concerning installation of high-range noble gas monitors on the auxiliary building vent.

January 26, 1984	Letter from applicant forwarding Amendment 49 to FSAR.
January 30, 1984	Letter from applicant concerning Underwriters Laboratory test report for the 1-hour-fire-rated barriers being used to separate redundant safe shutdown circuits.
January 30, 1984	Letter from applicant responding to NRC concerns regarding the Physical Security Plan.
January 30, 1984	Letter to applicant issuing Supplement 2 to Safety Evaluation Report.
January 30, 1984	Letter from applicant concerning compliance with Regulatory Guide 1.97.
January 31, 1984	Letter from applicant concerning changes to Technical Specification surveillance requirements for the diesel fuel impurity level tests.
February 1, 1984	Letter to applicant concerning safety evaluation of Westinghouse topical reports dealing with elimination of postulated pipe breaks in PWR primary main loops (Generic Letter 84-04).
February 3, 1984	Letter to applicant concerning submittal of offsite dose calculations manual.
February 6, 1984	Letter to applicant concerning Physical Security Plan - designation of vital equipment.
February 6, 1984	Letter from applicant concerning control of heavy loads at nuclear power plants, Generic Letter 81-07.
February 7, 1984	Letter from applicant concerning limiting tank concentration value.
February 7, 1984	Letter from applicant submitting "TVA Watts Bar Nuclear Plant Auxiliary Feedwater System Independent Review."
February 9, 1984	Letter from applicant concerning facility staffing survey.
February 9, 1984	Letter from applicant concerning updated status on fire protection and miscellaneous commitments.
February 10, 1984	Meeting with applicant to discuss welding codes used. (Summary issued March 1, 1984)
February 14-16, 1984	Meeting with applicant at site to audit equipment qualification program. (Summary issued March 14, 1984)
February 15, 1984	Letter to applicant concerning review status of the application.

February 15, 1984	Letter from applicant providing additional information concerning the Radiological Emergency Plan emergency action levels.
February 17, 1984	Letter to applicant concerning compliance with Item II.K.3.17 of NUREG-0737.
February 21, 1984	Letter from applicant concerning status of outstanding and confirmatory items.
February 27, 1984	Letter from applicant concerning calculation of vertical temperature gradients.
February 28, 1984	Letter from applicant forwarding Revision 4 to Physical Security Plan.
March 6, 1984	Letter to applicant concerning diesel generator auxiliary systems design deficiencies.
March 6, 1984	Letter to applicant requesting additional information concerning use of containment high-range monitor readings as an indicator of core damage.
March 7, 1984	Letter from applicant concerning various NRC positions described in the SER.
March 8, 1984	Letter to applicant requesting additional information concerning use of containment high-range monitor readings.
March 9, 1984	Letter to applicant requesting additional information regarding the initial test program.
March 13, 1984	Letter to applicant concerning changes to the FSAR.
March 13, 1984	Letter to applicant concerning deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans.
March 14, 1984	Letter to applicant concerning equipment qualification audit.
March 14, 1984	Letter from applicant concerning 175-ton polar crane.
March 14, 1984	Letter to applicant concerning compliance with GDC 51.
March 15-16, 1984	Meeting with applicant to conduct the confirmatory site visit.
March 20, 1984	Letter from applicant concerning control of heavy loads.
March 21, 1984	Letter from applicant concerning qualification of the diesel generator auxiliary system piping.

March 21, 1984	Letter from applicant forwarding revision to the Radiological Emergency Plan.
March 26, 1984	Letter to applicant issuing order extending construction completion dates for Unit 1 to January 1, 1985, and Unit 2 to July 1, 1986.
March 27, 1984	Letter from applicant concerning proposed modifications to the draft Technical Specifications.
March 28, 1984	Letter from applicant providing additional information relating to the installation and preoperational testing schedules for the reactor vessel level instrumentation system.
March 28, 1984	Letter from applicant concerning modifications to the radiological environmental monitoring program.
March 29, 1984	Meeting with applicant to discuss proposed changes to TVA's Nuclear Safety Review Board. (Summary issued April 11, 1984)
March 29, 1984	Letter from applicant concerning TVA's evaluation of the Black and Veatch independent review of the Watts Bar plant.
March 30, 1984	Letter from applicant concerning radiographic inspection of the wedge located in each of the main feedwater isolation valves.
April 2, 1984	Letter to applicant forwarding change to NUREG-1021, "Operator Licensing Examiner Standard" (Generic Letter 84-05).
April 4, 1984	Letter to applicant concerning interim procedures for NRC management of plant-specific backfitting (Generic Letter 84-08).
April 6, 1984	Letter from applicant concerning remedial action to resolve the issue of potential liquefaction.
April 6, 1984	Letter from applicant concerning environmental qualification of the safety-related mechanical equipment.
April 6, 1984	Letter from applicant responding to NUREG-0737, Item II.K.3.10.
April 9, 1984	Meeting with applicant to discuss the Black and Veatch independent design review. (Summary issued May 22, 1984)
April 10, 1984	Letter from applicant forwarding Revision 6 to Physical Security Plan.

April 13, 1984	Letter to applicant concerning buckling criteria for Class 2 and 3 supports.
April 13, 1984	Letter from applicant forwarding revision of Technical Instruction to Preservice Inspection Program.
April 13, 1984	Letter from applicant forwarding revision to Implementing Procedures Document.
April 18, 1984	Letter from applicant concerning the prompt notification system.
April 19, 1984	Letter from applicant responding to Office of Inspection and Enforcement (IE) Bulletin 80-06 regarding controls for feedwater isolation and containment purge isolation valves.
April 23, 1984	Letter to applicant requesting additional information regarding deliberate ignition hydrogen control.
April 24, 1984	Letter from applicant concerning the offsite power automatic transfer scheme, undervoltage alarms, and potential submergence of electrical equipment.
April 25, 1984	Letter to applicant concerning supplement regarding the seismic and dynamic qualification of safety-related electrical and mechanical equipment.
April 25, 1984	Letter to applicant concerning use of ASME Code Case N-32-4 for hydrostatic testing of Section III, Class 3, embedded piping.
April 26, 1984	Letter to applicant concerning administration of operating tests before initial criticality (10 CFR 55.25) (Generic Letter 84-10).
April 30, 1984	Letter to applicant concerning compliance with 10 CFR 61 and implementation of the Radiological Effluent Technical Specifications (RETS) and attendant process control program (PCP) (Generic Letter 84-12).
May 1, 1984	Letter from applicant forwarding Amendment 50 to FSAR.
May 2, 1984	Letter from applicant concerning deletion of startup tests from FSAR.
May 2, 1984	Letter from applicant concerning Westinghouse study, "Report for the BIT Concentration Reduction/BIT Elimination Study for Watts Bar Units 1 and 2."
May 3, 1984	Letter to applicant concerning Technical Specification for snubbers (Generic Letter 84-13).

May 3, 1984	Letter to applicant requesting additional information concerning steam generator tube rupture.
May 3, 1984	Letter to applicant concerning compliance with GDC 51.
May 4, 1984	Letter from applicant forwarding Revision 2 to Radiological Emergency Plan IPD and revision to Radiological Emergency Plan.
May 7, 1984	Letter from applicant concerning status of various commitments concerning control room modifications.
May 8, 1984	Letter from applicant concerning proposed changes to draft Technical Specifications.
May 8, 1984	Letter from applicant concerning current status list for outstanding and confirmatory issues.
May 8, 1984	Letter from applicant providing updated status on certain items related to fire protection and updated information on miscellaneous commitments.
May 8, 1984	Letter from applicant concerning requests for exemptions from requirements of NUREG-0737, Item II.F.1, and requirement to install the high-range noble gas monitors on the auxiliary building vent.
May 9, 1984	Letter to applicant requesting additional information regarding the financial qualifications review.
May 9, 1984	Letter from applicant forwarding Amendment 51 to FSAR.
May 10, 1984	Letter from applicant advising that fuel load date for Unit 1 is now July 1984 and for Unit 2 is now June 1986.
May 10, 1984	Meeting with applicant at site to discuss resolution of open items remaining in the Physical Security Plan.
May 14, 1984	Letter from applicant concerning buckling criteria for Class 2 and 3 supports.
May 14, 1984	Letter from applicant transmitting proposed modifications to draft Technical Specifications related to testing of circuit breakers.
May 15, 1984	Letter to applicant providing comments on the proposed offsite dose calculation manual.
May 16, 1984	Letter from applicant concerning containment design in comparison with current buckling studies.
May 16, 1984	Letter to applicant requesting additional information regarding the Physical Security Plan.

May 17, 1984	Letter from applicant concerning seismic qualification of equipment.
May 25, 1984	Letter from applicant concerning hydrogen control for degraded core accidents.
May 25, 1984	Letter from applicant concerning seismic qualification of equipment.
May 29, 1984	Letter to applicant requesting additional information regarding the financial qualifications review.
May 30, 1984	Letter to applicant concerning review of utility onshift operating experience.
May 30, 1984	Letter from applicant requesting license contain exemptions for performing natural circulation tests.
June 4, 1984	Letter from applicant concerning NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations."
June 4, 1984	Letter from applicant concerning Generic Letter 82-28, "Required Actions Based on Generic Implications of Salem ATWS Events."
June 5, 1984	Letter to applicant concerning TVA's reorganization.
June 6, 1984	Letter to applicant concerning vital area definition reports.
June 7, 1984	Letter from applicant concerning overpressure protection.
June 7, 1984	Letter from applicant concerning feedwater isolation valve wedges.
June 11, 1984	Letter from applicant concerning the analysis and mitigation of steam generator tube rupture events.
June 11, 1984	Letter from applicant providing Revision 7 to Physical Security Plan.
June 13, 1984	Letter from applicant concerning audit of electrical equipment qualification program.
June 13, 1984	Letter from applicant concerning deletion of the rod drop and plant trip test.
June 14, 1984	Meeting with applicant to discuss TVA's generic control room design review program.
June 15, 1984	Letter from applicant concerning cement mortar lining in the essential raw cooling water piping.

June 18, 1984	Letter from applicant forwarding Amendment 52 to FSAR.
June 19, 1984	Letter from applicant providing comments/proposed modifications to Unit 1 Technical Specifications.
June 19, 1984	Letter from applicant concerning seismic qualification of equipment.
June 25, 1984	Letter from applicant responding to Generic Letter 83-10c concerning automatic trip of reactor coolant pumps.
June 25, 1984	Letter from applicant concerning NRC review of confirmatory items, license conditions, and miscellaneous items.
June 26, 1984	Letter from applicant forwarding Amendment 53 to FSAR.
June 26, 1984	Letter from applicant concerning status of control room modifications required for fuel loading.
June 27, 1984	Letter to applicant concerning adequacy of onshift operating experience for near-term operating license applicants (Generic Letter 84-16).
July 3, 1984	Letter to applicant concerning annual meeting to discuss recent developments regarding operator training, qualifications, and examinations (Generic Letter 84-17).
July 6, 1984	Letter from applicant concerning experience levels of operating crew personnel.
July 6, 1984	Letter from applicant concerning financial qualifications.
July 10, 1984	Letter from applicant concerning ASME Code, Section XI, Preservice Inspection Program Technical Instruction TI-50A.
July 11, 1984	Letter from applicant providing Revision 8 to Physical Security Plan.
July 13, 1984	Letter from applicant concerning postaccident sampling system.
July 19, 1984	Letter from applicant concerning relationship between the containment high-range radiation monitor reading and the radioactivity uniformly dispersed in the containment atmosphere.
July 20, 1984	Letter from applicant concerning NUREG-0737, Item II.D.1.1, "Integrity of Systems Outside Containment."
July 24, 1984	Letter from applicant concerning audit of electrical equipment qualification files.

July 24, 1984	Letter from applicant specifying new fuel load date for Unit 1 as October 1984 and for Unit 2 as August 1986.
July 27, 1984	Letter from applicant concerning various comments/ proposed modifications to Unit 1 draft Technical Specifications.
July 27, 1984	Letter from applicant concerning NUREG-0737, Item I.C.7, "NSSS Vendor Review of Procedures."
August 6, 1984	Letter to applicant concerning availability of Supplement to NUREG-0933, "A Prioritization of Generic Safety Issues" (Generic Letter 84-19).
August 7-8, 1984	Meeting with applicant to discuss the Technical Specifications for Unit 1.
December 19, 1984	Letter to applicant requesting additional information concerning the initial test program.

APPENDIX B

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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-46 Seismic Qualification of Equipment in Operating Plants

The scope of Task A-46 is limited to dealing with the seismic qualification of equipment in currently operating plants. The staff's evaluation of the Watts Bar seismic qualification of equipment is discussed in Section 3.10 of this report. The evaluation will not be handled under USI A-46 because it is being handled on a case basis; therefore, the open item identified under this USI should be deleted.

APPENDIX E

PRINCIPAL CONTRIBUTORS

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16. ABSTRACT (200 words or less)

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982), Supplement No. 1 (September 1982), and Supplement No. 2 (January 1984) 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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