NUREG–0847
Supplement No. 5

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

November 1990

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ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), and Supplement No. 4 (March 1985), issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items and proposed license conditions identified in the SER.

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**Titled "General" in SER.
†Titled "Conclusions" in SER.

ABBREVIATIONS

ABGTS	auxiliary building gas treatment system
ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BISI	bypassed and inoperable status indication
BTP	branch technical position
CAP	corrective action program
CFR	Code of Federal Regulations
CIS	containment isolation system
CNPP	Corporate Nuclear Performance Plan
CSB	Control Systems Branch
CSOC	closed system outside containment
DCRDR	detailed control room design review
DOE	Department of Energy
ECCS	emergency core cooling system
EGTS	emergency gas treatment system
EMD-GM	Electromotive Division of General Motors
ERCW	emergency raw cooling water
ERCWS	emergency raw cooling water system
ESF	engineered safety feature
FEMA	Federal Emergency Management Administration
FSAR	final safety analysis report
GDC	general design criterion
GL	generic letter
HAUP	hanger and analysis update program
HED	human engineering deficiency
HVAC	heating, ventilation, and air conditioning
IDVP	independent design verification program
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
ISI	inservice inspection
IST	inservice testing
LANL	Los Alamos National Laboratory
LBB	leak before break
LOCA	loss-of-coolant accident
LPMS	loose-parts monitoring system

MERITS	Methodically Engineered, Restructured, and Improved Technical Specifications
MSLB	main steamline break
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSRS	Nuclear Safety Review Staff
NSSS	nuclear steam supply system
PHMS	permanent hydrogen mitigation system
PMF	probable maximum flood
PORV	power-operated relief valve
PSI	preservice inspection
PWR	pressurized-water reactor
QA	quality assurance
RBPVS	reactor building purge ventilation system
RCCA	rod cluster control assembly
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RTD	resistance temperature detector
SAIC	Science Applications International Corporation
SALP	systematic assessment of licensee performance
SER	safety evaluation report
SGTR	steam generator tube rupture
SI	safety injection
SNM	special nuclear material
SP	special program
SPDS	safety parameter display system
SSER	supplement to SER
SSI	soil-structure interaction
STS	Standard Technical Specifications
TAC	technical assignment control
TIPTOP	turbine integrity program with turbine overspeed protection
TMI-2	Three Mile Island Unit 2
TVA	Tennessee Valley Authority
USI	unresolved safety issue
WBNPP	Watts Bar Nuclear Performance Plan
WISP	Workload Information and Scheduling Program
WOG	Westinghouse Owners Group

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 followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), and Supplement No. 4 (SSER 4, March 1985).

The SER and SSERs were written in accordance with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues that arose as a result of the SRP review that were not closed out at the time the SER was published were classified into outstanding issues, confirmatory issues, and proposed license conditions (see Sections 1.7, 1.8, and 1.9 which follow).

In addition to the guidance of the SRP, the staff would from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The staff is preparing a summary of the implementation status of the applicable ones, and will publish such status in SSER 6 and future SSERs.

Since SSER 4 was issued, Watts Bar licensing activities have been put on hold because of problems identified at TVA plants (see Section 1.13 for details). Thus, no supplements were issued in the ensuing five years. This supplement (SSER 5) provides more recent information regarding the resolution or status of some of the outstanding and confirmatory issues, and license conditions identified in the SER and its supplements. Some of the issues addressed in previous SSERs may be subject to further review as a result of the applicant's corrective actions which are under way (see Section 1.13).

Each of the following sections or appendices of this supplement (SSER 5) is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.* Appendix E is a list of principal contributors to this supplement. This supplement made no changes in Appendices C, D, F, G, and H. In Appendix I, the staff's safety evaluation of September 11, 1990, is reproduced. Similary, in Appendix J, the staff's safety evaluation of May 17, 1990, is reproduced.

The Project Manager is Peter S. Tam. Mr. Tam may be contacted by calling (301) 492-7000, or by writing to the following address:

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

Mr. Peter S. Tam
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Supplement No. 5 updates the status of some of those items. The current status of each of the 17 original issues is tabulated below and the relevant SER or SSER section is indicated. Resolution of those issues that are, to date, unresolved will be addressed in future supplements.

Issue*	Status	Section
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Inservice pump and valve test program (TAC 74801)	Revised (SSER 5)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC 71919)	Updated (SSER 5)	3.10
(b) Environmental (TAC 63591)	Under review	3.11
(5) Preservice inspection program (TAC 63627)	Under review	5.2.4, 6.6
(6) Pressure-temperature limits for Unit 2	Under review	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) BTP CSB 6-4	Resolved (SSER 3); see License Condition 8	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC 63649)	Under review (SSER 2, SSER 3)	8.2
(12) Fire protection program (TAC 63648)	Under review	9.5.1

* The TAC (technical assignment control) number that appears in parentheses after the title is an internal NRC control number, by which relevant documents are filed. Documents associated with each TAC number can be listed by the NRC document control system, NUDOCS/AD.

Issue*	Status	Section
(13) Quality classification of diesel generator auxiliary system piping and components (TAC 63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC 63638)	Resolved (SSER 3, SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC 63657)	Under review	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC 76972)	Revised (SSER 5)	17

In addition to these 17 issues, the staff has, in the 5 years since SSER 4 was published, identified a number of new issues that require resolution. However, these issues have not yet been reviewed to the degree that the staff can classify them as outstanding issues, confirmatory issues, or proposed license conditions. The status of the staff's reviews will be published in future SSERs; for the time being, these issues are tracked by the NRC WISP (Workload Information and Scheduling Program) with the following titles and TAC numbers assigned:

TAC 63607	T_{SAT} Indication in the Auxiliary Control Room
TAC 63621	Main Steam Line Break Inside Containment
TAC 63632	Main Steam Line Break Outside Containment
TAC 63644	Hydrogen/Oxygen Monitoring System
TAC 63647	Health Physics Program
TAC 63657	Physical Security Plan
TAC 77136	Regulatory Guide 1.47, Bypass and Inoperable Status Indication System
TAC 77195	Elimination of Upper Head Injection System
TAC 77550	Conformance With Regulatory Guide 1.97, Instruments To Follow Course of Accident
TAC 77553	Offsite Dose Calculation Manual
TAC 77569	Steam Generator Tube Rupture Analysis per Westinghouse Topical Report WCAP-11698
TAC 77661	Offsite Radiological Monitoring Program

1.8 Confirmatory Issues

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

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* The TAC (technical assignment control) number that appears in parentheses after the title is an internal NRC control number, by which relevant documents are filed. Documents associated with each TAC number can be listed by the NRC document control system, NUDOCS/AD.

Issue		Status	Section
(1)	Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2)	Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3)	Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4)	Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5)	Upgrading ERCW system to seismic Category I (TAC 63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6)	Seismic classification of structures, systems, and components important to safety (TAC 63618)	Resolved (SSER 5)	3.2.1
(7)	Tornado-missile protection of diesel generator exhaust	Resolved (SSER 2)	3.5.2, 9.5.4.1, 9.5.8
(8)	Steel containment building buckling research program	Resolved (SSER 3)	3.8.1
(9)	Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC 63625)	Under review	3.9.3.4
(10)	Thermal performance analysis	Resolved (SSER 2)	4.2.2
(11)	Cladding collapse	Resolved (SSER 2)	4.2.2
(12)	Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13)	Loose-parts monitoring system	Resolved (SSER 3); see License Condition 42	4.4.5
(14)	Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15)	Natural circulation tests (TAC 63603)	Awaiting information	5.4.3
(16)	Atmospheric dump valve testing	Resolved (SSER 2)	5.4.3
(17)	Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18)	Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC 63623)	Resolved (SSER 5)	6.2.4
(19)	Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20)	Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21)	Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1

Issue	Status	Section
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	8.3.1.1
(28) Low and/or degraded grid voltage condition (TAC 63649)	Under review	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC 63649)	Under review	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2 (TAC 63649)	Under review	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC 63649)	Under review	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC 63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC 63648)	Under review	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14
(42) Submergence of electrical equipment as result of a LOCA (TAC 63649)	Under review	8.3.3.1.1

1.9 License Conditions

In Section 1.9 of the SER and Supplement Nos. 1, 2, and 3 to the SER, the staff identified 42 license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The license conditions are tabulated below, with the corresponding NUREG-0737 item number given in parentheses (as appropriate) and the relevant SER or SSER section indicated.

Condition	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2
(2) Inservice testing of pumps and valves (TAC 74801)	Updated (SSER 5)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TAC 63629)	Under review	4.4.8
(4) Inservice Inspection Program (TAC 76881)	Unchanged (SSER 3)	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Resolved (SSER 5)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) noble gas monitor (TAC 63645)	Resolved (SSER 5)	11.7.1
(b) iodine particulate sampling (TAC 63645)	Revised (SSER 5)	11.7.1
(c) high range in-containment radiation monitor (TAC 63645)	Resolved (SSER 5)	12.7.2
(d) containment pressure	Resolved (SSER 5)	6.2.1
(e) containment water level	Resolved (SSER 5)	6.2.1
(f) containment hydrogen	Resolved (SSER 5)	6.2.5
(7) Modification to chemical feedlines (TAC 63622)	Resolved (SSER 5)	6.2.4
(8) Containment isolation dependability (II.E.4.2) (TAC 63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC 77208)	Under review (SER)	6.2.5, App. C
(10) Status monitoring system	Unchanged (SER)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Resolved (SSER 5)	7.8.1
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6

Condition	Status	Section
(13) DC monitoring and annunciation (TAC 63649)	Under review (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
(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) (TAC 63649)	Under review	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (II.B.3) (TAC 77543)	Updated (SSER 3, SSER 5)	9.3.2
(20) Fire protection program (TAC 63648)	Under review	9.5.1
(21) Performance testing for communications systems (TAC 63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC 63640)	Resolved (SSER 5)	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Resolved (SSER 5)	10.3.4
(24) Primary coolant outside containment (III.D.1.1) (TAC 63646)	Updated (SSER 5)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC 63592)	Under review	13.4
(26) Use of experienced personnel during startup (TAC 63592)	Under review	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC 63656)	Under review	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC 63592)	Under review	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC 63592)	Under review	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3
(31) Initial test program (TAC 63651)	Updated (SSER 3, SSER 5)	14.2

Condition	Status	Section
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC 63631)	Resolved (SSER 5)	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC 77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC 63655)	Updated (SSER 5)	18.1
(38) Physical Security Plan (TAC 63657)	Under review	13.6
(39) Control of heavy loads (NUREG-0612) (TAC 77560)	Unchanged (SSER 3)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC 64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC 77569)	Resolved (SSER 3, SSER 5)	15.4.3
(42) Loose-parts monitoring system (TAC 77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC 73723 and 73724)	Opened (SSER 5)	18.2

1.11 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that NRC shall not issue or renew a license for a nuclear power reactor unless the utility has signed a contract with the Department of Energy for disposal services.

By letter dated February 16, 1985, the applicant stated that it has such an agreement (Contract No. DE-CR01-83-NE 44420) with the Department of Energy. This agreement is applicable to both Watts Bar units.

1.12 Approved Technical Issues for Incorporation in the License as Exemptions

The applicant applied for exemptions from certain provisions of the regulations. These have been reviewed by the staff and approved in appropriate sections of the SER and SSERs. These technical issues are listed below and the actual exemptions will be incorporated in the operating license:

(1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4)

(2) Criticality monitor (Section 9.1, SSER 5) (TAC 63615)

1.13 Implementation of Corrective Action Programs and Special Programs

On September 17, 1985, the NRC sent a letter to the applicant, pursuant to Title 10 of the Code of Federal Regulations, Section 50.54(f), requesting that the applicant submit information on its plans for correcting problems with the overall management of its nuclear program as well as its plans for correcting plant-specific problems. In response to this letter, TVA prepared a Corporate Nuclear Performance Plan (CNPP) that identified and proposed corrections to problems with the overall management of its nuclear program, and a site-specific plan for Watts Bar entitled, "Watts Bar Nuclear Performance Plan" (WBNPP). The staff reviewed both plans and documented results in two safety evaluation reports, NUREG-1232 Vol. 1 (dated July 1987) and NUREG-1232 Vol. 4 (dated January 1990).

NUREG-1232 Vol. 4 documented the staff's general review of most of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847. In accordance with that commitment, this new section is introduced in SSER 5 and subsequent SSERs. The current status of all CAPs and SPs follows.

1.13.1 Corrective Action Programs

(1) Cable Issues (TAC 71917)

Program review status:	NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by March 1991.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990).

(2) Cable Tray and Tray Supports (TAC R00516)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by July 1991.
NRC inspections:	Inspection Reports 50-390, 391/89-14 (December 18, 1989).

(3) Design Baseline and Verification Program (TAC 63594)

Program review status: Inspection Reports 50-390, 391/89-12 (November 20, 1989); NUREG-1232, Vol. 4.; review in progress.

Implementation status: Full implementation expected by April 1991.

NRC inspections: Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990).

(4) Electrical Conduit and Conduit Support (TAC R00508)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by July 1991.

NRC inspections: Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989).

(5) Electrical Issues (TAC 74502)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: To come.

(6) Equipment Seismic Qualification (TAC 71919)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by August 1991.

NRC inspections: Inspection Report 50-390, 391/90-05 (May 10, 1990).

(7) Fire Protection (TAC 63648)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; fire protection program will be reviewed in a future SSER.

Implementation status: Full implementation expected by May 1991.

NRC inspections: To come.

(8) Hanger and Analysis Update Program (TAC R00512)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by May 1991.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990).

(9) Heat Code Traceability (TAC 71920)

Program review status: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4.

Implementation status: 100% (certified by letter, E. Wallace (TVA) to NRC, July 31, 1990); staff concurrence to come later.

NRC inspections: Inspection Reports 50-390, 391/90-02 (March 15, 1990); 50-390, 391/89-09 (September 20, 1989).

(10) Heating, Ventilation, and Air-Conditioning Duct and Duct Supports (TAC R00510)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by July 1991.

NRC inspections: Inspection Report 50-390, 391/90-05 (May 10, 1990).

(11) Instrument Lines (TAC 71918)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: Inspection Report 50-390, 391/90-14 (August 3, 1990).

(12) Prestart Test Program (TAC 71924)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; completed in NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by September 1991.

NRC inspections: Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-12 (June 19, 1990); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-17 (August 14, 1990); 50-390, 391/90-20 (September 25, 1990).

(13) Quality Assurance Records (TAC 71923)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by July 1991.
NRC inspections:	Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990).

(14) Q-List (TAC 63590)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by March 1991.
NRC inspections:	Inspection Report 50-390, 391/90-08 (September 13, 1990).

(15) Replacement Items Program (TAC 71922)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by August 1991.
NRC inspections:	To come.

(16) Seismic Analysis (TAC R00514)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by July 1991.
NRC inspections:	Inspection Report 50-390, 391/89-21 (May 10, 1990).

(17) Vendor Information Program (TAC 71921)

Program review status:	Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (the safety evaluation is reproduced as Appendix I in this SSER).
Implementation status:	Full implementation expected by April 1991.
NRC inspections:	To come.

(18) Welding (TAC 72106)

Program review status: Inspection Report 50-390, 391/90-04 (May 17, 1990); NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by January 1991.

NRC inspections: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990).

1.13.2 Special Programs

(1) Concrete Quality (TAC 63596)

Program review status: NUREG-1232, Vol. 4.

Implementation status: Full implementation certified by letter, E. Wallace to NRC, August 31, 1990; staff concurrence to come later.

NRC inspections: NUREG-1232, Vol. 4; Inspection Report 50-390, 391/89-200 (December 12, 1989).

(2) Containment Cooling (TAC 77284)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by February 1991.

NRC inspections: To come.

(3) Detailed Control Room Design Review (TAC 63655)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: To come.

(4) Environmental Qualification Program (TAC 63591)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: To come.

(5) Master Fuse List (TAC 76973)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by November 1990.

NRC inspections: To come.

(6) Mechanical Equipment Qualification (TAC 76974)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: To come.

(7) Microbiologically Induced Corrosion (TAC 63650)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by June 1991.

NRC inspections: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990).

(8) Moderate Energy Line Break Flooding (TAC 63595)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by June 1991.

NRC inspections: To come.

(9) Radiation Monitoring Program (TAC 76975)

Program review status: NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by June 1991.

NRC inspections: To come.

(10) Soil Liquefaction (TAC 77548)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by March 1991.

NRC inspections: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990).

(11) Use-as-Is CAQs (TAC 77549)

Program review status: NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by October 1990.

NRC inspections: Inspection Report 50-390, 391/90-19 (October 15, 1990).

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

3.2 Classification of Structures, Systems, and Components

In Sections 3.2.1 and 3.2.2 of Supplement No. 3 to the SER, the staff found that the seismic classification of the emergency raw cooling water system (ERCWS) was acceptable pending verification that the applicant made certain modifications to the ERCWS.

The staff has verified that these portions of the ERCWS have been upgraded or replaced satisfactorily (see Inspection Report 50-390/84-37, dated July 13, 1984), and therefore, considers Confirmatory Issues 5 and 6 resolved.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.3 Deterministic "Leak-Before-Break" Evaluation to Eliminate Postulated Breaks as a Design Basis for High-Energy Piping

By letter dated April 17, 1989, the applicant submitted a request for the elimination of the dynamic effects of postulated primary loop pipe ruptures from the design basis using "leak-before-break" (LBB) technology as permitted by General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50. The staff has reviewed the request and has concluded that the applicant is in compliance with GDC 4 for Watts Bar. By letter dated May 17, 1989 (S. Black, NRC, to O. D. Kingsley, TVA), the staff transmitted the review results. That evaluation is reproduced as Appendix J in the SSER.

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

The staff issued Generic Letter 89-04 addressing acceptable inservice testing (IST) programs. By letter dated August 21, 1989, the applicant committed to submit, six months before the expected date of operating license issuance, a revised American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Inservice Pump and Valve Test Program." The applicant stated that this program will comply with requirements in 10 CFR 50.55a(g)(4)(i) and the guidance in Generic Letter 89-04.

The staff will report results of its review in a future supplement to the Watts Bar Safety Evaluation Report. Proposed License Condition 2 remains unresolved.

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

SSER 1 and SSER 3 described a number of generic and specific concerns. The staff has updated the status of several of these issues in Inspection Report 50-390, 391/90-05 (May 10, 1990), and plans to provide a summary in a future SSER when all concerns are resolved. Outstanding issue 4(a) remains unresolved.

4 REACTOR

4.4 Thermal-Hydraulic Design

Flow Measurement Uncertainties

By letter dated June 19, 1984, the applicant proposed certain changes to the reactor coolant system (RCS) total flow rate and flow measurement uncertainty from 403,600 gpm ±3.5 percent to 396,000 gpm ±1.5 percent, respectively. By letter dated August 30, 1984, the applicant provided a new measurement uncertainty analysis yielding a value of ±1.8 percent which included a venturi fouling uncertainty of ±0.1 percent.

The staff reviewed the flow measurement uncertainty analysis which includes Rosemount resistance temperature detectors and compared it with the Westinghouse generic flow measurement analysis. The generic Westinghouse flow measurement analysis has a value of ±1.5 percent for the calorimetric uncertainty in a four-loop plant. When using normalized elbow taps (one elbow tap per loop), the total uncertainty is given as ±2.1 percent. The comparable analysis submitted for the Watts Bar plant yields a value of ±1.5 percent for the calorimetric uncertainty and total uncertainty of ±1.7 percent, which includes the normalized elbow taps (one elbow tap per loop). A conference call was held between TVA, Westinghouse Electric Corporation, and NRC on February 5, 1985, to clarify the results of the analysis. It was found that several of the parameters that are used to arrive at the measurement uncertainty for the elbow tap flow readings (such as sensor temperature and pressure effects) have a negligible effect on accuracy. Therefore, the Watts Bar flow measurement uncertainties are less than those in the generic Westinghouse analysis. The staff noted that the Watts Bar flow measurement analysis values are identical to those of a similar analysis made for the William B. McGuire plant, and therefore accepts the flow measurement uncertainty value of ±1.7%, which, with an additional 0.1% penalty required to account for venturi fouling brings the value to ±1.8%. The staff is currently developing the Watts Bar Technical Specifications, and will report the final approved flow measurement uncertainty, if different from the value here, in a future SER supplement. This issue was tracked under TAC 63628.

4.4.5 Loose-Parts Monitoring System

In SSER 3, the staff stated that all issues related to the loose-parts monitoring system (LPMS) are resolved, thus closing Confirmatory Issue 13. However, the staff stated that 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.

By letter dated September 19, 1990, the applicant committed to provide such information. The staff will track this commitment by licensing action TAC 77177. Hence, there is no more need for proposed License Condition 42, which is now considered resolved.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The staff has partially completed its review of TVA's application to replace the resistance temperature detector (RTD) bypass system with a new design. The staff issued the results of its review in a letter (S. C. Black, NRC, to O. D. Kingsley, Jr., TVA, dated June 13, 1989), stating that the Eagle-21 microprocessor system is acceptable for monitoring reactor coolant temperature. That document is hereby incorporated by reference. In addition, the applicant has incorporated the information for the approved new design in Final Safety Analysis Report (FSAR) Amendment 63, Sections 5.1, 5.3.2, 5.4, 7.1.3, 7.2, and 15.2. The staff will report results of the FSAR review in a future SSER. Remaining actions will be tracked by TAC 63599.

5.4 Component and Subsystem Design

5.4.3 Residual Heat Removal System

In the SER, the staff stated that the applicant committed to install a residual heat removal (RHR) flow alarm that will alert the operator to initiate alternate cooling modes in the event RHR pump suction is lost. The staff found this commitment acceptable pending verification that the alarm was installed.

The staff has verified that this alarm has been installed and that the alarm will annunciate RHR low flow conditions in the control room (see Inspection Report 50-390/84-28, dated May 11, 1984). Therefore, the staff considers Confirmatory Issue 14 resolved.

5.4.5 Reactor Coolant System (TMI Item II.B.1)

In the SER and Supplement No. 2, the staff found that the applicant's commitments regarding the installation of the reactor coolant system (RCS) vent system were acceptable pending verification that the system was installed.

The staff has verified that the RCS vent system has been installed (see Inspection Report 50-390/84-37, dated July 13, 1984) and, therefore, concludes that License Condition 5 is no longer required.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

Weight of Ice in the Ice Condenser

In the SER, the staff described the long-term response of the Watts Bar primary containment to a postulated loss-of-coolant accident (LOCA). The controlling reactor coolant system pipe break accident (design-basis LOCA) was identified as the double-ended rupture of the reactor coolant system pump suction cold leg. The peak calculated containment pressure for this accident was reported to be 12.3 psig, which is less than the containment design pressure of 15.0 psig. The staff also described the other aspects and its review of the long-term LOCA analysis and concluded that the analysis was acceptable.

By letter dated February 15, 1985, the applicant submitted a revised long-term containment analysis for the design-basis LOCA in support of a proposed reduction in the draft Technical Specifications limit for the minimum allowable weight of the ice in the ice condenser. The significant changes in the revised analysis, when compared with the original analysis, include (1) a reduced initial ice inventory and (2) changes to the mass and energy release model and the containment heat sink model. The staff's review of the assumptions and results of the revised analysis indicates that the analysis is consistent with a similar reanalysis of the Sequoyah plant in 1981, which the staff found acceptable (Amendment 4 to the Sequoyah Unit 1 license, DPR-77, issued on March 6, 1981). The Sequoyah analysis assumed an ice weight of 2.10×10^6 lb, whereas the Watts Bars analysis assumes 2.125×10^6 lb (both reduced from an original assumption of 2.45×10^6 lb of ice). Because of the similarity between the Watts Bar and Sequoyah analyses, the staff finds the method of analysis, modeling assumptions, and results of the Watts Bar reanalysis acceptable.

As discussed in the Watts Bar SER, the original analysis indicates a containment peak pressure of 12.3 psig occurring at about 1 hour and 50 minutes after the onset of the accident. The revised analysis shows a lower containment peak pressure (11.21 psig) occurring at 1 hour after onset of the accident. The calculated peak containment pressure of 11.21 psig is less than the containment design pressure of 15.0 psig and is about 10 percent lower than the peak pressure calculated in the original analysis; therefore, the staff finds the applicant's revised long-term containment response analysis for a LOCA acceptable.

The revised analysis is based on 2.125×10^6 lb of ice initially being in the ice condenser, instead of 2.45×10^6 lb, which was assumed in the original analysis. Therefore, the staff concludes that the lower ice weight may be used as the basis for establishing the technical specification limit for the minimum allowable weight of ice in the ice condenser. This review was tracked by licensing action TAC 63620.

Containment Instrumentation

In the SER, the staff stated that it will require the applicant to install and have operational acceptable containment pressure monitors before the full-power license is issued.

The staff has verified that the applicant has installed these monitors and that there is redundant, continuous, containment pressure indication (0-60 psig) in the control room (see Inspection Report 50-390/84-59, dated November 7, 1984). The staff is reviewing the qualification of this equipment and will address it in its evaluation of the Watts Bar Equipment Qualification Program (Outstanding Issue 4). Therefore, the staff concludes that proposed License Condition 6d is resolved.

In addition, the staff stated that it would require the applicant to install and have operational acceptable containment water level monitors before the full-power license is issued. The staff has verified that the applicant has installed these monitors (see Inspection Reports 50-390/84-77, dated November 14, 1984, and 50-390/84-85, dated January 8, 1985). The staff is reviewing the qualification of this equipment and will address it in its evaluation of the Watts Bar Equipment Qualification Program (Outstanding Issue 4) in a supplement to the SER. Therefore, the staff concludes that proposed License Condition 6e is resolved.

6.2.4 Containment Isolation System

The SER stated that the containment isolation provisions for the main and auxiliary feedwater lines, feedwater bypass lines, and the chemical feedlines to the steam generators did not meet the requirements of General Design Criterion (GDC) 57. The SER further stated that, if certain valves already in place were designated as containment isolation valves, the requirements of GDC 57 would be satisfied for all except the chemical feedlines, for which a safety-grade isolation valve of an appropriate type (such as automatic, remote manual, or locked-closed manual) would have to be installed in the safety-grade portion of each line.

By Amendment 55 to the FSAR, the applicant has provided the additional information for resolving this issue. In FSAR Table 6.2.4-1, the applicant has designated appropriate in-place valves as containment isolation valves for the main and auxiliary feedwater lines and the feedwater bypass lines. The chemical feedlines have been modified by removing them from their original connection points to the feedwater system, eliminating some connections, and consolidating the rest into the new steam generator wet layup system. The wet layup lines connect to feedwater system lines outside the feedwater system's containment isolation valves so that separate containment isolation valves are not needed for the wet layup lines, with the exception of four wet layup lines that connect inside the containment isolation valves in the four main feedwater lines. These wet layup lines are appropriately equipped with locked-closed manual containment isolation valves, which satisfies the requirements of GDC 57.

Therefore, the staff concludes that the design of the containment isolation provisions for the main and auxiliary feedwater lines, feedwater bypass lines, and the chemical feedlines (wet layup lines) is acceptable and meets the requirements of GDC 1, 2, 4, 16, 54, and 57. This resolves Confirmatory Issue 18 and eliminates the need for proposed License Condition 7.

In addition, Supplement No. 3 to the SER stated that operability of the containment purge/vent isolation valves during LOCA-induced pressure transients inside containment was still being reviewed by the staff. That review was completed and has been issued (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated July 12, 1990). The staff concluded that the isolation valves can close against the buildup of pressure in containment in the event of a design-basis accident if the lower containment isolation valves are physically blocked to an opening angle of 50° or less. The requirement will be reflected in the Technical Specifications. Thus, proposed License Condition 8 is considered resolved.

6.2.5 Combustible Gas Control Systems

In the SER, the staff stated that it would require the applicant to install and have operational qualified containment hydrogen monitors before the full-power license is issued. The staff has verified that the applicant has installed these monitors (see Inspection Report 50-390/84-85, dated January 8, 1985). The staff is reviewing the qualification of this equipment and will address it in the staff's evaluation of the Watts Bar Equipment Qualification Program (Outstanding Issue 4). Therefore, the staff concludes that proposed License Condition 6f is resolved.

6.2.6 Containment Leakage Testing

In the SER, the staff stated that the applicant would limit the total potential leakage, which could bypass the emergency gas treatment system (serving the annulus) and be treated instead by the auxiliary building gas treatment system, to 10 percent of the containment design-basis leakage rate. By Amendment 49 to the FSAR, the applicant revised the bypass leakage rate upward to 25 percent of the containment design-basis leakage rate. On the basis of its review (presented in Section 15.4.1 of this supplement) and finding that there is no radiological consequence associated with the change, the staff finds this change acceptable. This review was tracked under TAC 63624.

6.3 Emergency Core Cooling System

6.3.2 Evaluation

In the SER, the staff stated that the applicant will lock out power from certain valves in the emergency core cooling system (ECCS) whose misalignment might affect ECCS effectiveness. Some of these valves would be required to operate following a LOCA, and the manual restoration of power would add to post-accident operational complexity. By letters dated September 15, 1982, and April 10, 1985, the applicant stated Watts Bar would use modified control circuits for these valves to ensure that no single failure would be able to energize the opening or closing coils of the valve operators. The design uses redundant contacts that are wired before and after each opening and closing coil. In addition, clear protective covers will be attached to the main control board over each respective control switch to prevent inadvertent actuation. As discussed in SER Sections 7.6.4 and 8.3.1.8, the staff found this design acceptable. Accordingly, power will not be locked out from the following valves during operation:

(1) hot-leg injection line valves

(2) valves from residual heat removal (RHR) discharge to safety injection (SI) and charging pump suction

(3) RHR suction valves from containment sump

(4) RHR discharge valves

(5) SI pump suction valve from refueling water storage tank

(6) SI miniflow valve

In addition, the applicant evaluated other valves that may be used for SI miniflow, RHR to SI cross-connect, and SI injection, but for which the consequences of single failure would be acceptable. Power will also not be locked out from these valves. This revision is acceptable to the staff. This review was tracked under TAC 63630.

The applicant's response to Issue 4 of NUREG-0138 is also discussed in SER Section 6.3.2. This issue involves the resequencing of ECCS loads following SI signal reset followed by a loss of offsite power. The SER states that the applicant will delay SI reset by at least 10 minutes after an actuation. The time delay was designed to improve the likelihood that the plant would be in a safe and stable condition before reset. The 10-minute requirement is superseded by the SI reset criteria in the Emergency Response Guidelines designed to ensure that the plant is in a safe and stable condition before SI is reset. This modification is acceptable to the staff. This review was tracked under TAC 63630.

6.4 Control Room Habitability

In an April 26, 1985, letter, the applicant proposed to remove the main control room air intake chlorine detectors. By letter dated May 15, 1985, TVA supplied additional information. The Technical Specifications will appropriately reflect this change. Sodium hypochlorite is used for water treatment instead of chlorine, and thus only a small quantity of chlorine will be stored on site for laboratory use. Negligible amounts of chlorine will be stored off site in close proximity to the plant. In addition, review of the number of chlorine shipments past the Watts Bar plant site indicates the number is sufficiently low that a toxic gas consequence analysis for chlorine shipment accidents need not be performed. The staff concludes that the removal of the chlorine detectors will not impair the safe operation of the plant.

The staff concludes that the Watts Bar control room habitability system meets the acceptance criteria of SRP Section 6.4. This conclusion is based on the staff's review of the applicant's analysis and the staff's evaluation using the procedures described in Regulatory Guide 1.78. Therefore, the staff concludes that this modification is acceptable. This review was tracked under TAC 63635.

6.5 Engineered Safety Feature (ESF) Filter Systems

6.5.1 ESF Atmosphere Cleanup System

6.5.1.3 Reactor Building Purge Ventilation System (RBPVS)

In FSAR Amendment 54, the applicant revised the description of the environmental consequences of a postulated fuel handling accident. Because no relative humidity control is provided, the assigned filter efficiency is 90 percent for elemental iodine and 30 percent for organic iodide. The staff finds this acceptable and in its evaluation has assigned the system decontamination efficiencies of 90 percent for elemental iodine and 30 percent for organic iodide. On the basis of this evaluation, the staff finds that the RBPVS is designed to control the releases of radioactive materials in gaseous effluents in accordance with applicable guidelines of following a postulated design-basis accident, and is, therefore, acceptable. This review was tracked under TAC 63636.

7 INSTRUMENTATION AND CONTROLS

7.8 NUREG-0737 Items

7.8.1 Relief and Safety Valve Position Indication (TMI Item II.D.3)

In the SER, the staff stated that it found the applicant's design for providing a positive indication of the position of the reactor coolant system relief and safety valves to be acceptable pending confirmation of the installation of the acoustic monitoring system.

The staff has verified that the acoustic monitors have been installed; these provide indication in the control room (see Inspection Report 50-390/84-35, dated June 21, 1984). Therefore, the staff concludes that proposed License Condition 11 is no longer required.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

Criticality Monitor

By letters dated July 27, 1983; June 19, 1984; and January 30 and February 16, 1985, the applicant requested relief from the requirement of 10 CFR 70.24, to have a criticality monitor installed in the fuel storage area, until irradiated fuel is placed into that area. Only nonirradiated fuel is currently on site. The applicant would be required to maintain a criticality monitor for the nonirradiated fuel without the requested exemption.

Because of the inherent features associated with the storage and inspection of nonirradiated fuel, the staff determined that good cause was shown for granting this exemption in Special Nuclear Material (SNM) License Number SNM-1861 (September 5, 1979). The SNM license and its exemption expire once the operating license is issued, hence the applicant has asked that the exemption be incorporated into the Watts Bar Unit 1 operating license once it is issued. The staff has determined that the bases supporting this exemption in the SNM license are applicable to the operating license. Therefore, an exemption from this requirement (10 CFR 70.24) is justified and will be granted, in accordance with Commission regulations, with the operating license.

9.1.2 Spent Fuel Storage

By letter dated February 16, 1985, the applicant confirmed that it has executed a contract (DE-CR01-83-NE-44420) with the Department of Energy (DOE) whereby DOE agreed to accept spent nuclear fuel from Watts Bar Nuclear Plant. The staff acknowledges receipt of this confirmation and has no more concerns in this issue.

9.2 Water Systems

9.2.2 Component Cooling System (Reactor Auxiliaries Cooling Water System)

In the SER, the staff stated that the applicant committed to relocate the component cooling booster pumps above the probable maximum flood (PMF) level (elevation 738.1 feet). The staff found this commitment acceptable, pending verification that the modifications were completed by fuel loading time.

The staff has verified that these pumps have been relocated above the PMF level (see Inspection Report 50-390/84-20, dated April 12, 1984), and therefore considers Confirmatory Issue 37 resolved.

9.3 Process Auxiliaries

9.3.2 Process Sampling System

Postaccident Sampling Capability (TMI Item II.B.3)

In SSER 3, the staff found the applicant's postaccident sampling system met all of the 11 criteria of Item II.B.3 of NUREG-0737. However, the staff stated that before restart following the first refueling outage, the applicant will be required to provide a final procedure for estimating the degree of core damage. However, in light of the five-year delay in licensing, the applicant should commit to submit this procedure at an earlier date. License Condition 19 remains unresolved.

9.5 Other Auxiliary Systems

9.5.2 Communication Systems

In the SER, the staff stated it would require the communication systems to be preoperationally tested in accordance with Regulatory Guide (RG) 1.68 to demonstrate that the systems will function properly and will provide adequate communication with the maximum potential background noise levels.

By letter dated March 18, 1985, the applicant justified not performing the functional preoperational test as specified in RG 1.68. The applicant stated that all the communication systems in the safety-related areas had been either (1) preoperationally tested, (2) functionally tested during yearly emergency drills and/or performance of startup tests, (3) used routinely over the past several years, and/or (4) successfully tested at the Sequoyah Nuclear Plant for certain specified areas. The staff has evaluated the information submitted and finds it provides an acceptable means of meeting the functional preoperational testing requirements of the communication systems. Therefore, the staff concludes that the requirement on performance testing has been met.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the installed communication systems to the acceptance criteria in Section II of SRP Section 9.5.2. Other bases for acceptance were conformance to industry standards and the ability of the systems to provide effective communications from diverse means within the Watts Bar plant during normal and emergency conditions under maximum potential noise levels.

On the basis of its review, the staff concludes that the communication systems are in conformance with the above-cited standards, criteria, and design bases; can perform their design functions; and are, therefore, acceptable. The staff considers proposed License Condition 21 resolved.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.1 Emergency Diesel Engine Auxiliary Support Systems (General)

Quality Classification of Diesel Generator Auxiliary System Piping and Components

The following discussion is also applicable to SER Sections 9.5.4.2, 9.5.5, 9.5.6, 9.5.7, and 9.5.8. In these sections, the staff stated that the quality standards to which the diesel generator skid-mounted auxiliary system piping and components were designed were unacceptable. The staff required that the piping be designed to meet ASME Section III Class 3 requirements, and the recommendation of RG 1.26.

By letter dated February 15, 1985, the applicant provided the standards to which the engine skid-mounted auxiliary systems (fuel oil, cooling water, air starting, lubrication, and combustion air intake and exhaust) piping and associated components were designed. This engine-mounted piping and the associated components, such as valves, fabricated headers, and fabricated special fittings, are designed, manufactured, and inspected in accordance with the guidelines and requirements of American National Standards Institute (ANSI) Standard B31.1, "Code for Pressure Piping"; ANSI Standard N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR Part 50, Appendix B. The engine skid-mounted auxiliary system piping and associated components are intentionally overdesigned (subjected to low working stresses) for the application, thereby resulting in high operational reliability. The applicant also provided a comparison of their design with the requirements of ASME Code, Section III, Class 3. The results of the comparison indicate they differ from ASME Code, Section III, Class 3, in three areas as follows:

(1) The ASME Code requires that American Society for Testing and Materials (ASTM) materials be used and mill test reports be provided for the piping. The applicant stated that mill test reports were provided and the piping was marked in accordance with the ASTM material specification. The staff finds this acceptable.

(2) The ASME Code requires liquid penetrant examination for welds that exceed 4-inch iron pipe size. The applicant stated that only a few welds in cooling water system piping 4 inches and over were not liquid penetrant examined; in those few cases, welds were only visually examined with the system at design pressure and temperature for acceptability of weld. The acceptability of this item is discussed below.

(3) The ASME Code requires a hydrostatic test to 125 percent of the design pressure. The applicant stated that some piping and components were hydrostatically tested to 150 percent of design pressure and that the rest of the piping would be leak tested at operating pressure during engine operation. The acceptability of this item is discussed below.

In lieu of performing liquid penetrant examination of all piping 4 inches and over and the hydrostatic tests on all piping, the staff required that all diesel engine auxiliary system piping be hydrostatically tested to a minimum of 125 percent of design pressure. Because of the low working stresses, the hydrostatic tests would provide adequate assurance of piping leaktightness and weld integrity.

In a letter dated March 18, 1985, the applicant sent the staff information to show that the design of the fuel oil storage and transfer system did not easily lend itself to a hydrostatic test per RG 1.137, Position C.1.e(1). In lieu of the 10-year hydrostatic test, the applicant proposed to visually inspect for leaks (every 18 months) all of the exposed fuel oil piping while the diesel was running. The staff finds this acceptable.

By letters dated April 18 and August 30, 1985, the applicant provided additional information concerning the remainder of the auxiliary system piping. Morrison-

The auxiliary system piping is configured in such a way that there are virtually no means of isolation between the auxiliary systems and engine block. The system would require complete disassembly, and, in effect, the test would become a series of piping assembly pressure tests rather than a system test. The vast majority of piping connections are threaded, and experience has been that threaded joints are subject to weepage when subjected to repeated disassembly. The diesel generators at the Watts Bar facility have had a successful operating history over the past 5 to 7 years in that there has never been a failure caused by mechanical malfunction of the engine-mounted auxiliary systems. This successful operational experience has ensured auxiliary system piping leak-tightness and joint integrity.

As further assurance of maintaining leaktightness integrity, the applicant has operating procedures in place that require periodic checks of the diesel generator auxiliary systems during standby and during operation. In the event leakage is observed, the applicant will take immediate remedial action to correct the problem.

On the basis of the current condition and proven reliability of the Watts Bar emergency diesel generators, the applicant concluded that implementation of the pressure test requirement will result in a potential degradation of diesel generator reliability and will not result in an increase in the diesel generator system's margin of safety.

The staff concurs with the applicant's conclusion and, therefore, considers it unnecessary to hydraulically test the standby diesel generator engine-mounted auxiliary systems to a minimum of 125 percent of design pressure.

On the basis of its review, the staff concludes that the engine-mounted piping and components of emergency diesel engine auxiliary systems (cooling water, air starting, lubrication, and combustion air intake and exhaust systems) and the diesel generator fuel oil system meet the requirements of GDC 2, 4, 5, and 17; meet the guidance of the cited regulatory guides and standard review plans; can perform their design safety function; meet the recommendations of NUREG/CR-0660 and industry codes and standards; and are acceptable. Therefore, the staff considers Outstanding Issue 13 resolved.

In the SER, the staff stated that it would condition the license to ensure completion of modifications to the diesel generator auxiliary support systems to comply with NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," by the first refueling outage. In addition, the staff stated that it would verify the incorporation of certain requirements into the plant operating procedures to ensure that operation at no-load and low-load conditions will not harm the diesel generators.

The staff has verified that the applicant has completed these modifications to the diesel generators and has revised the operating procedures (see Inspection Reports 50-390/84-90, dated February 11, 1985, and 50-390/85-08, dated March 29, 1985). Therefore, the staff considers Confirmatory Issues 39 and 40 acceptably resolved and no longer considers proposed License Condition 22 necessary.

Reports 50-390/84-90, dated February 11, 1985, and 50-390/85-08, dated March 29, 1985). Therefore, the staff considers Confirmatory Issues 39 and 40 acceptably resolved and no longer considers proposed License Condition 22 necessary.

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System

In the SER, the staff stated that the design of the emergency diesel engine fuel oil storage and transfer system did not conform to ANSI Standard N195, "Fuel Oil System for Diesel Generators," and the fuel oil quality and tests and guidelines of RG 1.137, "Fuel Oil Systems for Standby Diesel Generators," Positions C.2.a through C.2.g.

In letters dated March 17, 1982, and March 18, 1985, the applicant stated that by fuel loading time it will be in compliance with ANSI Standard N195 and RG 1.137, Position C.2, except for the following sections of ANSI Standard N195:

(1) Sections 7 and 9 with regard to piping system design, quality group classification, and system testing. This item is discussed in Section 9.5.5.

(2) Section 8 with regard to lack of differential pressure indicators on the fuel oil strainers and fuel oil tank level indicators. The staff has reviewed this deviation and finds it acceptable.

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

In the SER, the staff stated that the applicant had committed to provide missile protection for the fuel oil storage tank vent lines. The staff found this commitment acceptable pending verification that the modifications were completed before fuel loading time.

The staff has verified that the missile protection has been installed for the vent lines supporting the four diesel generators required for operation of the facility (see Inspection Report 50-390/84-28, dated May 11, 1984). Therefore, the staff considers Confirmatory Issue 36 resolved.

9.5.5 Emergency Diesel Engine Cooling Water System

The diesel engine cooling water keep-warm system is described in Section 9.5.5 of the SER. With regard to diesel generator room temperature, the staff raised the following concern.

The diesel generator room supply ventilation air does not appear to be preheated nor does the room seem to be heated in any manner. This plant is located in an area in which the temperature can drop below freezing. Therefore, temperature in the diesel generator room could conceivably approach outside ambient temperature. With the diesel generator room at or below freezing level, the means of preheating the entire engine cooling water volume may not be adequate to keep the engine sufficiently preheated to ensure a successful fast start and load-accepting capability in an emergency. Improper preheating of the diesel engine units may keep them from performing their required safety function and may degrade availability of the diesel generators to an unacceptable level.

In a letter dated December 14, 1982, the applicant provided additional information on the cooling water preheat system. The applicant stated, "Each engine is equipped with a low standby jacket water alarm which will annunciate should the standby jacket water temperature fall below the supplier's recommended value. This method provides notification of an improper temperature condition for each diesel generator unit in sufficient time to perform corrective action."

A review of the FSAR shows that this alarm is located in the cooling water piping between the lube oil cooler and the engine-driven water pumps, and because of its location, the staff has determined that it would not give an adequate indication of the temperature in those portions of the system (engine block and cooling water heat exchanger) that are remote from the natural circulation keep-warm system near the lube oil cooler.

Further, communication with the engine manufacturer, Electromotive Division of General Motors (EMD-GM) (see March 6, 1984, letter to applicant), has shown that satisfactory performance of the cooling water preheat system is based on maintaining a diesel engine room temperature of 65°F or higher. The diesel generator room heating, ventilating, and air conditioning system, evaluated in Section 9.4.5 of the Watts Bar SER, is designed to maintain a minimum room air temperature of approximately 40°F during all operating conditions. Normally, temperature in the diesel generator room is maintained at 65°F or higher. No alarm has been provided to warn the control room operator that the temperature in the diesel generator room is below 65°F. Therefore, the staff recommended that an alarm be installed in the diesel generator room before fuel load, that annunciates in the control room when the diesel generator room temperature drops below 65°F. This would provide enough time to take corrective actions before engine cooling water temperature drops to unacceptable levels.

By letters dated October 19, 1984, and March 18, 1985, and during meetings on February 28 and April 2, 1985, the applicant provided information for justifying a room temperature limitation of 40°F instead of 65°F. The staff evaluated the information and found the surveillance program of monitoring the temperature on a 12-hour cycle, and the commitment to take immediate remedial action to restore the room temperature acceptable. However, the staff did not agree with the applicant that the data provided justified changing the temperature limit from 65°F to 40°F.

By letters dated March 17, March 18, and April 17, 1985, the applicant provided additional information consisting of diesel generator operating test data (for the winter months) over a 2-year period at the Sequoyah Nuclear Plant (located in the same general area as the Watts Bar plant and employing similar diesel generators manufactured by EMD-GM). The data demonstrated the capability of these diesel engines to start at 40°F ambient or lower temperatures within 10 seconds and assume load as designed. In addition, the applicant provided the following response from the diesel generator assembler, Morrison-Knudsen Company, Inc., dated April 15, 1985. "In our best engineering judgment and based on our worldwide sales and units operating in various climatic conditions, the diesel generators operating at Watts Bar will start and come up to speed in 10 seconds with the temperature of the engine room at 40°F."

The staff finds the submitted information adequate justification for lowering the ambient room temperature limit to 40°F and finds the design of the system acceptable.

In addition to addressing the above problem in the December 14, 1982, letter, the applicant also provided information on two other open items in the cooling water system: (1) the time period the diesel generator is capable of operating fully loaded without secondary cooling and (2) the possibility of the cooling water system becoming air bound as a result of the expansion tank location.

The applicant stated that there is enough water in the cooling water system to allow operation of the diesel engine for 1.875 minutes without secondary cooling, which would be restored within 25 seconds after the diesel start signal. The applicant also showed that the system vented to the expansion tank under all modes of operation. The staff found these items acceptable.

On the basis of its review, the staff concludes that the emergency diesel engine cooling water system meets the requirements of GDC 2, 4, 5, 17, 44, 45, and 46; conforms to the acceptance criteria in SRP Section 9.5.5; and meets the recommendations of NUREG/CR-0660 and industry codes and standards. It can perform its design safety function and is, therefore, acceptable.

9.5.7 Emergency Diesel Engine Lubricating Oil System

In the SER, the staff stated that the applicant had been asked to describe the features that protect the diesel engine crankcase from exploding. In a letter dated March 18, 1985, the applicant described the crankcase overpressure detection trip/alarm and justified how this meets the guidelines of RG 1.9, Institute of Electrical and Electronics Engineers (IEEE) Standard 387-1977, and SRP Section 9.5.7. The applicant stated that, because of the configuration of the diesel generators, a crankcase explosion in one diesel generator set cannot affect the operability of the other diesel generator sets. The design of the facility allows the plant to be safely shut down under all design-basis events assuming the loss of one of the four diesel generator sets.

In addition, the applicant stated that, if a loss-of-offsite-power condition developed and one of the diesels developed a problem that resulted in a crankcase explosion rendering the diesel generator inoperable, remedial measures could be taken in a shorter time period than would be required to repair the damage to the diesel internals (5 days). A skid-mounted diesel generator set can be hooked up in about 3 days. A temporary power line could be run from the Watts Bar hydrogenerating station in about 2 days. The applicant concluded that the diesel generator's system design and separation are the protective measures that prevent unacceptable crankcase explosions and mitigate the consequences of such an event.

On the basis of its review, the staff concludes that the emergency diesel engine lubricating oil system meets the requirements of GCD 2, 4, 5, and 17; industry codes and standards; the guidance of the cited regulatory guides and SRP Section 9.5.7; and the recommendations of NUREG/CR-0660. It can perform its design safety function and is, therefore, acceptable. This resolves Outstanding Issue 14.

10 STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

In letters dated March 7 and June 19, 1984, and March 25, 1985, and at meetings in NRC offices on January 22, February 6, and February 28, 1985, the applicant requested the deletion of Standard Technical Specification (STS) 3/4.3.4, "Turbine Overspeed Protection," for Watts Bar Units 1 and 2. As a result of these meetings and discussions with the applicant, the applicant, in letters dated March 25 and April 9, 1985, modified its request so that the surveillance requirements for the turbine integrity program with turbine overspeed protection (TIPTOP) would become part of the administrative controls (Section 6.8.5) of the Technical Specifications and the turbine valve testing frequency would change from once a week to once a month.

The staff's current position, which requires weekly testing of turbine valves as stated in Standard Review Plan (SRP) Section 10.2 (NUREG-0800), was established after extensive discussions with major steam turbine manufacturers and is based largely on engineering judgment and the recommendations of these manufacturers.

Westinghouse, in a meeting on March 23, 1983, with the staff, presented results of an ongoing study on the generation of turbine missiles being conducted on behalf of some licensees and applicants. This study specifically included consideration of the testing requirements for the turbine overspeed protection valves and turbine valve arrangement of the type installed at Watts Bar. This study noted that turbine valve operability and reliability will not be significantly affected by increasing the periodic valve testing interval from the present weekly schedule to a much longer interval. In Westinghouse's judgment, lack of a significant number of valve failures, good operating experience, and a well-planned turbine valve maintenance and inspection program provide reasonable bases for increasing the periodic test interval for turbines with valve arrangements as installed at Watts Bar from weekly to monthly. Westinghouse has made a formal recommendation to its customers who have turbines employing turbine valves and steam chest arrangements of the type installed at Watts Bar to change from weekly to monthly valve testing. The monthly valve testing frequency is further supported by Westinghouse Topical Report WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," which was approved by the staff in a safety evaluation dated February 7, 1989.

The staff has evaluated the information submitted by the applicant in the March 25 and April 9, 1985, letters and Westinghouse information presented at the March 23, 1983, meeting. Considering the information presented by the applicant and Westinghouse and the staff's original basis for the STS, the staff concludes that the interval between periodic turbine valve testing can be increased for Watts Bar from weekly to monthly without significantly affecting the capability of the turbine valves to function on demand.

In summary, the basis for considering Technical Specifications relief at Watts Bar was:

(1) Provision of a satisfactory statistical basis to determine frequency of turbine valve testing. Up to now, test frequency of these valves has been largely based on experience with turbine generators installed in fossil-fuel plants. The Westinghouse turbine-missile study (Westinghouse, March 1974) provides a basis for Westinghouse to establish turbine valve test frequency on a monthly basis for all nuclear units having a steam chest and valve arrangement similar to that at Watts Bar.

(2) The applicant's maintenance, inspection, and turbine valve test program described in the FSAR and the proposed modification to the Technical Specifications appear to be satisfactory. This program performed on a periodic basis coupled with monthly testing of all turbine valves is satisfactory to the staff.

(3) The data and rationale presented by Westinghouse at the March 23, 1983, meeting and earlier meetings with the staff and the staff's understanding of the data presented to date, including WCAP-11525.

(4) Testing of turbine control valves on base-loaded machines necessitates reducing generator output for several hours. The valve testing sequence during turbine operation requires placing the turbine on manual control and repositioning all turbine control valves in the steam chest to permit individual full-valve stroking. All valves are aligned to equal position. Repositioning the control valves (on a base-loaded machine) results in reduced steam flow to the turbine with a consequent reduction in generator output of about 5 percent. All turbine control valves are tested in a relatively short time (about 35 to 40 minutes). The bulk of the time (approximately 2½ to 3 hours) is consumed in slowly lowering reactor output to permit control valve testing. On completion of valve tests, a similar time period is consumed in slowly increasing reactor power to permit full-load operation of the turbine generator. Reactor output must be changed slowly to minimize xenon spiking. Although this economic impact is not a safety consideration, the staff factored it into its action.

On this basis, the staff concludes the Technical Specifications changes proposed in the April 9, 1985, letter are acceptable. However, the Watts Bar Technical Specifications are being developed according to a new industry format which does not contain the subject requirements. Therefore, the above approved requirements will be incorporated in other plant documents as appropriate. This review was tracked by licensing action TAC 63642.

10.3 Main Steam Supply System

10.3.4 Secondary Water Chemistry

In the SER, the staff found the applicant's secondary water chemistry monitoring and control program met requirements. The staff further stated that the license will be conditioned to require that the program be implemented. The Technical Specifications are being developed for Watts Bar Unit 1 and this program is being included in the administrative section. This will ensure implementation of the program and eliminates the need for a license condition. Thus, proposed License Condition 23 is considered resolved.

10.4 Other Features

10.4.4 Turbine Bypass System

In the SER, the staff stated that it would require the applicant (per the Technical Specifications) to stroke the valves of the turbine bypass system on a periodic basis (at least once a quarter).

As a result of further discussion that took place in open meetings (TAC 76742) with the applicant, the staff has concluded that this periodic stroking of the bypass valves may be performed according to the plant operating procedures, and no technical specification will be necessary to ensure this testing. This review was tracked by licensing action TAC 63643.

11 RADIOACTIVE WASTE MANAGEMENT

11.7 NUREG-0737 Items

11.7.1 Wide-Range Noble Gas, Iodine, and Particulate Effluent Monitors (TMI Items II.F.1(1) and II.F.1(2))

SRP Section 11.5 requires that provisions be made for the instrumented monitoring or sampling and analysis of identified gaseous effluent release paths in the event of postulated accident releases. In a letter dated April 26, 1985, TVA informed NRC that the Unit 2 shield building vent monitor could not be installed by the time fuel was loaded in Unit 1 because of procurement problems. This monitor, a high-range noble gas effluent monitor, is necessary because the release path for the train B gas treatment system of the auxiliary building passes through the shield building stack in Unit 2. The applicant is now committing (letter, E.G. Wallace to NRC, dated October 11, 1990) to have this monitor and its sampler operational before fuel is loaded in Unit 1. The vent monitor for the shield building is described in the applicant's letter to NRC dated August 31, 1990. That letter concerns compliance with Regulatory Guide 1.97, Revision 2; the staff is reviewing it under TAC 77550 and 77551. This commitment eliminates the staff's concern and resolves proposed License Condition 6a.

The original TMI action plan required pressurized-water reactor (PWR) steam safety valve discharge and atmospheric steam dump valve discharge to be monitored by high-range noble gas effluent monitors. This was further clarified by NUREG-0737 Table II.F.1-1 and clarifying note 3 which allowed the monitors to be in the steamline upstream of the valve. In a letter dated November 8, 1983, the applicant requested an exception to the requirement to have the monitors by stating that adequate instrumentation is provided to detect a steam generator tube rupture (SGTR). In a meeting between the NRC staff and the applicant on December 20, 1983, the staff said this was not acceptable. In a letter dated October 11, 1990, the applicant stated that the required high-range noble gas effluent monitors have been installed and will be operational before fuel load. This commitment eliminates the staff's concern and resolves proposed License Condition 6a.

Item II.F.1, Attachment 2, "Noble Gas Effluent Monitor," states that applicants submit, no later than 4 months before the issuance of an operating license, a description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown. RG 1.97 gives numerical criteria for system accuracies and other characteristics of accident-range noble gas effluent monitors.

Item II.F.1, Attachment 2, "Sampling and Analysis of Plant Effluents," states that licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates; the licensees shall provide onsite laboratory capabilities to analyze or measure these samples;

the sampling system design shall be such that plant personnel could remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5 rem whole-body exposure and 75 rem to the extremities during the duration of the accident; and sampling shall be representative in accordance with the criteria of American National Standards Institute (ANSI) Standard N13.1-1969. Complete information has not been provided to give reasonable assurance of compliance with the above requirements. By letter dated April 26, 1985, the applicant committed to have the capability for continuous collection in place (i.e., procedures and any minor system modifications necessary) before exceeding 5-percent power. Operation at or below 5-percent power will not generate sufficient fission products to warrant the need for the capability required to be described by this information. The staff has evaluated this commitment and finds it acceptable. However, since licensing of Watts Bar has been postponed by at least five years, the applicant should submit a revised schedule with an implementation date reflecting the additional time available. Thus proposed License Condition 6b remains unresolved.

In addition, in a letter dated November 22, 1983, the applicant requested an exception to NUREG-0737, Item II.F.1, concerning the installation of high-range noble gas monitors on the auxiliary building vent at Watts Bar. The applicant gave the NRC staff additional information at a meeting in NRC offices on December 20, 1983, and in a submittal dated January 24, 1984.

The Watts Bar Nuclear Plant design does not include a high-range noble gas effluent monitor, as described in NUREG-0737, Item II.F.1, Attachment 1, for the auxiliary building vent because the release is diverted to the shield building vent for design-basis accidents. A low-range to high-range radiation monitor is provided in the shield building ventilation stack.

The secondary containment, which includes the auxiliary building vent isolation dampers, is automatically isolated, and both trains of the ABGTS are automatically placed in service on receipt of any one of five different activating signals, including high radiation from the auxiliary building vent monitor (gaseous activity ranges from 1.1×10^{-7} to 1.7×10^{-1} μCi/cc).

The staff concludes that the auxiliary building vent is not considered to be a potential accident release pathway and, therefore, the Watts Bar Nuclear Plant design, as described above, does not need to be changed to provide for the addition of a high-range noble gas effluent monitor, as described in NUREG-0737, Item II.F.1, Attachment 1, for the auxiliary building vent.

11.7.2 Primary Coolant Outside the Containment (TMI Item III.D.1.1)

By letter dated October 4, 1984, the applicant submitted a justification for excluding the waste gas system from the leak reduction program under NUREG-0737, Item III.D.1.1. The applicant stated that containment isolation will occur in all but two of the Condition 3 and Condition 4 events discussed in Chapter 15 of the FSAR. In one of the events, inadvertent loading of a fuel assembly into an improper position, which is unlikely because of administrative procedures, the resulting power distribution effects will either be readily detected or will cause sufficiently small disruption to be acceptable. The other event, an inadvertent rod cluster control assembly withdrawal, would require a double system failure to occur. The reactor would eventually trip, but the applicant stated it did not believe that a significant amount of fission gases would be released before the reactor tripped. The applicant also stated that the waste gas system would not be used to process highly radioactive gases during an accident.

The staff has evaluated the applicant's submittal and finds that the applicant has not submitted sufficient information to provide assurance that significant quantities of radioactive materials will not enter the waste gas system in the event of an accident. Since the capability exists for the chemical and volume control system to be used during an accident, radioactive materials could be transferred to the waste gas system. Further, there are no special design provisions or procedures identified to provide assurance of minimal leakage from the waste gas system.

Therefore, the following systems are to be included in the Item III.D.1.1 leakage reduction program:

(1) residual heat removal
(2) containment spray
(3) safety injection
(4) chemical and volume control
(5) sampling
(6) waste gas

On this basis, the staff concludes that the leakage reduction program is acceptable. Proposed License Condition 24 will be resolved if the applicant accepts the change stated as item 6 (above).

12 RADIATION PROTECTION

12.7 NUREG-0737 Items

12.7.2 High-Range In-Containment Monitor (TMI Item II.F.1(3))

In the SER, the staff stated that the applicant's commitments regarding the high-range in-containment monitor were satisfactory pending verification that the monitors had been installed.

The staff has verified that these monitors are installed (see Inspection Reports 50-390/84-09, dated February 22, 1984, and 50-390/84-28, dated May 11, 1984) and, therefore, considers License Condition 6c resolved.

14 INITIAL TEST PROGRAM

14.2 Test Program

By FSAR amendment, the applicant modified the Watts Bar Initial Test Program that was discussed in Supplement No. 3, which raised the concerns and questions documented by letters dated March 15 and 26, 1985. The changes were of three general types: editorial (deletions), clarification of commitments, and those that required additional justification. By letters dated February 20, March 27, April 5, May 31, 1985 and the FSAR up to Amendment 55, the applicant provided additional information with appropriate changes to the commitments for the staff to conclude that these concerns have been acceptably resolved. The staff will confirm that these letter commitments and information submittals are incorporated into the FSAR.

Evaluations of specific items of the Watts Bar Initial Test Program follow.

Rod Cluster Control Assembly (RCCA) for Control Rod Bank Reactivity Worth Measurement at Power

By letter dated April 15, 1985, the applicant requested that the test method for Procedure SU-4.3, "RCCA for Control Rod Bank Reactivity Worth Measurement at Power," be modified to include the measurement of control rod movement reactivity changes by changes in boron concentration as well as by the reactivity computer. The justification for the additional method of measuring reactivity change is to improve accuracy. The staff finds adding the boron concentration change method to Procedure SU-4.3 acceptable because it only modifies the test method slightly to improve the accuracy of the results.

Cold No-Flow, Cold Full-Flow, and Hot No-Flow Rod Drop Testing

By letter dated February 13, 1985, the applicant requested a deviation from RG 1.68, Revision 2, which would allow the removal of rod drop timing tests at cold no-flow, cold full-flow, and hot no-flow conditions from the Watts Bar Initial Test Program. In support of this request, the applicant stated that scram capability is only required for the hot full-flow conditions by the Watts Bar Technical Specifications, except when exempted by Special Test Exemption 3.10.4. The staff has reviewed the February 13, 1985, submittal, concurs with its contents, and finds the deviation acceptable on the basis of the following:

(1) There is nothing new or unique in the design of the Watts Bar control rod system.

(2) Scram capability is only required for hot full-flow conditions.

(3) The deviation is consistent with a similar one allowed for the Initial Test Program at the Callaway Nuclear Plant, Unit 1.

Therefore, the staff approves the deviation and has determined that the applicant does not have to perform cold no-flow, cold full-flow, and hot no-flow rod drop testing at the Watts Bar Nuclear Plant, Unit 1.

Noncritical Systems Test Program

In the SER, the staff stated that it would review seven preoperational tests listed under TVA's Noncritical Systems Test Program to determine the adequacy of the tests.

The staff has completed its review of these tests and has determined that they are adequate to demonstrate the functional adequacy of those items covered by the tests (see Inspection Reports 50-390/85-01, dated January 31, 1985, and 50-390/84-58, dated August 22, 1984).

Conclusion

On the basis of the staff's review of the Watt's Bar Initial Test Program through FSAR Amendment 55 and the letters referenced above, the staff concludes that the program meets the acceptance criteria of SRP Section 14.2 and is acceptable to the staff.

15 ACCIDENT ANALYSIS

15.3 Limiting Accidents

15.3.6 Anticipated Transients Without Scram

Status of Salem ATWS Event Issues

On July 8, 1983, the NRC issued Generic Letter (GL) 83-28 as a result of the anticipated-transient-without scram (ATWS) events at Salem Nuclear Generating Station. This letter addressed actions to be taken by licensees and applicants to ensure that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in pressurized-water reactors.

The staff completed its review of the bulk of the applicant's response to GL 83-28 and found the applicant's response acceptable for the following items:

- Item 1.1, Post-Trip Review (Program and Procedure) (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated August 13, 1990)

- Item 1.2, Post-Trip Review (Data and Information Capability) (Inspection Report 50-390, 391/86-04, dated May 28, 1986)

- Item 2.1, Equipment Classification and Vendor Interface (Reactor Trip System Components) (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 18, 1990)

- Item 2.2, Part 1, Equipment Classification Program (letter from S. C. Black, NRC, to O. D. Kingsley, TVA, dated June 1, 1989); Part 2 (letter from F. J. Hebdon, NRC, to O. D. Kingsley, TVA, dated September 7, 1990)

- Items 3.1.1 and 3.1.2, Post-Maintenance Testing of Trip System Components, (Inspection Report 50-390, 391/86-04, dated May 28, 1986)

- Items 3.1.3 and 3.2.3, Post-Maintenance Testing in Technical Specifications That Could Degrade Safety (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated July 2, 1990)

- Items 3.2.1 and 3.2.2, Post-Maintenance Testing of All Other Components (Inspection Report 50-390, 391/86-04, dated May 28, 1986)

- Item 4.1, Trip System Reliability (Vendor-Related Modifications) (Inspection Report 50-390/84-53, dated August 1, 1984)

- Item 4.3, Shunt Attachment to Reactor Trip Breaker (SSER 3 Section 15.3.6, and letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 18, 1990; resolution of this issue eliminated proposed License Condition 40)

- Item 4.5.1, Reactor Trip System Reliability - Functional Testing [memorandum (available in the Public Document Room) from P. S. Tam to F. J. Hebdon, dated October 9, 1990]

- Items 4.5.2 and 4.5.3, Reactor Trip System On-Line Testing (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 28, 1990)

The staff is reviewing the remaining issues of GL 83-28. These will continue to be tracked by TAC 77019 and 77020 (Items 4.2.1 and 4.2.2) and by TAC 77086 and 77087 (Items 4.2.3 and 4.2.4).

15.4 Radiological Consequences of Accidents

15.4.1 Loss-of-Coolant Accident

In Amendment 49 to the FSAR, the applicant revised the leakage pathways for the release of radioactive materials following a loss-of-coolant accident (LOCA). Before FSAR Amendment 49 was submitted, 10 percent of the radioactive materials were released to the auxiliary building and 90 percent to the shield building annulus. Amendment 49 changed these estimated fractions to 25 percent to the auxiliary building and 75 percent to the shield building annulus. Because the engineered safety feature (ESF) filter efficiencies are identical for each of the release pathways, the radiological consequence values remain unchanged from the staff's calculated thyroid and whole-body doses from a hypothetical design-basis LOCA, as presented in Table 15.1 of the SER.

15.4.3 Steam Generator Tube Rupture

Following the steam generator tube rupture (SGTR) at the Ginna Nuclear Power Plant on January 25, 1982, the SGTR subgroup of the Westinghouse Owners Group (WOG) submitted WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," dated December 1984, for NRC staff review, which also references WCAP-10698, Supplement 1, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident." In its evaluation (letter, C. E. Rossi, NRC, to A. E. Ladieu, WOG, dated March 30, 1987) of these Westinghouse documents, the staff concluded that the WOG provided an acceptable and conservative methodology for the generic SGTR analysis, but that five specific and crucial parameters and assumptions used in the analysis may vary significantly from plant to plant, altering the steam generator overfill and radiological dose results. Westinghouse performed the SGTR analysis for Diablo Canyon Nuclear Power Plant, Units 1 and 2, using the LOFTTR2 computer code. The proprietary results of this plant-specific analysis are reported in WCAP-11723. The staff concluded that each member of the SGTR subgroup and all Westinghouse near-term operating licensees were required to submit plant-specific information as follows before use of the methodology from WCAP-10698 could be applied on a plant-specific basis:

(1) Each utility in the SGTR subgroup must confirm that it has in place simulators and training programs that provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design-basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overall prevention, using design-basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

(2) A site-specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1 should be performed using the methodology in SRP Section 15.6.3 (NUREG-0800), as supplemented by the guidance in WCAP-10698, Supplement 1.

(3) The structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill should be evaluated.

(4) Systems, components, and instrumentation credited for accident mitigation in the plant-specific SGTR emergency operating procedures (EOPs) should be listed. Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety-grade systems and components, state whether safety-grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety-grade components can be used for the design-basis event. Provide a list of all radiation monitors that could be used for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured steam generator identification, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

(5) Provide a survey of plant primary and balance-of-plant systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.

The applicant should submit the required information described above. The staff will track all followup actions under TAC 77569.

15.5 NUREG-0737 Items

15.5.3 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (TMI Item II.K.3.1) and Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (TMI Item II.K.3.2)

In the SER, the staff stated it was reviewing Westinghouse Topical Report WCAP-9804 (submitted on March 15, 1981) referenced by the applicant. The staff completed the review and concluded that there is no need for an automatic power-operated relief valve isolation system (see letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 29, 1990). On this basis, proposed License Condition 34 is resolved.

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

Item II.K.3.30 of NUREG-0737 outlines the Commission requirements for the industry to demonstrate that its small-break LOCA methods continue to comply with the requirements of Appendix K to 10 CFR Part 50. The technical issues to be addressed were listed in NUREG-0611 including comparison with semiscale experimental test results. In response to Item II.K.3.30, the Westinghouse Owners Group (WOG) elected to reference the NOTRUMP code as the new licensing small-break LOCA model. The NOTRUMP code and methodology are described in WCAP-10079 and WCAP-10054. The staff reviewed and approved NOTRUMP as the new licensing tool for calculating small-break LOCA response for Westinghouse plant designs. The staff further concluded that the Westinghouse Owners Group had met the requirements of Item II.K.3.30.

Referencing the new computer code did not imply deficiencies in the WFLASH code (which was previously used for small-break LOCA analysis) so that the code did not comply with Appendix K to 10 CFR Part 50. The decision to use NOTRUMP was based on the industry's desire to perform licensing evaluations by means of a computer program specifically designed to calculate small-break LOCAs with greater phenomenological accuracy than was capable by using WFLASH.

Item II.K.3.31 of NUREG-0737 required that each license holder or applicant submit a new small-break analysis using the model approved under Item II.K.3.30. NRC Generic Letter 83-35 clarified the II.K.3.31 requirements by allowing license holders and applicants to comply on a generic basis by demonstrating that the WFLASH analyses are conservative when compared to analyses performed using NOTRUMP.

In response to this guidance, the Westinghouse owners submitted WCAP-11145 which contains generic comparisons to WFLASH analyses for various plant types. These include comparisons for four-loop plants such as the Watts Bar Units 1 and 2 design. If plant-specific analyses were performed for Watts Bar Units 1 and 2 using NOTRUMP, no core uncovery would be expected. Although there was no core uncovery in the NOTRUMP analysis, the 6-inch break remains the limiting break size.

The staff has completed its review of WCAP-11145 and has accepted that report as a licensing basis for small-break LOCA analysis. By letter dated October 17, 1986, the applicant has referenced WCAP-11145 (which consists of the results from calculations using approved methodology) in lieu of submitting a plant-specific analysis and meets the criteria stated in NRC Generic Letter 83-35. The staff, therefore, concludes that the Watts Bar Units 1 and 2 FSAR analyses of small-break LOCAs have been demonstrated to be conservative in comparison with the NOTRUMP evaluation model. This meets the requirements of II.K.3.31 and 10 CFR 50.46 for Watts Bar Units 1 and 2, and resolves proposed License Condition 36.

16 TECHNICAL SPECIFICATIONS

The staff is developing the Watts Bar TS, closely following the proposed new industry standard MERITS (Methodically Engineered, Restructured, and Improved Technical Specifications). All specific issues proposed by the applicant are communicated in open meetings tracked under TAC 76742. Any issues that would have an impact on previous conclusions (SER and SSERs 1 through 5) will be reported in appropriate sections in future SER supplements.

17 QUALITY ASSURANCE

17.1 General

The quality assurance (QA) program for the operations phase of Watts Bar is described in TVA's Nuclear Quality Assurance Plan (see TVA submittals of February 15 and June 5, 1990). The staff evaluation of the QA program described is based on a detailed review of this information. The staff assessed TVA's QA plan to determine if it complies with the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Plants and Fuel Reprocessing Plants"; with the regulatory guides listed in Table 17.1; and with Chapter 17 of the Standard Review Plan (NUREG-0800, Rev. 2).

17.2 Organization for the QA Program

The structure of the organization responsible for the operation of Watts Bar and for the establishment and execution of TVA's nuclear QA program is shown in Figure 17.1.

The Senior Vice President, Nuclear Power has the overall responsibility for the establishment, implementation, and administration of the nuclear QA program. He establishes the management policies and requirements that provide administrative controls to ensure that activities are performed in a manner that achieves compliance with preestablished quality objectives and acceptance criteria. The Senior Vice President has issued a policy statement regarding the nuclear QA program which assigns the responsibility for implementing the program to each employee in the nuclear power organization.

Thus, each vice president reporting to the Senior Vice President is assigned general QA responsibilities such as (1) the development, control, maintenance, and use of instructions and procedures to implement quality-related activities and processes; (2) the maintenance of housekeeping and cleanliness; (3) the identification and resolution of conditions adverse to quality; (4) the training and qualification of personnel within the organization, and (5) the control of records generated or used by the organization. Each vice president is also assigned specific QA responsibilities. For example, the Vice President, New Projects is responsible for implementing the nuclear maintenance program during the construction phase and the nuclear QA program for deferred plants. The Vice President, Nuclear Engineering is responsible for design control. The Vice President and Nuclear Technical Director is responsible for review of QA programs for suppliers of nuclear fuels and fuel-related components and services. The Vice President, Nuclear Assurance and Services is responsible for auditing, quality data trending, and computer software control.

As shown on Figure 17.1, five managers report to the Vice President, Nuclear Assurance and Services. Each of these managers has specific QA responsibilities. The Manager, Management Programs is responsible for QA reports, configuration management, and computer software. The Manager, Nuclear Training is responsible for the indoctrination and training programs.

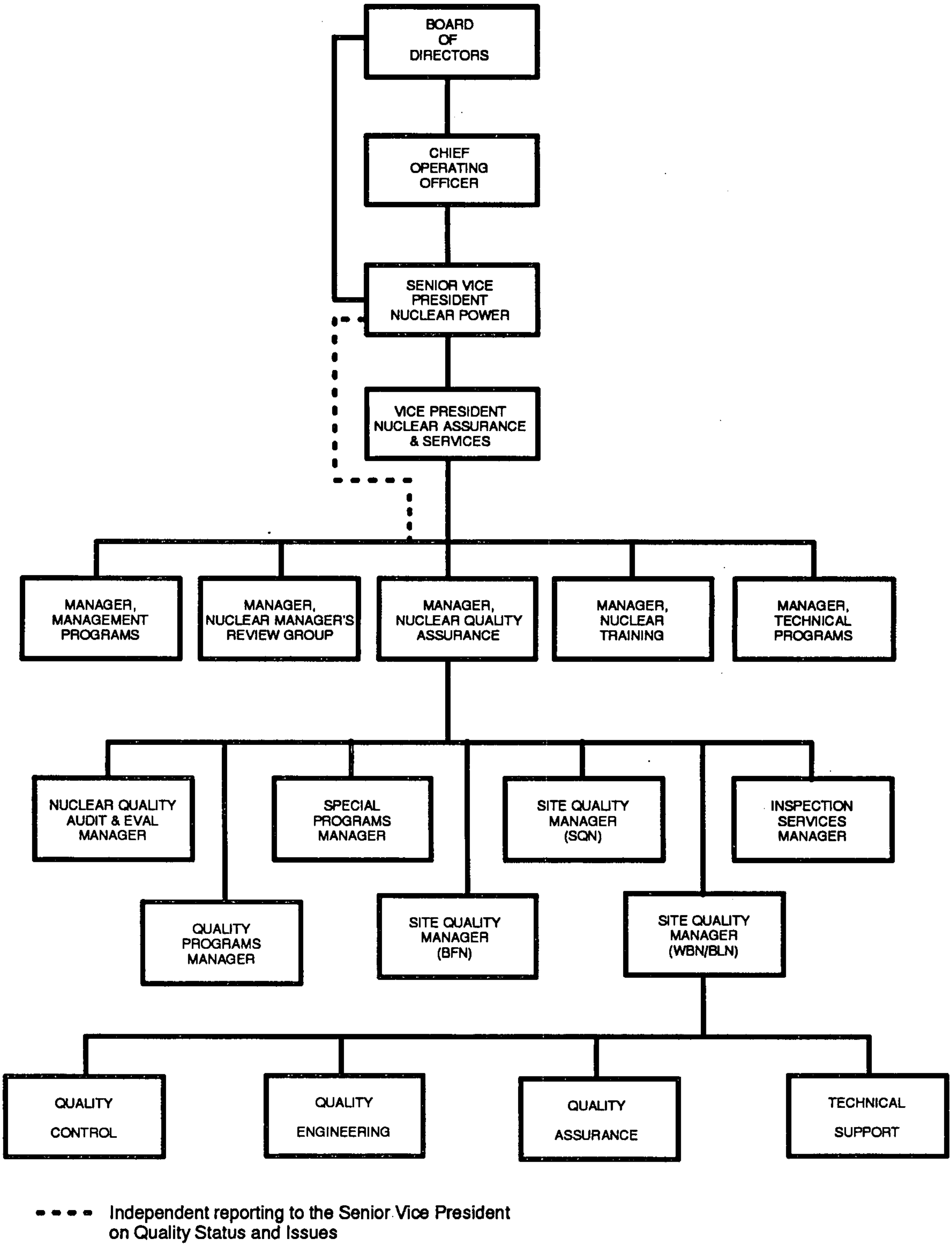

Figure 17.1 TVA organization for Watts Bar nuclear QA (Based on TVA Nuclear Quality Assurance Plan, dated February 15, 1990)

The Manager, Technical Programs is responsible for TVA's programs for environmental protection, protective services, emergency preparedness, radiological control, and radioactive waste management. The Manager, Nuclear Manager's Review Group is responsible for assessing engineering, design, construction, and operation activities and for checking the effectiveness of nuclear power programs and their implementation. The principal responsibility of the Manager, Nuclear Quality Assurance is the direction and management of the QA organization. The seven managers who report to the Manager, Nuclear Quality Assurance are responsible for such QA activities as internal and external audits, supplier surveillances and inspections, development and maintenance of the list of approved suppliers, development and implementation of the site quality control inspection program, and planning and implementing the ASME Code Section XI nondestructive examination inspection program.

17.3 Quality Assurance Program

The QA program includes the Nuclear Quality Assurance Plan and other documents approved by TVA management that are used to implement it. The program includes graded QA requirements and establishes the extent to which the graded requirements are imposed on specific items and activities. The Nuclear Quality Assurance Plan provides direction for implementing Appendix B to 10 CFR Part 50 and other TVA commitments for assuring and achieving quality.

On the basis of its review, the staff concludes that the Nuclear Quality Assurance Plan addresses the requirements of Appendix B to 10 CFR Part 50 and the provisions of the NRC regulatory guides shown in Table 17.1. The Quality Assurance Plan describes how the requirements of Appendix B to 10 CFR Part 50 are satisfied. It controls quality-related activities in a graded fashion to satisfy the requirements of Appendix B to 10 CFR Part 50.

The QA program requires that QA documents encompass detailed controls for:

(1) translating codes, standards, and regulatory requirements into specifications, procedures, and instructions

(2) developing, reviewing, and approving procurement documents, including changes

Table 17.1 Regulatory guidance applicable to the Quality Assurance Program

RG	Revision	Date	RG	Revision	Date
1.8*	2	4/87	1.74*	0	2/74
1.28	3	8/85	1.88*	2	10/76
1.30*	0	8/72	1.94*	1	4/76
1.33*	2	2/78	1.116*	0-R	5/77
1.37*	0	3/73	1.123*	1	7/77
1.38*	2	5/77	1.144*	1	9/80
1.39*	2	9/77	1.146*	0	8/80
1.58*	1	9/80	1.152*	0	11/85
1.64	2	6/76			

*With comments acceptable to the NRC.

(3) prescribing all activities that affect quality by documented instructions, procedures, or drawings

(4) issuing and distributing approved documents

(5) purchasing items and services

(6) identifying materials, parts, and components

(7) performing special processes

(8) inspecting and testing material, equipment, processes, and services

(9) calibrating and maintaining measuring and test equipment

(10) handling, storing, and shipping items

(11) identifying the inspection, test, and operating status of items

(12) identifying and dispositioning nonconforming items

(13) correcting conditions adverse to quality

(14) preparing and maintaining QA records

(15) auditing activities that affect quality

The QA program requires the establishment and continuous implementation of the QA indoctrination, training, and retraining programs to ensure that persons involved in safety-related activities are knowledgeable about QA instructions and implementing procedures and that they demonstrate a high level of competence and skill in the performance of their quality-related activities. Quality is verified through surveillance, inspection, testing, checking, and auditing of work activities. The QA program requires that quality verification activities be performed by qualified personnel who are not directly responsible for performing the work being verified. Verification is performed in accordance with procedures, instructions, and/or checklists by personnel who have been qualified and certified in accordance with codes, standards, and TVA training programs.

The Manager, Nuclear QA is responsible for QA audits. This includes planning, preparation, scheduling, performing, reporting, and verifying implementation of corrective and preventive action measures. The QA program establishes a comprehensive audit system to ensure that the QA program requirements and related supporting procedures are effective and properly implemented. Audits include an objective evaluation of QA practices, procedures, instructions, work areas, activities, processes, and items; of the effectiveness of implementation of the QA program; and of conformance with policy directives.

The QA program requires documentation of audit results and review by the management personnel who have responsibility in the area audited to determine and take corrective action as required. Reaudits are performed to determine that conditions adverse to quality have been effectively corrected and that the corrective action precludes repetitive occurrences.

The Nuclear Quality Assurance Plan commits TVA to develop, for Watts Bar, a Q-list which documents and classifies structures, systems, and components consistent with their importance to safety. The Q-list will include "quality-related" items. Quality-related items include safety-related items as well as items that are not safety related, but are important to reliable plant operation. The Q-list for each plant will reflect the guidance of Regulatory Guides 1.26 and 1.29, and the QA program will include such programs and features as radiological control, emergency preparedness, security, fire protection, chemistry, and the safety parameter display system.

The QA program provides for the graded application and verification of QA requirements to Q-listed items and related activities. The Nuclear Quality Assurance Plan provides criteria for grading QA program requirements and factors to be considered in the degree of QA verification required to ensure implementation of QA program requirements. This is acceptable to the staff.

17.4 Conclusion

The staff review of the TVA Nuclear Quality Assurance Plan for the operations phase of Watts Bar has verified that the criteria of Appendix B to 10 CFR Part 50 have been addressed.

On the basis of its review and evaluation of the QA program description contained in TVA's Nuclear Quality Assurance Plan, the staff concludes:

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

(2) TVA's Nuclear Quality Assurance Plan describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50 and with the acceptance criteria contained in SRP Chapter 17.

(3) The latest amendment of the FSAR pertinent to QA (Amendment 63) states that the identification of safety-related features will be addressed later. This will be addressed before issuance of an operating license. Hence Outstanding Issue 17 remains unresolved.

Except as noted in item 3 above, the staff concludes that TVA's nuclear QA program description is in compliance with applicable NRC regulations and is acceptable for the operations phase of Watts Bar. The staff's efforts will be tracked by TAC 76972.

18 HUMAN FACTORS ENGINEERING*

18.1 Detailed Control Room Design Review**

As a result of the Three Mile Island Unit 2 (TMI-2) accident, the staff developed an action plan (NUREG-0660) to minimize the possibility of such an accident recurring at commercial nuclear power plants. Item I.D.1, "Control Room Design Reviews," of NUREG-0660 requires operating reactor licensees and applicants for licenses to perform a detailed control room design review (DCRDR) to identify and correct design discrepancies. The goal of the DCRDR, as stated in NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent accidents or to cope with them, should they occur, by improving the information provided to those operators. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," confirmed and clarified the DCRDR requirements of NUREG-0660.

After the Watts Bar Nuclear Plant DCRDR was finished, TVA submitted a summary report to the NRC on October 2, 1987. The staff and Science Applications International Corporation (SAIC) reviewed the summary report and indicated they needed additional information. A preimplementation audit was scheduled in order to obtain this information and to resolve several concerns.

The audit was conducted at Watts Bar between November 11 and 18, 1988. The audit team evaluated the Watts Bar DCRDR in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews," and the nine DCRDR requirements contained in NUREG-0737, Supplement 1. Results of that audit and results of the staff's review of the summary report have been transmitted to the applicant (letter, S. C. Black, NRC, to O. D. Kingsley, TVA, dated April 28, 1989, but erroneously labeled as "April 28, 1988"). That letter is incorporated by reference. The staff noted that the applicant plans to correct all human engineering discrepancies (HEDs) at Unit 1 before fuel load.

By letter dated March 28, 1990, the applicant submitted a supplemental summary report, noting that the commitment, by letter dated February 23, 1990, to correlate the Watts Bar Safety Evaluation Report (NUREG-0847), Appendix D concerns to the DCRDR HEDs was completed and available for review on site. The staff performed another site audit August 21-23, 1990. The staff will report the results of the onsite audit in a future SSER, including an evaluation of whether the NUREG-0847, Appendix D concerns and NUREG-0737, Supplement 1 requirements have been satisfied. Proposed License Condition 37 remains unresolved.

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*Section 18 was titled "Control Room Design Review" in the SER. The current title is in accordance with the Standard Review Plan section published in 1984.

**Section 18.1 was titled "General" in the SER. The current title is in accordance with the Standard Review Plan section published in 1984.

18.2 Safety Parameter Display System*

On October 31, 1980, the staff issued NUREG-0737 which provided guidance for implementing TMI-2 action items. On December 17, 1982, Generic Letter 82-33 transmitted Supplement 1 to NUREG-0737 to all licensees and applicants to clarify the TMI-2 action items related to emergency response capability, including Item I.D.2, "Safety Parameter Display System" (SPDS). Supplement 1 extracted the fundamental requirements for emergency response capability from the wide range of regulatory documents issued on the subject. It was written at the conceptual level to allow for a high degree of flexibility in scheduling and design. In recognition of the interrelationships among the action items addressed in Supplement 1 to NUREG-0737, the staff made allowance for each licensee to negotiate a reasonable and achievable schedule for implementing its emergency response capability. However, the staff stated that because the SPDS can contribute to plant safety in an important way, it should be implemented promptly.

On April 12, 1989, the staff issued Generic Letter 89-06 to further clarify the SPDS requirements. The applicant responded by letter dated July 11, 1989. In the letter, the applicant reiterated its earlier commitment (letter, J. A. Domer, TVA, to NRC, dated June 25, 1985) to have the Watts Bar Unit 1 SPDS operational and to have operators trained to use it before startup following the first refueling outage. An "operational" SPDS will comply with NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342. The applicant plans to have a "functional" SPDS installed and operators trained to use it before fuel load. A "functional" SPDS will comply with NUREG-0737, Supplement 1, with the exception of documented availability, resolution of operator comments during the first cycle, and verification of displayed data with main control room indications.

To ensure that the applicant completes the activities necessary to fully satisfy the SPDS requirements of Supplement 1 to NUREG-0737, the following license condition (License Condition 43) is proposed for the operating license:

> Prior to startup following the first refueling outage TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

The applicant plans to complete an actual system test program before declaring the SPDS operational, which will include system availability testing. This testing cannot be completed until after commercial operation begins. The applicant committed to provide a supplemental response to Generic Letter 89-06 addressing certification of compliance with requirements of NUREG-0737, Supplement 1, within two months after the Unit 1 SPDS has become operational.

The staff performed a site audit August 21-23, 1990, and will report results of that audit in a future SSER.

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*Section 18.2 was titled "Conclusions" in the SER. The current title is in accordance with the Standard Review Plan section published in 1984.

APPENDIX A

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

NRC Letters

October 30, 1979	Letter from H. R. Denton to all licensees and applicants, "TMI Lessons Learned Short-Term Requirements and Implementation Schedule."
December 17, 1982	Generic Letter 82-33 issued, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability."
January 9, 1983	Summary of meeting with utility regarding exemption request from NUREG-0737, Item II.F.1, "Additional Accident Monitoring Instrumentation."
July 8, 1983	Generic Letter 83-28 issued, "Required Actions Based on Generic Implications of Salem ATWS Events."
November 2, 1983	Generic Letter 83-35 issued, "Clarification of TMI Action Plan Item II.K.3.31."
March 6, 1984	Letter from T. M. Novak to H. G. Parris (TVA) forwarding safety evaluation regarding installation of alarm in diesel engine cooling water keepwarm system. Installation of alarm required prior to fuel load.
January 9, 1985	Letter from T. M. Novak to H. G. Parris (TVA) requesting additional information on the fire protection program.
January 9, 1985	Letter from D. M. Verrelli to H. G. Parris (TVA) forwarding Federal Emergency Management Administration (FEMA) final report for 1984 emergency exercise.
January 14, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding safety evaluation on Open Items 13 and 14 concerning diesel generators and testing of communications systems.
January 15, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information concerning the analysis for main steamline break inside ice condenser containments.
January 28, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding November 28, 1984, memo from FEMA. Adequacy of state and local plans not yet determined.

January 28, 1985	Letter from T. M. Novak to H. G. Parris (TVA) stating that October 9, 1984, submittal concerning piping design criteria was not fully acceptable.
January 28, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information concerning associated circuits located in fire area.
January 30, 1985	Summary of January 22, 1985, meeting with utility concerning Technical Specifications issues.
February 13, 1985	Letter from D. M. Verrelli to H. G. Parris (TVA) advising that utility should furnish reference material in order to meet May 6, 1985, schedule for written and oral exams.
February 15, 1985	Letter from D. M. Verrelli to H. G. Parris (TVA) concerning NRC Bulletin 79-02 factor-of-safety question and missing calculation question.
February 21, 1985	Summary of February 6, 1985, meeting with utility regarding Technical Specifications.
March 4, 1985	Letter from T. M. Novak to H. G. Parris (TVA) forwarding draft license including Attachment 2 regarding requirements to be completed and Appendix B regarding nonradiological environmental protection plan.
March 6, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding information regarding the snubber issue.
March 8, 1985	Summary of February 28, 1985, meeting with utility regarding power system concerns.
March 12, 1985	Summary of March 3-4, 1985, meetings with utility to discuss readiness of facility to load fuel.
March 13, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding preliminary assessment of LOTIC-3 modules regarding equipment temperature environment following main steam line break accident in lower compartment of ice condenser containments.
March 15, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding list of open items based on review of Initial Test Program through FSAR Amendment 54.
March 18, 1985	Summary of March 7, 1985, meeting with utility concerning fire protection concerns and need to have T-cold indication in auxiliary control room.
March 26, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) noting that review of FSAR Amendment 55 regarding Initial Test Program revealed two additional open issues.

March 27, 1985	Letter from T. M. Novak to H. G. Parris (TVA) discussing Black & Veatch independent design verification program (IDVP) regarding auxiliary feedwater system.
March 28, 1985	Letter from T. M. Novak to H. G. Parris (TVA) requesting responses to questions concerning TVA's General Construction Specification G-29C as applied to American Welding Society Structural Welding Code D1.1.
April 2, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding responses to Generic Letter 83-28.
April 9, 1985	Summary of February 13, 1985, Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting with utility concerning status of open items, construction and quality assurance (QA) deficiencies, and fire protection and equipment qualification programs.
April 12, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding the perimeter intrusion detection system.
April 12, 1985	Transcript of April 12, 1985, meeting with utility.
April 15, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding notice of environmental assessment and finding of no significant impact regarding requests for exemption from Appendix J for airlock leakage tests and criticality monitor installation.
April 15, 1985	Letter from T. M. Novak to H. G. Parris (TVA) forwarding replacement pages to final draft Technical Specifications (TS). Issues requiring additional information to finalize TS were also enclosed.
April 15, 1985	Letter from T. M. Novak to H. G. Parris (TVA) determining that utility submittals amending the offsite dose calculation manual are consistent with the methodology and guidelines in NUREG-0133.
April 18, 1985	Summary of March 27, 1985, meeting with utility regarding use of Ruskin fire dampers.
April 23, 1985	Summary of April 12, 1985, meeting with utility regarding implementation of Westinghouse optimization program in Unit 1 TS.
April 23, 1985	Summary of March 25, 1985, meeting with utility regarding alternate shutdown capability.
April 23, 1985	Summary of April 10 and 12, 1985, meetings with utility regarding power systems concerns.

April 29, 1985	Letter from T. M. Novak to H. G. Parris (TVA) reporting that TVA's March 1, 1985, interpretation of Surveillance Requirement 4.8.1.1.2.f.11 of TS is incorrect.
May 16, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding list of documents required to address safety-related issues.
May 16, 1985	Letter from H. L. Thompson to H. G. Parris (TVA) forwarding 11 concerns regarding Watts Bar.
May 16, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) correcting portion of H. L Thompson letter of May 16, 1985.
May 20, 1985	Letter from T. M. Novak to H. G. Parris (TVA) requesting an update to the status of facility compliance with 10 CFR 50.49, addressing items that may be qualified before revised fuel load date.
May 20, 1985	Letter from T. M. Novak to H. G. Parris (TVA) forwarding the draft operating license and environmental protection plan, and replacement pages to final draft TS.
May 24, 1985	Summary of May 2, 1985, meeting with utility and public regarding TVA proposed enhancements to employee concerns program.
May 30, 1985	Letter from T. M. Novak to H. G. Parris (TVA) requesting additional information regarding Black & Veatch IDVP to support conclusion that design, construction, testing, and preparation for operation were completed per FSAR and licensing documents.
June 12, 1985	Transcript of June 12, 1985, meeting regarding Black & Veatch IDVP.
June 13, 1985	Summary of June 13, 1985, meeting with utility regarding preliminary results of employee concerns enhancement program.
June 17, 1985	Summary of May 30, 1985, meeting with utility to discuss status of licensing issues of plant.
June 18, 1985	Letter from D. M. Verrelli to H. G. Parris (TVA) forwarding FEMA interim report evaluating 1984 radiological emergency preparedness exercises.
June 26, 1985	Transcript of June 26, 1985, meeting of ACRS Subcommittee on Quality and QA in Design and Construction.
June 28, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding utility response to NRC Bulletin 79-02 concerning flexibility requirement in pipe support base plant design using concrete expansion anchors.

July 2, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) informing of adequacy of overall State and local radiological emergency response planning.
July 3, 1985	Letter from W. J. Dircks to C. Dean (TVA) forwarding brief review and discussion of major areas of performance deficiencies.
July 3, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding inspection of structural welds through paint.
July 9, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information regarding TVA's utilization of Black & Veatch IDVP.
July 9, 1985	Letter from J. N. Grace to H. G. Parris (TVA) forwarding appendix to systematic assessment of licensee performance (SALP) report.
July 10, 1985	Transcript of June 13, 1985, meeting with utility regarding preliminary resolution of enhanced employee concerns.
July 15, 1985	Summary of June 13, 1985, meeting with utility regarding resolution of nonconforming conditions involving installation of Unistrut clamp assemblies.
July 17, 1985	Letter from J. M. Taylor to H. G. Parris (TVA) advising that Item V.D of Order EA-85-49 is modified. Survey period need extend only from October 1 to March 21, 1985, and need only include nuclear-related portions of TVA's Office of Engineering.
July 22, 1985	Letter from T. M. Novak to H. G. Parris (TVA) forwarding draft safety evaluation regarding procedures generation package submitted in response to NUREG-0737, Supplement 1 (Generic Letter 82-33).
July 26, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting additional information to complete review of Generic Letter 83-28, Items 4.1, 4.2.1, and 4.2.2.2.
July 29, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) informing that NRC is experiencing delays in obtaining U.S. Dept. of Air Force report on fence disturbance system.
July 29, 1985	Letter from T. M. Novak to H. G. Parris (TVA) informing that named individuals may be used on shift as shift engineer and assistant shift engineer to meet criteria of Generic Letter 84-16 for on-shift hot operating experience.
July 30, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) requesting confirmation that offsite emergency plans include list of local or regional medical facilities with capabilities to provide treatment for radiation exposure.

August 7, 1985	Summary of June 6, 1986, meeting with Westinghouse Owners Group regarding proposed steam generator tube rupture generic program.
August 12, 1985	Summary of July 15-18, 1985, meeting with utility regarding Black & Veatch IDVP.
August 19, 1985	Letter from H. R. Denton to H. G. Parris (TVA) correcting misunderstanding that request for additional information regarding Generic Letter 83-28, Item 2.2.2 may have violated backfit requirements.
August 21, 1985	Letter from H. R. Denton to W. F. Willis (TVA) requesting information regarding how utility uses reports developed by TVA's Nuclear Safety Review Staff (NSRS).
August 23, 1985	Letter from J. N. Grace to H. G. Parris (TVA) informing of confirmatory action letter regarding inadequate and potentially inaccurate records of welder's recertifications.
September 11, 1985	Letter from T. M. Novak to H. G. Parris (TVA) forwarding replacement pages to final draft TS.
September 16, 1985	Transcript of September 16, 1985, meeting with utility regarding environmental qualifications.
September 17, 1985	Letter from W. J. Dircks to C. Dean (TVA) forwarding SALP Report 50-391/85-34 and requesting information under 10 CFR 50.54(f).
September 19, 1985	Letter from H. L. Thompson to H. G. Parris (TVA) denying request to withhold document.
September 20, 1985	Letter from H. R. Denton to H. G. Parris (TVA) forwarding Toledo Edison response regarding loss-of-feedwater event.
September 24, 1985	Letter from E. G. Adensam to C. Dean (TVA) forwarding pages inadvertently omitted from SALP report.
September 25, 1985	Transcript of September 25, 1985, meeting with utility regarding concerns and issues arising from welder recertification.
October 1, 1985	Letter from R. D. Walker to H. G. Parris (TVA) informing that actions regarding Inspection Report 50-390/85-32, concerning reactor protection system P-10 permissive bistable trip setpoint and reset values, meet the intent of the TS.
October 7, 1985	Letter from T. M. Novak to H. G. Parris (TVA) rejecting the request for TS modification for reactor coolant system (RCS) and charging pump flow at shutdown.

October 9, 1985	Transcript of September 16, 1985, meeting with utility regarding environmental qualification.
October 15, 1985	Summary of September 25, 1985, meeting with utility regarding the program for renewal of welder certifications.
October 29, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) forwarding the chronology of welding issues.
November 20, 1985	Letter from E. G. Adensam to H. G. Parris (TVA) extending an invitation to review U.S. Air Force test report regarding a perimeter intrusion detection system.
December 2, 1985	Summary of November 21, 1985, meeting with utility regarding the employee concern program.
December 5, 1985	Letter from B. J. Youngblood to H. G. Parris (TVA) requesting that all nuclear safety review staff reports be closed with line management. NRC cannot determine if the concerns were investigated.
December 10, 1985	Letter from B J. Youngblood to H. G. Parris (TVA) discussing the responses to Generic Letter 83-28, Items 3.1.3 and 3.2.3.
December 12, 1985	Transcript of December 12, 1985, meeting with utility regarding welding problems.
December 18, 1985	Transcript of December 12, 1985, meeting with utility regarding progress on concerns described in 10 CFR 50.54(f) letter.
December 20, 1985	Letter from B. J. Youngblood to H. G. Parris (TVA) requesting additional information regarding the TVA Corporate Performance Plan, per December 16, 1985, discussions.
December 26, 1985	Summary of December 18, 1985, meeting with utility regarding the response to the September 17, 1985, 10 CFR 50.54(f) letter on facility SALP.
January 7, 1986	Transcript of January 7, 1986, meeting with utility regarding welding issues.
January 15, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information concerning TVA's letter of November 1, 1985, regarding 10 CFR 50.54(f) and Sequoyah readiness. Watts Bar design control problems must be evaluated before startup.
January 15, 1986	Letter from D. M. Verrelli to H. G. Parris (TVA) advising of the availability of the Interactive Rapid Dose Assessment Model.

January 15, 1986	Letter from D. M. Verrelli to S. A. White (TVA) forwarding FEMA final report regarding limited number of available outside telephone lines and failure of medical drill to meet exercise objective.
February 3, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding enclosures to report to Commission on status of staff actions regarding TVA.
February 13, 1986	Letter from B. J. Youngblood to S. A. White (TVA) submitting observations and list of utility commitments from programmatic evaluation of employee concerns program.
March 5, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding draft reports on technical insights gained from probability risk analysis.
March 10, 1986	Letter from E. A. Trager to J. E. Atkins (TVA) responding to recent telephone conversations regarding AEOD/S401, "Human Error in Events Involving Wrong Unit/Wrong Train."
March 10, 1986	Letter from R. E. Alexander to E. A. Belvin (TVA) commenting on TVA's report, "Screening Labels for Radionuclides of Skin and Clothing."
April 10, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding environmental assessment and finding of no significant impact regarding requests for extension of construction completion dates.
April 15, 1986	Letter from J. A. Olshinski to S. A. White (TVA) advising that liquid samples spiked with radionuclides will be sent to facilities for radiochemical analysis as part of inspection program.
May 1, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding TVA's March 10, 1986, submittal concerning the revised nuclear corporate performance plan.
May 7, 1986	Summary of April 21, 1986, meeting with utility discussing the status of the evaluation of concrete concerns.
May 12, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting reconsideration of need for draft license conditions in May 20, 1985, draft license.
May 28, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding Volume 1 to "Preliminary Evaluation [of] TVA Corporate Nuclear Performance Plan."
June 10, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information on nuclear power plan quality assurance program implementation with respect to past and present issues.

June 10, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding cable pulling concerns.
June 10, 1986	Summary of May 29, 1986, meeting with utility regarding the physical security plan commitments concerning bullet-resistant barriers.
June 20, 1986	Summary of June 4, 1986, meeting with utility regarding schedules for completion of specific work items to support fuel load.
July 24, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding the project management plan for the Department of Energy (DOE) weld evaluation project.
July 24, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting a response to enclosed report concerning conformance to Revision 2 of Regulatory Guide 1.97.
July 24, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding safety evaluation supporting utility's use of the new Westinghouse small-break loss-of-coolant accident (LOCA) model NOTRUMP to satisfy Three Mile Island (TMI) Action Item II.K.3.30.
July 24, 1986	Summary of July 7-9, 1986, meetings with utility regarding Revision 9 to TVA-TR75-1A, "QA Program Description for Design, Construction and Operations for Nuclear Power Plants."
July 28, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information for review of FSAR Amendment 57 regarding emergency diesel generator justification for deviation from tornado missile spectrum criteria request.
July 28, 1986	Summary of July 17, 1986, meeting with utility regarding cable pulling.
July 30, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding the safety parameter display system (SPDS).
July 31, 1986	Letter from G. G. Zech to S. A. White (TVA) providing NRC positions on welding reinspection and welder qualification tests.
August 1, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding cable pulling and cable bending radii.
August 11, 1986	Letter from G. G. Zech to S. A. White (TVA) forwarding Sections I and II of the new incident response plan.

August 15, 1986	Letter from G. G. Zech to S. A. White (TVA) advising that the September 13, 1983, final deficiency report regarding shielded power cable bend radius was incomplete.
August 27, 1986	Letter from B. J. Youngblood to S. A. White (TVA) forwarding the ACRS report concerning TVA reorganization and shutdown of plants.
September 8, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting submittal of system component evaluation work sheets and tabs A and B from plant equipment documentation binders.
September 16, 1986	Summary of September 9-10, 1986, meeting with utility regarding cable pulling techniques.
October 14, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information concerning employee concerns report, "Heat Code as Related to Material Control."
October 14, 1986	Letter from R. H. Vollmer to S. A. White (TVA) providing NRC preliminary evaluation of EG&G structural weld reinspection data.
October 20, 1986	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding Generic Letter 83-28, Item 2.1 (Part 2), including description of vendor interface program.
October 23, 1986	Summary of October 14, 1986, meeting with utility regarding removal of resistance temperature bypass system replacement with Eagle 21 digital protection system.
November 7, 1986	Letter from T. J. Kenyon to S. A. White (TVA) forwarding topical report evaluation of WCAP-10858, "ATWS Mitigating System Actuating Circuitry Generic Design Package."
November 7, 1986	Letter from B. J. Youngblood to S. A. White (TVA) advising that utilization of NCIG-01 Rev. 2, "Visual Weld Acceptance Criteria for Structural Welding," should be reflected in safety analysis reports.
November 18, 1986	Transcript of November 18, 1986, management meeting with utility.
December 19, 1986	Summary of December 2, 1986, meeting with utility regarding corporate licensing activities.
December 22, 1986	Letter from H. R. Denton to W. Willis (TVA) forwarding "Staff Review of NRC Response to QA Breakdown Within TVA Nuclear Program."
January 5, 1987	Letter from V. Stello to D. H. Dean (TVA) advising that the welding aspects of the QA program were found to be deficient.

January 16, 1987	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding December 1, 1986, submittal on resistance temperature detector (RTD) bypass manifold removal.
January 20, 1987	Summary of January 15, 1987, meeting with utility regarding utility proposal to resolve piece parts replacement issue.
January 21, 1987	Transcript of January 21, 1987, meeting with utility regarding the plant welding reinspection program.
February 2, 1987	Letter from B. J. Youngblood to S. A. White (TVA) requesting that draft safety evaluation concerning concrete quality be forwarded.
February 10, 1987	Summary of January 31, 1987, meeting with utility regarding the plant welding reinspection program.
February 11, 1987	Letter from B. J. Youngblood to S. A. White (TVA) forwarding status of review efforts of utility response to selected items in Generic Letter 83-28.
March 10, 1987	Letter from B. J. Youngblood to S. A. White (TVA) forwarding Franklin Research Center, "Evaluation of Watts Bar Units 1 & 2 Cable Pulling & Bend Radii Concerns," technical evaluation report.
March 10, 1987	Transcript of meeting with utility regarding the status of activities and plans.
March 25, 1987	Letter from B. J. Youngblood to S. A. White (TVA) requesting additional information regarding RTD bypass removal modification.
April 22, 1987	Letter from J. G. Keppler to S. A. White (TVA) requesting additional information regarding Revision 4 to TVA Corporate Nuclear Performance Plan.
April 23, 1987	Letter from G. G. Zech to S. A. White (TVA) advising that liquid samples spiked with radionuclides will be sent to facilities for radiochemical analysis as part of NRC program.
April 27, 1987	Letter from J. A. Zwolinski to S. A. White (TVA) forwarding final results of first audit of verification and validation plan for Eagle 21 system utilization.
April 27, 1987	Letter from J. A. Zwolinski to S. A. White (TVA) forwarding NRC letter regarding results of evaluation of natural circulation, boron mixing, and cooldown test.
April 28, 1987	Summary of March 19, 1987, meeting with utility regarding utility position on welding issues and implementation of August 23, 1985, confirmatory action letter.

May 1, 1987	Summary of March 10, 1987, meeting with utility regarding recovery efforts to date.
May 20, 1987	Letter from J. A. Zwolinski to S. A. White (TVA) requesting additional information regarding the status of nuclear safety review staff reports.
July 2, 1987	Summary of June 26, 1987, meeting with utility regarding TVA commitment to requirements of American Society of Mechanical Engineers (ASME) Section III for welding activities.
August 10, 1987	Letter from G. G. Zech to S. A. White (TVA) requesting additional information regarding design control inadequacies and cleanliness classes specified by American National Standards Institute (ANSI) Standard N45.2.1.
August 27, 1987	Letter from J. A. Zwolinski to S. A. White (TVA) forwarding LA-UR-86-2053, "Evaluation of Revised LOTIC-3 Drain-Flow Heat Transfer Models," regarding containment temperature.
September 30, 1987	Letter from J. A. Zwolinski to S. A. White (TVA) forwarding safety evaluation of utility employee concerns program.
January 11, 1988	Summary of December 15, 1988, meeting with utility regarding TVA program to address microbiologically induced corrosion.
January 20, 1988	Summary of November 23, 1988, meeting with utility regarding "Revised LOTIC-3 Drain-Flow Heat Transfer Models," for main steam break analysis.
January 25, 1988	Letter from G. G. Zech to S. A. White (TVA) granting utility authorization to use ASME Code Cases N-341 and N-356 to extend certification period for Level III examiners to 5 years.
February 2, 1988	Letter from G. G. Zech to S. A. White (TVA) requesting additional information regarding May 10, 1986, response to NRC Bulletin 85-03, "Motor-Operated Common Mode Failures During Plant Transients Due to Improper Switch Settings."
February 17, 1988	Letter from S. D. Ebneter to S. A. White (TVA) responding to November 9, 1987, letter and identifying concerns with proposed plan for assessment of facilities.
February 24, 1988	Summary of February 4, 1988, meeting with utility regarding transfer of Watts Bar operators to Sequoyah and status of operator licenses at Watts Bar.
March 16, 1988	Letter from K. P. Barr to S. A. White (TVA) informing that waiver of eligibility of operator exams is granted, pending submittal of NRC Form 398.

April 8, 1988	Letter from G. G. Zech to S. A. White (TVA) requesting additional information regarding main steam line breaks in ice condenser containments in support of national laboratory analysis of WCAP-10986P.
June 23, 1988	Summary of June 3, 1988, meeting with utility and Sargent & Lundy regarding Watts Bar program plan and schedule.
June 23, 1988	Summary of June 7, 1988, meeting with utility regarding utility overall plan for completion of plant.
June 27, 1988	Letter from S. D. Richardson to S. A. White (TVA) responding to May 27, 1988, letter regarding program plan for completion of facilities.
July 15, 1988	Letter from S. C. Black to S. A. White (TVA) requesting additional information regarding main steam line breaks in ice condenser plants, per review of Addendum 1 to WCAP-10988, "Ice Condenser Drain Test Results."
July 20, 1988	Letter from S. C. Black to S. A. White (TVA) informing that utility response to Generic Letter 88-03 was satisfactory and no further action was necessary. Safety evaluation included.
August 12, 1988	Letter from J. G. Partlow to S. A. White (TVA) forwarding "Review and Evaluation of DOE Weld Evaluation Project Report, Watts Bar Nuclear Plant, Unit 1."
August 31, 1988	Letter from S. D. Richardson to S. A. White (TVA) responding to June 24, 1988, letter regarding plant vertical slice review plan.
September 15, 1988	Letter from F. R. McCoy to S. A. White (TVA) advising that the one-year onsite eligibility requirement will be waived on the basis of training and experience completed by plant operators.
October 4, 1988	Letter from J. G. Partlow to M. A. Runyon (TVA) informing that notification satisfied request contained in NRC safety evaluation report of Corporate Nuclear Performance Plan, regarding O. D. Kingsley succeeding S. A. White.
October 11, 1988	Corrected transcript of October 11, 1988, meeting with utility regarding plant weld review.
November 23, 1988	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding the November 17, 1988, letter from NRC's Executive Director for Operations to all utility executives regarding presence of NRC inspector at plants, revision to enforcement policy, and so forth.

December 5, 1988	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding environmental assessment and finding of no significant impact regarding June 29, 1988, request for extension of construction completion dates to July 1, 1991, and December 31, 1992, for Units 1 and 2, respectively.
December 5, 1988	Letter from S. D. Richardson to O. D. Kingsley (TVA) requesting utility assurance that Generic Letter 88-17 regarding loss of decay heat removal during nonpower operation is being addressed accordingly.
January 6, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation of utility responses to NRC Bulletin 87-02, Supplements 1 and 2, "Fastener Testing."
February 8, 1989	Letter from B. D. Liaw to O. D. Kingsley (TVA) granting extension for continued use of QA Topical Report TVA-TR75-1A, Rev. 10 until June 30, 1989.
February 9, 1989	Summary of October 11, 1988, meeting with utility regarding status of plant welding review program.
February 9, 1989	Letter from D. D. Crutchfield to O. D. Kingsley (TVA) discussing criteria for reactor licensing reviews. Utility should identify nature of FSAR changes since NUREG-0847, Supplement 4 and evaluate each.
February 15, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting additional information regarding the corrective action program plan for containment isolation.
February 16, 1989	Summary of January 18-19, 1989, meeting with utility regarding corrective action programs in civil/structural areas.
March 3, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting comments on draft, "COMMIX Analysis of Main Steamline Break in Catawba Lower Containment," concerning exceeding qualified temperatures.
March 9, 1989	Summary of February 7-8, 1989, meeting with utility regarding Watts Bar Corrective Action Programs.
April 3, 1989	Generic Letter 89-04 issued, "Guidance on Developing Acceptable Inservice Testing Programs."
April 12, 1989	Generic Letter 89-06 issued, "Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54(f)."
April 13, 1989	Summary of March 16-17, 1989, meeting with utility regarding electrical review program.

April 28, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation and SAIC-88-1821, "Technical Evaluation Report of DCRDR for Watts Bar." Additional information requested.
May 17, 1989	Summary of April 18, 1989, meeting with utility regarding plant prestart test corrective action program.
May 26, 1989	Letter from B. D. Liaw to O. D. Kingsley (TVA) informing that Revision 0 to "Nuclear QA Plan," may reduce commitments in Topical Report TVA-TR75-1A. QA plan cannot be implemented until approval recommended.
June 1, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting utility response to Generic Letter 83-28, Item 2.2 (Part 1) regarding equipment classification program.
June 6, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) informing that modification of sample frequency and suspension of background data collection until 6 months preceding fuel load is unacceptable.
June 13, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting utility use of Eagle 21 microprocessor system for monitoring RCS temperature and performing other safety-related functions.
June 23, 1989	Letter from B. D. Liaw to O. D. Kingsley (TVA) discussing integration of engineering assurance functions into quality assurance and engineering areas.
June 27, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting additional information regarding seismic design considerations for safety-related vertical steel tanks, per Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria."
June 29, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) advising that December 9, 1988, response adequately addresses items requested in Generic Letter 88-11, "NRC Position on Radiation Embrittlement."
July 10, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting additional information regarding the nuclear quality assurance plan.
July 21, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) advising that Revision 6 to Corporate Nuclear Performance Plan is acceptable.
August 4, 1989	Summary of July 21, 1989, meeting with utility regarding employee concerns programs.

August 18, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) advising that response to Bulletin 87-02, Supplements 1 and 2, "Fastener Testing," satisfactorily addressed requirements.
August 23, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding environmental assessment and finding of no significant impact regarding temporary exemption concerning physical inventory of material.
September 1, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation supporting utility corrective action program plan for electrical conduit and conduit support.
September 7, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) informing of completion of preliminary review of "Watts Bar Fire Protection Corrective Action Program."
September 7, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) advising that utility overall approach for development of amplified response spectra provides technically viable approach.
September 8, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting Revision 2 to corrective action plan for instrument lines.
September 11, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting corrective action program for electrical issues.
September 11, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting corrective action program for seismic issues.
September 11, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) advising that NRC completed review of Q-list corrective action plan.
September 13, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting programmatic aspects of corrective action plan for Category 1 cable tray and cable tray supports.
September 27, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) informing that applicant activities over last 11 years must be updated.
October 6, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation regarding corrective action program plan for hanger and analysis update program (HUAP).
October 17, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting prestart test corrective action program.
October 24, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation supporting corrective action program plan for safety-related heating, ventilation, and air conditioning (HVAC) duct and duct supports.

October 24, 1989	Summary of October 11, 1989, meeting with utility regarding plans for training and testing operators before licensing. Utility needs requalification plan which conforms to NUREG-1021.
October 31, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation for validation of Bechtel computer code SASSI for plant soil-structure interaction analysis.
November 1, 1989	Letter from T. A. Peebles to C. H. Noe (TVA) forwarding Forms A and B with answers for generic fundamentals exam section of written operator licensing exam given on October 4, 1989.
November 7, 1989	Summary of July 20, 1989, meeting with utility regarding vertical slice review program.
November 22, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting corrective action program plan for replacement items program (piece parts).
November 22, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting additional information regarding Topical Report TVA-NPOD89, "Nuclear Power Organization Description."
November 29, 1989	Summary of November 17, 1989, meeting with utility regarding cable damage resolution plan.
December 5, 1989	Letter from B. A. Wilson to O. D. Kingsley (TVA) advising that Revisions 3 and 4 to radiological emergency plan meet the planning standards of 10 CFR 50.47(b) and 10 CFR Part 50, Appendix E.
December 7, 1989	Summary of November 21, 1989, meeting with utility regarding priority licensing activity. No technical issues were discussed.
December 8, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) discussing the review of programmatic aspects of corrective action plan for quality assurance records. Program is acceptable with stated qualifications.
December 11, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) authorizing the use of ASME Code Case N-460, "Alternative Exam Coverage for Class 1 and Class 2 Welds."
December 13, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) requesting additional information regarding utility request to apply leak-before-break technology to primary loop piping (General Design Criterion (GDC) 4).
December 21, 1989	Summary of December 12, 1989, meeting with utility regarding methodology being applied for ensuring free expression of safety concerns.

December 28, 1989	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting anticipated transient without scram (ATWS) mitigation system actuation circuitry design. The design is in compliance with 10 CFR 50.62 ATWS rule.
January 3, 1990	Summary of December 18, 1989, meeting with utility regarding integrated living schedule program.
January 12, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) advising that the response to Generic Letter 88-20 regarding individual plant exams is acceptable. Notify NRC of any schedule changes.
January 18, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation accepting Quality Assurance Plan, Revision 0. This plan supersedes TVA-TR75-1A, Revision 10.
January 19, 1990	Summary of January 11, 1990, meeting with utility regarding status of integration of Engineering Assurance group into Nuclear QA group.
February 1, 1990	Summary of January 11, 1990, meeting with utility regarding security upgrade and interface with Unit 2 construction activities.
February 6, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) responding to question asked in December telephone conversation regarding fire brigade training.
February 26, 1990	Letter from D. D. Crutchfield to O. D. Kingsley (TVA) advising that prior notification of permanent changes to utility organization and senior managers is no longer required.
February 27, 1990	Summary of February 7, 1990, meeting with utility regarding status and scheduling of various licensing issues.
March 6, 1990	Summary of February 15-16, 1990, meeting with utility to discuss plans to resolve issue of damaged electrical cables in conduits.
March 23, 1990	Letter from T. A. Peebles to O. D. Kingsley (TVA) advising of operating and written exams scheduled for week of June 25, 1990.
April 24, 1990	Summary of March 27, 1990, meeting with utility regarding status of prestart test program.
April 26, 1990	Letter from B. A. Wilson to O. D. Kingsley (TVA) providing information regarding interpretation of bioassay measurements to assess intakes of radioactive material per NRC Information Notice 82-18.

May 1, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding understanding of current status of all unimplemented unresolved safety issues (USIs). Reference Generic Letter 89-21.
May 17, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) forwarding safety evaluation supporting utility request for elimination of dynamic effects of postulated primary loop pipe ruptures (GDC 4).
May 24, 1990	Letter from T. E. Murley to O. D. Kingsley (TVA) advising of Office of Nuclear Reactor Regulation (NRR) reorganization and appointment of P. S. Tam as new project manager.
May 24, 1990	Letter from S. C. Black to O. D. Kingsley (TVA) advising that response to NRC Bulletin 88-04, "Potential Safety-Related Pump Loss," is complete.
May 29, 1990	Summary of May 14, 1990, meeting with utility regarding status and schedule of various licensing activities.
June 4, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that the revised submittal date regarding applicability of Sequoyah main steam line break is acceptable.
June 6, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information regarding December 5, 1989, response to updated Regulatory Guide 9.3.
June 7, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that March 31, 1988, letter was found fully responsive to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes."
June 8, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that the response to TMI Action Item II.K.3.5 regarding coolant pump trip criteria is acceptable.
June 15, 1990	Summary of May 31, 1990, meeting with utility regarding structural adequacy of HVAC duct welds.
June 15, 1990	Summary of May 22, 1990, meeting with utility regarding status of electrical cable issues.
June 18, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) forwarding safety evaluation accepting response to Generic Letter 83-28, Item 2.1, Part 1, "Equipment Classification for Reactor Trip System Components."
June 18, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting utility response to Generic Letter 83-28, Item 4.3, "Shunt Attachment to Reactor Trip Breaker."

June 19, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting consideration of observations to TVA responses to Generic Letter 88-17 regarding expeditious actions for loss of decay heat removal.
June 22, 1990	Summary of June 12, 1990, meeting with utility regarding general policy and procedural issues.
June 22, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that response to NRC Bulletin 89-03, "Potential Loss of Required Shutdown Margin During Refueling Operations," was satisfactory.
June 22, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that response to NRC Bulletin 89-02, "Stress Corrosion Cracking of Anchor Darling Check Valve Bolts and Pins," was acceptable.
June 22, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information regarding June 16, 1990, response to NRC Bulletin 89-01. Response did not explicitly commit to not installing plugs from bad heats.
June 28, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) forwarding safety evaluations accepting November 7, 1983, response to Generic Letter 83-28, Items 4.5.2 and 4.5.3, regarding reactor trip system reliability.
June 29, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that existing power-operated relief valve (PORV), safety valve, and high-pressure reactor trip setpoints meet NUREG-0737, Item II.K.3.1, requirements.
July 2, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising of resolution of Items 3.1.3 and 3.2.3 of Generic Letter 83-28 per utility responses of November 7, 1983, and January 17, 1986.
July 9, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting January 26, 1990, response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."
July 12, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting response to TMI Item II.E.4.2 and forwarding Revisions 0, 1, and 2 to technical evaluation report, "Watts Bar Demonstration of Containment Purge and Vent Valve Operability."
July 26, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting response to Generic Letter 88-14, "Instrument Air Safety System Problems."
August 8, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting response to Generic Letter 88-05.

August 13, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting response to Generic Letter 83-28, Item 1.1, "Post-Trip Review."
August 13, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information regarding the January 29, 1987, response concerning "Functional Requirements Document for the Bypassed and Inoperable Status Indication (BISI) System."
August 16, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information regarding the July 20, 1990, response concerning Revision 3 of the Q-list corrective action program.
August 17, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting the July 20, 1989, response concerning the meteorological monitoring program.
September 7, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting the August 24, 1990, response to Generic Letter 90-03.
September 11, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting Revision 3 of the corrective action program plan for vendor information and Watts Bar Nuclear Performance Plan on vendor information.
September 14, 1990	Letter from F. J. Hebdon to O. D. Kingsley (TVA) accepting response to Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

TVA Letters

March 17, 1982	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding information regarding compliance of diesel engine fuel oil system to ANSI N195-76 and Regulatory Guide 1.37, Position C.2. Information resolves Open Item 57 in draft safety evaluation report.
December 14, 1982	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding information to resolve safety evaluation report Open Item 14 regarding diesel generator operation without secondary coolant.
January 25, 1983	Letter from D. S. Kammer to E. G. Adensam (NRC) commenting on NUREG-0847.
May 10, 1983	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding additional justification required by NUREG-0847 regarding hydrogen control capabilities, also known as permanent hydrogen mitigation system (PHMS).
November 8, 1983	Letter from L. M. Mills to E. G. Adensam (NRC) requesting exemption from installing high-range noble gas monitors on steam generator safety and PORV release lines, per NUREG-0737, Item II.F.1.

November 22, 1983	Letter from L. M. Mills to E. G. Adensam (NRC) requesting exemption from installing high-range noble gas monitors on auxiliary building vent as required by NUREG-0737, TMI Item II.F.1.
January 24, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) requesting exemption from installing high-range noble gas monitors on auxiliary building vent as required by NUREG-0737, TMI Item II.F.1.
May 8, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) requesting exemption from requirements of NUREG-0737, Item II.F.1, regarding installation of high-range noble gas monitors. Monitors will be installed by first refueling outage.
May 25, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding additional information regarding deliberate ignition hydrogen control (PHMS).
August 30, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding Westinghouse report, "RCS Flow Uncertainties With Use of Rosemount Resistance Temperature Devices."
October 4, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding revised response to NUREG-0737, TMI Item III.D.1.1, "Integrity of System Outside Containment."
October 19, 1984	Letter from L. M. Mills to E. G. Adensam (NRC) forwarding information regarding action to ensure operability of diesel generators by maintaining minimum room temperature.
November 27, 1984	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding information regarding permanent hydrogen mitigation system.
February 7, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding revised TS Figure 3.2-3, "RCS Total Flow Rate vs. R."
February 13, 1985	Letter from J. W. Hufham to E. G. Adensam (NRC) requesting exemption from cold no-flow, cold full-flow, and hot no-flow drop testing, based on requirements for rod drop testing in Regulatory Guide 1.68, Revision 2.
February 14, 1985	Letter from J. W. Hufham to E. G. Adensam (NRC) forwarding final draft proposed marked-up TS 4.6.4.3 regarding surveillance requirements for demonstrating operability of hydrogen ignitors.
February 15, 1985	Letter from J. W. Hufham to E. G. Adensam (NRC) proposing revision to TS 3/4.4.6.2 regarding RCS pressure isolation valve leakage criteria.
February 15, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) responding to NRC question regarding diesel generator piping classification and crankcase explosion protection.

February 15, 1985	Letter from J. W. Hufham to E. G. Adensam (NRC) forwarding results of reduced ice weight analysis performed by Westinghouse and corresponding FSAR revisions.
February 16, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) confirming that contract for acceptance of spent nuclear fuel was executed with U.S. Department of Energy.
February 16, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) requesting exemption from 10 CFR 70.24(d) as granted in special nuclear material license.
February 16, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding information regarding change in TS requirements for diesel fuel oil sampling.
March 18, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) forwarding information clarifying appropriateness of CLASIX heat transfer models regarding PHMS.
March 18, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) forwarding information regarding the diesel generators.
March 24, 1985	Letter from D. E. McCloud to E. G. Adensam (NRC) requesting exemption from reporting requirements of 10 CFR 50.73 during venting operations.
March 25, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) forwarding March 12, 1985, letter to Westinghouse submitting proposed FSAR and TS revisions regarding turbine integrity program.
March 27, 1985	Letter from D. E. McCloud to E. G. Adensam (NRC) responding to concerns noted in March 15, 1985, letter regarding initial test program.
April 5, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding calculations regarding minimum subatmospheric pressure inside containment and PHMS.
April 15, 1985	Letter from D. E. McCloud to E. G. Adensam (NRC) forwarding response to Open Items 14A and 14B regarding initial test program and rod cluster control assembly.
April 17, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding additional information demonstrating Morrison-Knudsen Company diesel generator satisfactory start in 40°F ambient environment.
April 18, 1985	Letter from R. H. Shell to E. G. Adensam (NRC) forwarding additional information regarding response to Generic Letter 83-28.

April 26, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) requesting deletion of TS 3.3.3.7 and Surveillance Requirement 4.7.7.e.4 regarding chlorine detectors.
April 26, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) responding to concerns regarding monitoring instrumentation.
May 31, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) clarifying Startup Test 3.9, "Natural Circulation Test," of initial test program.
June 25, 1985	Letter from J. A. Domer to E. G. Adensam (NRC) forwarding additional information regarding human factors engineering for SPDS.
October 17, 1986	Letter from R. L. Gridley to B. J. Youngblood (NRC) referencing WCAP-11145, TMI Action Item II.K.3.31, and Generic Letter 83-35.
October 2, 1987	Letter from R. L. Gridley (TVA) to NRC forwarding "DCRDR Summary Report for Watts Bar Nuclear Plant, Units 1 and 2."
October 20, 1988	Letter from S. A. White to NRC forwarding Revision 0 to "Watts Bar Nuclear Plant Containment Isolation Corrective Action Program Plan for Unit 1 and Common Features."
April 17, 1989	Letter from C. H. Fox to NRC forwarding proprietary WCAP-11985 and non-proprietary WCAP-11984, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as Structural Design Basis for Watts Bar, Units 1 and 2."
May 12, 1989	Letter from O. D. Kingsley to NRC forwarding additional information regarding containment isolation.
August 21, 1989	Letter from M. J. Ray to NRC requesting deferral of 6-month review and confirmation letter requested by Generic Letter 89-04.
February 15, 1990	Letter from M. J. Ray to NRC forwarding Revision 0 to "TVA Nuclear QA Plan," including all corrected and revised pages and changes.
March 28, 1990	Letter from E. G. Wallace to NRC forwarding "DCRDR Supplemental Summary Report," dated February 1990.
June 5, 1990	Letter from M. O. Medford (TVA) to NRC forwarding proposed changes to utility QA plan.

APPENDIX B

BIBLIOGRAPHY

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

--- Topical Report TVA-TR 75-1A "Quality Assurance Program Description for Design, Construction, and Operation for Nuclear Power Plants," Revs. 5, 6, 7, and 8.

U.S. General Services Administration, Office of the Federal Register, National Archives and Records Service, Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C. (includes general design criteria).

U.S. Nuclear Regulatory Commission, NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," December 1976.

---, NUREG-0452 "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," various revisions.

---, NUREG-0472, "Draft Radiological Technical Specifications for PWRs," November 1978.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980.

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

---, NUREG-0700, "Guidelines for Control Room Design Reviews," September 1981.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supp. 1, January 1983 (also issued as Generic Letter 82-33).

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," July 1981 (includes Branch Technical Positions).

---, NUREG-0847, Supplement 3, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2, Tennessee Valley Authority," January 31, 1985.

---, NUREG-0847, Supplement 4, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2, Tennessee Valley Authority," March 31, 1985.

---, NUREG-0954, "Safety Evaluation Report Related to the Operation of Catawba Nuclear Station, Units 1 and 2," February 9, 1983; Supp. 1, April 1983; Supp. 2, June 1984; Supp. 4, December 1984.

---, NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," Vol. 1, April 1983; Vol. 2, August 1983.

---, NUREG-1232, Volume 1, "Safety Evaluation Report on TVA Revised Corporate Nuclear Performance Plan," July 31, 1987.

---, NUREG-1232, Volume 4, "Safety Evaluation Report on TVA Watts Bar Nuclear Performance Plans," January 31, 1990.

--- NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," February 1979.

---, NUREG/CR-2628, "Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement," March 1982.

---, Office of Inspection and Enforcement (IE) Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," March 8, 1979.

---, IE Bulletin 80-06, "Engineered Safety Features (ESF) Reset Controls," March 12, 1980.

---, Inspection Report 50-390/84-09, February 22, 1984.

---, Inspection Report 50-390/84-20, April 12, 1984.

---, Inspection Report 50-390/84-28, May 11, 1984.

---, Inspection Report 50-390/84-35, June 21, 1984.

---, Inspection Report 50-390/84-37, July 13, 1984.

---, Inspection Report 50-390/84-58, August 22, 1984.

---, Inspection Report 50-390/84-59, November 7, 1984.

---, Inspection Report 50-390/84-77, November 14, 1984.

---, Inspection Report 50-390/84-85, January 8, 1985.

---, Inspection Report 50-390/84-90, February 11, 1985.

---, Inspection Report 50-390/85-01, January 31, 1985.

---, Inspection Report 50-390/85-08, March 29, 1985.

Westinghouse Electric Corporation, "Analysis of the Probability of the Generation and Strike of Missiles From a Nuclear Turbine," Westinghouse Steam Turbine Generation Division, March 1974.

---, Topical Report WCAP-7769, "Overpressure Protection for Westinghouse Pressurized Reactors," Rev. 1, October 8, 1971.

---, Topical Report WCAP-8354(P), "Long Term Ice Condenser Containment Code, LOTIC Code," Supp. 2, February 1979; and Supp. 3, September 1984. (Proprietary, not publicly available.)

---, Topical Report WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency", June 30, 1987. (Proprietary, not publicly available.)

---, Topical Report WCAP-10054, "Westinghouse Small-Break ECCS Evaluation Model Using NOTRUMP Code", December 31, 1982. (Proprietary, not publicly available.)

---, WSTG-1-P, "Procedures for Estimating the Probability of Steam Turbine Disc Rupture From Stress Corrosion Cracking," Westinghouse Steam Turbine Generator Division, May 1981. (Proprietary, not publicly available.)

---, WSTG-2-P, "Missile Energy Analysis Methods for Nuclear Steam Turbines," Westinghouse Steam Turbine Generator Division, May 1981. (Proprietary, not publicly available.)

---, WSTG-3-P, "Analysis of the Probability of a Nuclear Turbine Reaching Destructive Overspeed," Westinghouse Steam Turbine Generator Division, July 1984. (Proprietary, not publicly available.)

---, Topical Report WCAP-10079, "NOTRUMP Nodal Transient Small Break and General Network Code", November 15, 1982. (Propriety, not publicly available.)

---, Topical Report WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With NOTRUMP Code," June 11, 1986. (Proprietary, not publicly available.)

APPENDIX E

PRINCIPAL CONTRIBUTORS

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APPENDIX I

CORRECTIVE ACTION PROGRAM PLAN

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION VOLUME 4, NUCLEAR PERFORMANCE PLAN ON VENDOR INFORMATION AND REVISION 3 OF VENDOR INFORMATION CORRECTIVE ACTION PROGRAM PLAN WATTS BAR NUCLEAR PLANT UNIT 1 DOCKET NO. 50-390

TVA identified a number of problems with their vendor information program at Watts Bar Nuclear Plant (WBN) through condition adverse to quality (CAQ) reports, employee concerns and TVA and NRC audit findings. Specific problems identified include: (1) vendor information that was inadequately evaluated for implementation; (2) vendor information that did not match the plant configuration; (3) vendor information that was inconsistent with related TVA developed design input/output documents; (4) incorrect or out-of-date vendor documents; (5) inadequate vendor document control program; (6) manuals lost or uncontrolled; and (7) installations not approved by TVA Nuclear Engineering (NE).

TVA identified the root causes of these problems to be: (1) vendor documents were not considered as documents requiring configuration control; (2) inadequate procedural requirements to govern the receipt, review, distribution, filing, control, maintenance and use of information; and (3) a lack of attention to detail. The Vendor Information (VI) CAP was established to resolve and prevent recurrence of problems with vendor information at WBN.

2.0 EVALUATION

The VI CAP was established to provide reasonable assurance that vendor technical documents for safety-related equipment at WBN are current, complete, and appropriately updated for the life of the plant. The CAP will ensure that information in these documents is appropriately used as input to TVA design output documents, plant instructions and procedures, and the plant as-built configuration.

2.1 Vendor Technical Documents/Vendor Technical Manuals

The CAP provides for the identification of sets of vendor documentation defined as Vendor Technical Documents (VTDs). VTDs contain vendor technical information necessary to support safety-related equipment installation, operation, maintenance and testing. These VTDs are consolidated into Vendor Technical Manuals (VTMs), which are reviewed and "Approved for Use" by Nuclear Engineering (NE). Control of the "Approved for Use" VTMs is accomplished by Document Control and Records Management for all site organizations. DCRM establishes a master set of "Approved for Use" VTMs which are updated when affected by design changes and plant modifications. These activities assure that:

Specific components to which each manual applies are identified;

Manuals are complete and up-to-date (by vendor contract when possible);

Information is provided in the manual for the identification of engineering requirements which may be contained within the manual; and

TVA design documents are revised when appropriate to reference or incorporate upgraded vendor technical information in the vendor manual.

2.2 Drawings

Vendor supplied drawings, that provide information to support safety-related plant activities, are maintained in an as-constructed or configuration controlled status independent of the VTMs. These drawings are included in the TVA Drawing Management System and are maintained and controlled in accordance with WBN's drawing control procedures. In instances where copies of vendor drawings are contained in VTMs, the drawings are considered as "information only" copies. Only drawings which are statused as "As Constructed" or "Configuration Control Drawings" are used for safety-related work.

2.3 Reconciliation of Plant Procedures/Instructions

Once "Approved for Use" VTMs are issued and/or revised, they are reviewed by affected plant organizations. These organizations evaluate plant instructions and procedures (e.g., operating procedures, maintenance instructions, inservice test/inspection procedures) and revise them if necessary to incorporate current information.

2.4 Confirmation of Plant Adequacy

The VI CAP provides for the confirmation of the adequacy of the installed configuration for vendor supplied features. Included in this confirmation is a review of WBN activities of the Vertical Slice Review, Special Programs and other CAPs. These reviews determine the extent of verification of vendor requirements under these activities and the components and attributes involved. The review also identifies any problems related to vendor information and the corrective action taken.

Nuclear Engineering analyses the data gathered during this review and identifies those areas/attributes for which plant adequacy is confirmed. Areas/attributes that are nonconforming with vendor engineering data are further analyzed for extent of condition and safety significance. A confirmation process is performed for those areas/attributes related to vendor engineering requirements which are not covered by analysis or other programs. This process includes a review of vendor documents against design input and output requirements. Any conflicts or omissions are identified and analyzed to determine the need to perform a physical confirmation of the adequacy of plant features.

2.5 Identified Inconsistencies

As inconsistencies are identified between vendor technical information in "Approved for Use" VTMs and existing documents or between "Approved for Use" VTMs and installed equipment, Open Item Reports (OIRs) are generated, tracked, and controlled in an open item management system. Inconsistencies requiring a design change document are entered into the WBN design control system and tracked to completion. Hardware modifications are to be implemented as required. If an OIR is determined to be a CAQ, it is tracked and controlled by the CAQ process.

2.6 Recurrence Control

Included as part of this CAP is the establishment of methods to prevent recurrence of deficiencies with vendor information. The controls consist of the development of standards and procedures to improve the control and maintenance of vendor information. Corporate TVA procedures addressing the processing and control of vendor information are being revised and strengthened. Project and site procedures are also being developed to implement the corporate guidance and requirements relative to vendor manuals and other vendor information. A long term enhancement is the preparation of a cross-reference index to be used in the development of design changes. This index will be used to assist in the location of affected vendor documents during design change development.

3.0 CONCLUSION

The staff review of the VI CAP Plan determined that the described plan establishes methods for resolving identified deficiencies with vendor information at WBN, coordination of vendor problems with other WBN CAPs and Special Programs to ensure vendor problems are resolved, confirmation of plant adequacy relative to vendor information, identification of organizational responsibilities for the implementation of the VI CAP, provisions for recurrence control to prevent future problems with vendor information and documentation of results at the completion of the CAP.

While the staff determined that the plan established methods for resolving the issues of the VI CAP, one comment is provided relative to the use of uncontrolled vendor drawings as "information only" copies. The staff finds this to be acceptable as long as TVA ensures that sufficient administrative procedures and personnel training is in place so that WBN personnel are fully aware of the proper use of "information only" drawings.

Additionally, Enclosure 2 to the letter dated March 15, 1990, submitting Revision 3 of the CAP Plan to the NRC defines specific tasks to which TVA has committed. Included as a commitment is the submittal of a revised response to NRC Notice of Violation 50-390/87-05-01 concerning specific instances where WBN failed to comply with vendor information requirements for equipment installed in the plant. NRC review and acceptance of the revised response to the NOV will be handled outside the scope of this CAP.

In conclusion, the staff finds that the VI CAP Plan, as submitted to the NRC, establishes acceptable program guidelines for resolving WBN problems in the area of vendor information. The staff concludes that with proper implementation, the CAP Plan provides reasonable assurance that vendor technical documents for safety-related equipment will be current, complete, and appropriately updated for the life of the plant and that information in these documents will be used as input to TVA design output documents, plant instructions and procedures, and the plant as-built configuration.

4.0 REFERENCES

Volume 4 of the Tennessee Valley Authority Nuclear Performance Plan, dated May 1989.

TVA letter to the NRC dated December 14, 1988, enclosing the WBN Vendor Information Corrective Action Program Plan, Revision 1.

TVA letter to the NRC dated March 15, 1990, enclosing the WBN Vendor Information Corrective Action Program Plan, Revision 3.

Principal Contributor : George Hubbard

Dated: September 11, 1990

APPENDIX J

ELIMINATION OF POSTULATED PRIMARY LOOP PIPE RUPTURES AS A DESIGN BASIS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ELIMINATION OF POSTULATED PRIMARY LOOP PIPE RUPTURES

AS A DESIGN BASIS

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-390 & 50-391

1.0 INTRODUCTION

By letter dated April 17, 1989, Tennessee Valley Authority (the applicant) requested the elimination of the dynamic effects of postulated primary loop pipe ruptures from the design basis of Watts Bar Nuclear Plant Units 1 and 2 using "leak-before-break" (LBB) technology as permitted by the revised General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR Part 50.

The applicant submitted the technical basis for the elimination of primary loop pipe ruptures for Watts Bar Nuclear Plant Units 1 and 2 in Westinghouse report WCAP-11985 (Reference 1). The applicant also referenced Westinghouse reports WCAP-10456 (Reference 2) and WCAP-10931, Revision 1 (Reference 3), which have been reviewed previously by the staff as discussed in References 4 and 5, respectively. By letter dated February 14, 1990, the applicant submitted additional information in Westinghouse Report WCAP-12500 (Reference 6).

The revised GDC-4 is based on the development of advanced fracture mechanics technology using the LBB concept. On October 27, 1987, a final rule was published (52 FR 41288), effective November 27, 1987, amending GDC-4 of Appendix A to 10 CFR Part 50. The revised GDC-4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures in high energy piping in nuclear power units. The new technology reflects an engineering advance which allows simultaneously an increase in safety, reduced worker radiation exposures, and lower construction and maintenance costs. Implementation permits the removal of pipe whip restraints and jet impingement barriers as well as other related changes in operating plants, plants under construction, and future plant designs. Although functional and performance requirements for containments, emergency core cooling systems, and environmental qualification of equipment remain unchanged, local dynamic effects uniquely associated with postulated ruptures in piping which qualified for LBB may be excluded from the design basis (53 FR 11311). The acceptable technical procedures and criteria are defined in NUREG-1061, Volume 3 (Reference 7).

Using the criteria in Reference 7, the staff has reviewed and evaluated the applicant's submittal for compliance with the revised GDC-4. This Safety Evaluation Report provides the staff's findings.

2.0 EVALUATION

2.1 Watts Bar Nuclear Plant Units 1 and 2 Primary Loop Piping

The Watts Bar Nuclear Plant Units 1 and 2 primary loop piping consists of 34-inch, 36-inch, and 32-inch nominal diameter hot leg, cross-over leg, and cold leg, respectively. The piping material in the primary loops is austenitic cast stainless steel (SA-351 CF8A). The piping is centrifugally cast and the fittings are statically cast.

2.2 Staff Evaluation Criteria

The staff's criteria for evaluation of compliance with the revised GDC-4 are discussed in Chapter 5.0 of Reference 7 and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer are not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits, and controls; and resistance of material to various forms of stress corrosion and performance under cyclic loadings.

(3) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe ends; the materials properties including the fracture mechanics parameter "J-integral" (J) resistance (J-R) curve used in the analyses; and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, and maximum crack growth).

(4) A through-wall flaw should be postulated at the highest stressed locations determined from criterion (1) above. The size of the flaw should be large enough so that the leakage is assured of detection with at least a factor of 10 using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(5) It should be demonstrated that the postulated leakage flaw is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be at least 1.4 and should be determined by a flaw stability analysis, i.e., that the leakage-size flaw will not experience unstable crack growth even if larger loads (larger than design loads) are applied. However, the final rule permits a reduction of the margin of 1.4 to 1.0 if the individual normal and seismic (pressure, deadweight, thermal expansion, SSE, and seismic anchor motion) loads are summed absolutely. This analysis should demonstrate that crack growth is stable and the final flaw size is limited, such that a double-ended pipe break will not occur.

(6) The flaw size should be determined by comparing the leakage-size flaw to the critical-size flaw. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 between the leakage-size flaw and the critical-size flaw to account for the uncertainties inherent in the analyses and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

2.3 Staff Evaluation of GDC-4 Compliance

The staff has evaluated the information presented in References 1 and 6 for compliance with the revised GDC-4. Furthermore, the staff performed independent flaw stability computations using an elastic-plastic fracture mechanics procedure developed by the staff (Reference 8).

On the basis of its review, the staff finds the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping in compliance with the revised GDC-4. The following paragraphs in this section present the staff's evaluation.

(1) Normal operating loads, including pressure, deadweight, and thermal expansion, were used to determine leak rate and leakage-size flaws. The flaw stability analyses performed to assess margins against pipe rupture at postulated faulted load conditions were based on normal plus SSE loads. In the stability analysis, the individual normal and seismic loads were summed absolutely. In the leak rate analysis, the individual normal load components were summed algebraically. Leak-before-break evaluations were performed for the limiting location in the piping.

(2) For Westinghouse facilities, there is no history of cracking failure in reactor coolant system (RCS) primary loop piping. The RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 450 reactor-years, including 5 plants each having over 17 years of operation and 15 other plants each with over 12 years of operation.

(3) The material tensile and fracture toughness properties were provided in Reference 1. Because the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping consists of cast stainless steel, the thermal aging toughness properties of cast stainless steel materials were estimated according to procedures in References 2 and 3. The material tensile properties were estimated using plant specific material certifications and generic procedures. For flaw stability evaluations, the lower-bound stress-strain properties were used. For leakage rate evaluations, the average stress-strain properties were used.

(4) Watts Bar Nuclear Plant Units 1 and 2 have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 such that a leakage of one gallon per minute (gpm) in one hour can be detected. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems; the margin is a factor of 10 on leakage and is consistent with the guidelines of Reference 7.

(5) In the flaw stability analyses, the staff evaluated the margin in terms of load for the leakage-size flaw under normal plus SSE loads. The staff's calculations indicated the margin exceeded 1.0 when the individual normal and seismic loads were summed absolutely. The margin is consistent with the guidelines of the final rule.

(6) Similar to item (5) above, the margin between the leakage-size flaw and the critical-size flaw was also evaluated in the flaw stability analyses. The staff's calculations indicated the margin in terms of flaw size exceeded 2 for the load combination method considered. The margin is consistent with the guidelines of Reference 7.

3.0 CONCLUSION

The staff has reviewed the information submitted by the applicant and has performed independent flaw stability computations. On the basis of its review, the staff concludes that the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping complies with the revised GDC-4 according to the criteria in NUREG-1061, Volume 3 (Reference 7). Thus, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of Watts Bar Nuclear Plant Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks need not be a design basis.

4.0 REFERENCES

(1) Westinghouse Report WCAP-11985, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Watts Bar Units 1 & 2", November 1988, Westinghouse Proprietary Class 2.

(2) Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems", November 1983, Westinghouse Proprietary Class 2.

(3) Westinghouse Report WCAP-10931, Revision 1, "Toughness Criteria for Thermally Aged Cast Stainless Steel", July 1986, Westinghouse Proprietary Class 2.

(4) Letter from B. J. Youngblood of NRC to M. D. Spence of Texas Utilities Generating Company dated August 28, 1984.

(5) Letter from D. C. DiIanni of NRC to D. M. Musolf of Northern States Power Company dated December 22, 1986.

(6) Westinghouse Report WCAP-12500, "Additional Information in Support of Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Watts Bar Units 1 and 2", January 1990, Westinghouse Proprietary Class 3.

(7) NUREG-1061, Volume 3, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks", November 1984.

(8) NUREG/CR-4572, "NRC Leak-Before-Break (LBB.NRC) Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads", May 1986.

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10. SUPPLEMENTARY NOTES

Docket Nos. 50-390 and 50-391

11. ABSTRACT *(200 words or less)*

Supplement No. 5 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of (1) additional information submitted by the applicant since Supplement No. 4 was issued, and (2) matters that the staff had under review when Supplement No. 4 was issued.

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