NUREG-0847
Supplement No. 14

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

December 1994

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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), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), Supplement No. 6 (April 1991), Supplement No. 7 (September 1991), Supplement No. 8 (January 1992), Supplement No. 9 (June 1992), Supplement No. 10 (October 1992), Supplement No. 11 (April 1993), Supplement No. 12 (October 1993), and Supplement No. 13 (April 1994), 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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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AMSAC	ATWS (anticipated transient without scram) mitigating system actuation circuitry
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATI	acceptance test instruction
CAP	corrective action program
CFR	Code of Federal Regulations
CHT	cold hydrostatic test
CIV	containment isolation valve
CNPP	Corporate Nuclear Performance Plan
CSST	common station service transformer
CT	component test
CVCS	chemical and volume control system
CVN	Charpy V-notch
DAC	derived airborne concentration
DBA	design-basis accident
DC	design criterion
DG	diesel generator
DG	draft regulatory guide
DNBR	departure from nucleate boiling ratio
EDG	emergency diesel generator
EHC	electrohydraulic control
EOL	end of life
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERCW	essential raw cooling water
ERGs	Emergency Response Guidelines
ESF	engineered safety feature
FF	failed fuel
FFD	failed fuel detector
FSAR	final safety analysis report
GDC	general design criterion
GFFDS	gross failed fuel detection system
GL	generic letter
HFT	hot functional test
IEEE	Institute of Electrical and Electronics Engineers
ISO	International Standards Oganization

IST	inservice testing
ITP	Initial Test Program
JTG	Joint Test Group
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LP	low pressure
LTOP	low-temperature overpressure protection
MFLB	main feedwater line break
MSIV	main steam isolation valve
MSLB	main steamline break
MSS	median signal selector
MSSV	main steam safety valve
MSV	main steam valve
MSVV	main steam valve vault
NASA	National Aeronautics and Space Administration
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
O&M	operations and maintenance
OBE	operating basis earthquake
ODCM	Offsite Dose Calculation Manual
OL	operating license
PORV	pilot-operated relief valve
QA	quality assurance
RAI	request for additional information
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RP	regulatory position
RPS	reactor protection system
RPV	reactor pressure vessel
RSs	response spectra
RTD	resistance temperature detector
RWST	refueling water storage tank
SCV	steel containment vessel
SD	standard drawing
SER	safety evaluation report
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal
SP	special program
SPT	special performance test
SRP	standard review plan

SSER	supplement to safety evaluation report
SSI	soil-structure interaction
TAC	technical assignment control
TEDE	total effective dose equivalent
TI	temporary instruction
TSs	Technical Specifications
TTD	trip time delay
TVA	Tennessee Vally Authority
USE	upper shelf energy
USST	unit station service transformer
WBNPP	Watts Bar Nuclear Performance Plan
WGFE	Working Group on Flaw Evaluation
WISP	Workload Information and Scheduling Program

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 SER Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985) Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), Supplement No. 7 (SSER 7, September 1991), Supplement No. 8 (SSER 8, January 1992), Supplement No. 9 (SSER 9, June 1992), Supplement No. 10 (SSER 10, October 1992), Supplement No. 11 (SSER 11, April 1993), Supplement No. 12 (October 1993), and Supplement No. 13 (SSER 13, April 1994). As of this date, the staff has completed its review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 87.

The SER and its supplements were written to agree with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues raised by the SRP review that were not closed out when the SER was published were classified into outstanding issues, confirmatory issues, and proposed license conditions (see Sections 1.7, 1.8, and 1.9, respectively, which follow).

In addition to the guidance in the SRP, the staff would 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 implementation status was addressed in Section 1.14 of SSER 6. The staff is reevaluating the status of implementation of all bulletins and generic letters.

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER, unless otherwise noted. Accordingly, Appendix A continues the chronology of the safety review. Appendix E lists principal contributors to this supplement. The other appendices are not changed by this supplement.

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1.7 Summary of Outstanding Issues

In SER Section 1.7, the staff listed 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SER supplements that followed. In this section, the staff updates the status of those items. The completion status of each of the issues is tabulated below with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status information for still-unresolved issues is conveyed in the staff's summaries of the monthly licensing status meetings.

Issue[1]	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 M74801)	Resolved (SSER 14)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC M71919)	Resolved (SSER 9)	3.10
(b) Environmental (TAC M63591)	Under review (SER)	3.11
(5) Preservice inspection program (TAC M63627)	Resolved for Unit 1 (SSERs 10 and 12)	5.2.4, 6.6, App. Z
(6) Pressure-temperature limits for Unit 2	On hold	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) Branch Technical Position CSB 6-4	Resolved (SSER 3)	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 M63649)	Resolved (SSER 13)	8.2
(12) Fire-protection program (TAC M63648)	Under review (SER)	9.5.1

[1]The TAC (technical assignment control) number that appears in parentheses after the issue title is an internal NRC control number by which the issue is managed through the Workload Information and Scheduling Program (WISP) and by which relevant documents are filed. Documents associated with each TAC number can be located by the NRC document control system, NUDOCS/AD.

Issue	Status	Section
(13) Quality classification of diesel generator auxiliary system piping and components (TAC M63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC M63638)	Resolved (SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC M63657)	Under review (SER)	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC M76972)	Resolved (SSER 13)	17
(18) Seismic classification of cable trays and conduit (TACs R00508, R00516)	Resolved (SSER 8)	3.2.1, 3.10
(19) Seismic design concerns (TACs M79717, M80346):		
(a) Number of OBE events	Resolved (SSER 8)	3.7.3
(b) 1.2 multi-mode factor	Resolved (SSER 9)	3.7.3
(c) Code usage	Resolved (SSER 8)	3.7.3
(d) Conduit damping values	Resolved (SSER 8)	3.7.3
(e) Worst case, critical case, bounding calculations	Resolved (SSER 12)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Resolved (SSER 11)	3.7.2.12
(h) Category 1(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Resolved (SSER 9)	3.8
(k) Update FSAR per 12/18/90 letter	Resolved (SSER 8)	3.7
(20) Mechanical systems and components (TACs M79718, M80345)		
(a) Feedwater check valve slam	Resolved (SSER 13)	3.9.1
(b) New support stiffness and deflection limits	Resolved (SSER 8)	3.9.3.4
(21) Removal of RTD bypass system (TAC M63599)	Resolved (SSER 8)	4.4.3
(22) Removal of upper head injection system (TAC M77195)	Resolved (SSER 7)	6.3.1
(23) Containment isolation using closed systems (TAC M63597)	Resolved (SSER 12)	6.2.4
(24) Main steamline break outside containment (TAC M63632)	Resolved (SSER 14)	3.6.1

Issue	Status	Section
(25) Health Physics Program (TAC M63647)	Resolved (SSER 10)	12
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TACs M77550, M77551)	Resolved (SSER 9)	7.5.2
(27) Containment sump screen design anomalies (TAC M77845)	Resolved (SSER 9)	6.3.3
(28) Emergency procedure (TAC M77861)	Resolved (SSER 9)	13.5.2.1

1.8 Summary of Confirmatory Issues

In SER Section 1.8, the staff listed 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. Issue 43 was added in SSER 6. In this section, the staff 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 completion status of each of the issues is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status information for still-unresolved issues is conveyed in the staff's summaries of the monthly licensing status meetings.

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 M63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC M63618)	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

Issue	Status	Section
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC M63625)	Resolved (SSER 8)	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)	4.4.5
(14) Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15) Natural circulation tests (TACs M63603, M79317, M79318)	Resolved (SSER 10)	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 M63623)	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
(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 M63649)	Resolved (SSER 13)	8.3.1.2

Issue	Status	Section
(29) Diesel generator reliability qualification testing (TAC M63649)	Resolved (SSER 7)	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) Update FSAR on sharing of dc and ac distribution systems (TAC M63649)	Resolved (SSER 13)	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 M63649)	Resolved (SSER 7)	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC M63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC M63648)	Under review (SER)	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC M63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC M63639)	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 M63649)	Resolved (SSER 13)	8.3.3.1.1
(43) Safety parameter display system (TAC M73723)	Updated (SSER 6)	18.2, App. P

1.9 Summary of Proposed License Conditions

In Section 1.9 of the SER and in SSERs that followed, the staff listed 43 proposed 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 completion status of the proposed license conditions is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status of still-unresolved issues is conveyed in the staff's summaries of the monthly licensing status meetings.

Proposed 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 M74801)	Resolved (SSER 12)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TACs M77132, M77133)	Resolved (SSER 10)	4.4.8
(4) Inservice Inspection Program (TAC M76881)	Resolved (SSER 12)	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 M63645)	Resolved (SSER 5)	11.7.1
(b) Iodine particulate sampling (TAC M63645)	Resolved (SSER 6)	11.7.1
(c) High-range in-containment radiation monitor (TAC M63645)	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 M63622)	Resolved (SSER 5)	6.2.4
(8) Containment isolation dependability (II.E.4.2) (TAC M63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC M77208)	Resolved (SSER 8)	6.2.5, App. C
(10) Status monitoring system/BISI (TACs M77136, M77137)	Resolved (SSER 7)	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
(13) DC monitoring and annunciation (TAC M63649)	Resolved (SSER 13)	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

Proposed Condition	Status	Section
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC M63649)	Resolved (SSER 7)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (TAC M77543)	Resolved (SSER 14)	9.3.2
(20) Fire protection program (TAC M63648)	Under review (SER)	9.5.1.8
(21) Performance testing for communications systems (TAC M63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC M63640)	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) (TACs M63646, M77553)	Resolved (SSER 10)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC M63592)	Resolved (SSER 8)	13.4
(26) Use of experienced personnel during startup (TAC M63592)	Resolved (SSER 8)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC M63656)	Resolved (SSER 13)	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC M77861)	Resolved (SSER 10)	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC M77861)	Resolved (SSER 10)	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 M79872)	Resolved (SSER 7)	14.2
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1

Proposed Condition	Status	Section
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC M63631)	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 M77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC M63655)	Updated (SSER 6)	18.1
(38) Physical Security Plan (TACs M63657, M83973)	Resolved (SSER 10)	13.6.4
(39) Control of heavy loads (NUREG-0612) (TAC M77560)	Resolved (SSER 13)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC M64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC M77569)	Resolved (SSER 14)	15.4.3
(42) Loose-parts monitoring system (TAC M77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC M73723)	Opened (SSER 5)	18.2

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) (TAC M63615)

(2) Criticality monitor (Section 9.1, SSER 5) (TAC M63615)

The staff has reevaluated three technical issues previously approved for exemption from various provisions of Appendix G to 10 CFR Part 50 in SSER 14. As a result, Section 5.3.1.1 of SSER 14 reports that these exemptions are no longer needed.

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 concerning the overall management of its nuclear program as well as on 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 concerning 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 (July 1987), and NUREG-1232, Vol. 4 (January 1990).

In a letter of September 6, 1991, the applicant submitted Revision 1 of the WBNPP. In SSER 9, the staff concluded that Revision 1 of the WBNPP does not necessitate any revision of the staff's safety evaluation report, NUREG-1232, Vol. 4.

In NUREG-1232, Vol. 4, the staff documented its general review 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; accordingly, the staff prepared Temporary Instructions (TIs) 2512/016-043 for the Inspection Manual and adhered to the TIs to perform inspections of the CAPs and SPs. This new section was introduced in SSER 5 and will be updated in subsequent SSERs. The current status of all CAPs and SPs follows. The status described here fully supersedes that described in previous SSERs.

1.13.1 Corrective Action Programs

(1) Cable Issues (TAC M71917; TI 2512/016)

Program review status:	Complete: NUREG-1232, Vol. 4; Letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation was reproduced in SSER 7 as Appendix P); supplemental safety evaluation dated April 24, 1992 (Appendix T of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), February 14, 1994.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-01 (March 17, 1992); audit report of June 12, 1992 (Appendix Y of SSER 9); 50-390, 391/92-05

(April 17, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-18 (August 14, 1992); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-30 (November 13, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-10 (March 19, 1993); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-48 (August 13, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-85 (January 14, 1994); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-35 (June 20, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-51 (August 11, 1994); 50-390, 391/94-53 (September 20, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/94-61 (October 12, 1994); to come.

(2) <u>Cable Tray and Tray Supports (TAC R00516; TI 2512/017)</u>

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by December 1994.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); to come.

(3) <u>Design Baseline and Verification Program (TAC M63594; TI 2512/019)</u>

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

Implementation status: Full implementation expected by December 1994.

NRC inspections: Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20; (September 25, 1990); 50-390/91-201 (March 22, 1991); 50-390, 391/91-20 (October 8, 1991); 50-390, 391/91-25 (December 13, 1991); 50-390, 391/92-06 (April 3, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-29 (May 14, 1993); 50-390, 391/93-66 (October 29, 1993); 50-390, 391/94-69 (November 18, 1994); to come.

(4) Electrical Conduit and Conduit Support (TAC R00508; TI 2512/018)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by December 1994.
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); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-32 (May 16, 1994); to come.

(5) Electrical Issues (TAC M74502; TI 2512/020)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/90-30 (February 25, 1991); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-31 (May 11, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-53 (September 20, 1994); to come.

(6) Equipment Seismic Qualification (TAC M71919; TI 2512/021)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.10.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February

19, 1993); 50-390, 391/93-79 (March 4, 1994); to come.

(7) Fire Protection (TAC M63648; TI 2512/022)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-63 (November 2, 1994); 50-390, 391/94-62 (November 16, 1994); to come.

(8) Hanger and Analysis Update Program (TAC R00512; TI 2512/023)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by December 1994.
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); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-45 (July 20, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994); to come.

(9) Heat Code Traceability (TAC M71920; TI 2512/024)

Program review status:	Complete: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991.
Implementation status:	100% (certified by letter, E. Wallace (TVA) to NRC, July 31, 1990); staff concurrence in SSER 7, Section 3.2.2.
NRC inspections:	Complete: 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; TI 2512/025)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-08 (May 15, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-08 (March 11, 1994); to come.

(11) Instrument Lines (TAC M71918; TI 2512/026)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), October 26, 1990 (Appendix K of SSER 6) and May 5, 1994.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/90-29 (January 29, 1991); 50-390, 391/91-02 (March 6, 1991); 50-390, 391/91-03 (April 15, 1991); 50-390, 391/91-26 (December 6, 1991); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-24 (July 1, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994), to come.

(12) Prestart Test Program (TAC M71924)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 27, 1991.
Implementation status:	Withdrawn by letter (J. H. Garrity (TVA) to NRC, February 13, 1992). Applicant will re-perform preoperational test program per Regulatory Guide 1.68, Revision 2.

(13) Quality Assurance Records (TAC M71923; TI 2512/028)

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

4; letter, P. S. Tam (NRC) to M. O. Medford (TVA) June 9, 1992 (Appendix X of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), January 12, 1993; letter, F. J. Hebdon (NRC) to M. O. Medford (TVA), August 12, 1993; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), April 25, 1994.

Implementation status: 100% (certified by letter, W. J. Museler (TVA), to NRC, April 27, 1994); staff concurrence in Inspection Report 50-390, 391/94-40 (June 24, 1994).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990); 50390, 391/91-08 (May 30, 1991); 50-390, 391/91-15 (September 5, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-10 (June 11, 1992); 50-390, 391/92-21 (September 18, 1992); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-21 (April 9, 1993); 50-390, 391/93-29 (May 14, 1993); 50-390, 391/93-34 (July 5, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-50 (September 3, 1993); 50-390, 391/93-59 (October 25, 1993); 50-390, 391/93-69 (November 12, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-78 (December 16, 1993); 50-390, 391/93-86 (January 24, 1994); 50-390, 391/94-04 (February 23, 1994); 50-390, 391/94-09 (March 11, 1994); 50-390, 391/94-17 (April 1, 1994); 50-390, 391/94-28 (May 5, 1994); 50-390, 391/94-40 (June 24, 1994) .

(14) Q-List (TAC M63590; TI 2512/029)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; letters, P. S. Tam (NRC) to O. D. Kingsley (TVA), January 23, 1991 and March 17, 1994 (enclosure of this letter reproduced as Appendix AA in SSER 13).

Implementation status: 100% (certified by letter, W. J. Museler (TVA), to NRC, January 28, 1994); staff concurrence in Inspection Report 50-390, 391/94-27 (April 21, 1994).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/93-20 (April 16, 1993); 50-390, 391/93-68 (November 12, 1993); 50-390, 391/94-27 (April 21, 1994).

(15) Replacement Items Program (TAC M71922; TI 2512/027)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol.

	4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991 (Appendix N of SSER 6); letter, P. S. Tam (NRC) to M. O. Medford (TVA), July 27, 1992, and April 5, 1994.
Implementation status:	Full implementation expected by November 1994.
NRC inspections:	Inspection Reports 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-03 (March 16, 1992); 50-390, 391/92-11 (June 12, 1992); 50-390, 391/92-17 (July 22, 1992); 50-390, 391/92-21 (September 18, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-22 (April 25, 1993); 50-390, 391/93-34 (July 9, 1993); 50-390, 391/93-38 (June 24, 1993); to come.

(16) <u>Seismic Analysis (TAC R00514; TI 2512/030)</u>

Program review status:	Complete: Letters, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7 and October 31, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7.
Implementation status:	100% (certified by letter, J. H. Garrity (TVA) to NRC, December 2, 1991); staff concurrence in SSER 9, Section 3.7.1.
NRC inspections:	Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); audit report by L. B. Marsh, October 10, 1990.

(16)(a) <u>Civil Calculation Program (TAC R00514)</u>

Program review status:	No program review. A number of civil calculation categories are required by the Design Baseline and Verification Program CAP and constitute parts of the applicant's corrective actions. This program is regarded as complementary to but not part of the Seismic Analysis CAP. Staff efforts consist mainly of audits performed at the site and in the office.
Implementation status:	100%. Final calculations transmitted by letter, W. J. Museler (TVA) to NRC, July 27, 1992.
NRC audits:	Complete: Memorandum (publicly available), T. M. Cheng (NRC) to P. S. Tam, January 23, 1992; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; letters, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992 and July 2, 1993; 50-390, 391/93-07 (February 19, 1993); letter, P. S. Tam (NRC) to M. O. Medford (TVA), November 26, 1993.

(17) Vendor Information Program (TAC M71921; TI 2512/031)

Program review status: Complete: Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (Appendix I of SSER 5); Appendix I of SSER 11.

Implementation status: Full implementation expected by January 1995.

NRC inspections: Inspection Reports 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/93-27 (May 14, 1993); to come.

(18) Welding (TAC M72106; TI 2512/032)

Program review status: Complete: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 5, 1991.

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, January 9, 1993); staff concurrence to come.

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); 50-390, 391/91-05 (May 28, 1991); 50-390, 391/91-18 (October 8, 1991); 50-390, 391/91-23 (November 21, 1991); 50-390, 391/91-32 (February 10, 1992); 50-390, 391/92-20 (August 12, 1992); 50-390, 391/92-28 (October 9, 1992); 50-390, 391/93-02 (February 2, 1993); 50-390, 391/93-19 (March 15, 1993); 50-390, 391/93-38 (June 24, 1993); 50-390, 391/93-84 (December 21, 1993); 50-390, 391/94-05 (February 19, 1994); 50-390, 391/94-16 (March 15, 1994); 50-390, 391/94-49 (July 21, 1994); to come.

1.13.2 Special Programs

(1) Concrete Quality (TAC M63596; TI 2512/033)

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

Implementation status: 100% (certified by letter, E. Wallace (TVA) to NRC, August 31, 1990); staff concurrence in SSER 7, Section 3.8.2.1.

NRC inspections: Complete: NUREG-1232, Vol. 4; Inspection Reports 50-390, 391/89-200 (December 12, 1989); 50-390, 391/90-26 (January 8, 1991).

(2) Containment Cooling (TAC M77284; TI 2512/034)

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 21, 1991 (Section 6.2.2 of SSER 7).

Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, December 30, 1993); staff concurrence to come.
NRC inspections:	Inspection Report 50-390, 391/93-56 (September 20, 1993); to come.

(3) Detailed Control Room Design Review (TAC M63655; TI 2512/035)

Program review status:	Complete: Appendix D of SER; NUREG-1232, Vol. 4; Section 18.1, and Appendix L of SSER 6.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/94-22 (April 28, 1994); to come.

(4) Environmental Qualification Program (TAC M63591; TI 2512/036)

Program review status:	NUREG-1232, Vol. 4; review in progress, results will be published in Section 3.11 of a future SSER.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/93-63 (October 18, 1993; 50-390, 391/94-28 (April 18, 1994); to come.

(5) Master Fuse List (TAC M76973; TI 2512/037)

Program review status:	Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 30, 1992 (Appendix U of SSER 9).
Implementation status:	100% (certified by letter, W. Museler (TVA) to NRC, April 2, 1993); staff concurrence in Inspection Report 50-390, 391/93-31 (May 6, 1993).
NRC inspections:	Complete: Inspection Reports 50-390, 391/86-24 (February 12, 1987); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-27 (September 25, 1992); 50-390, 391/93-31 (May 6, 1993).

(6) Mechanical Equipment Qualification (TAC M76974; TI 2512/038)

Program review status:	NUREG-1232, Vol. 4; review in progress, results to be published in Section 3.11 of a future SSER.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	To come.

(7) Microbiologically Induced Corrosion (TAC M63650; TI 2512/039)

Program review status:	Complete: NUREG-1232, Vol. 4; Appendix Q of SSER 8; Appendix Q of SSER 10.

Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, August 31, 1993); staff concurrence in Inspection Report 50-390, 391/93-67 (November 1, 1993).
NRC inspections:	Complete: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); 50-390, 391/93-01 (February 25, 1993); 50-390, 391/93-09 (March 26, 1993); 50-390, 391/93-67 (November 1, 1993).

(8) Moderate Energy Line Break Flooding (TAC M63595; TI 2512/040)

Program review status:	Complete: NUREG-1232, Vol. 4; Section 3.6 of SSER 11.
Implementation status:	Full implementation expected by October 1994.
NRC inspections:	Inspection Reports 50-390, 391/93-85 (January 14, 1994); to come.

(9) Radiation Monitoring Program (TAC M76975; TI 2512/041)

Program review status:	Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Chapter 12 of the SER and SSERs.
Implementation status:	Full implementation expected by December 1994.
NRC inspections:	Inspection Reports 50-390, 391/94-56 (October 6, 1994); to come.

(10) Soil Liquefaction (TAC M77548; TI 2512/042)

Program review status:	Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 19, 1992; Section 2.5 of SSER 9.
Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, July 27, 1992); staff concurrence in SSER 11, Section 2.5.4.4.
NRC inspections:	Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-03 (May 11, 1989); audit report by L. B. Marsh (NRC) (October 10, 1990); audit report, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; audit report, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992; 50-390, 391/92-45 (February 17, 1993).

(11) Use-as-Is CAQs (TAC M77549; TI 2512/043)

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

Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, July 24, 1992); staff concurrence in Inspection Report 50-390, 391/93-10 (March 19, 1993).
NRC inspections:	Complete: Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/93-10 (March 19, 1993).

2 SITE CHARACTERISTICS

2.3 Meteorology

In the staff's original Safety Evaluation Report (SER, 1982) and in previous supplements, the staff reviewed and accepted the meteorology program at Watts Bar. By Final Safety Analysis Report (FSAR) Amendment 83, the applicant updated information on portions of the meteorology program. The staff's review is documented in the material that follows and was tracked by TACs M88696 and M88697.

2.3.4 Short-Term (Accident) Diffusion Estimates

In FSAR Section 2.3.4.1, the applicant added a cumulative probability distribution of χ/Q for appropriate distance (e.g., the exclusion area boundary distance) and time periods as specified in Section 2.3.4.2 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." This change is thus acceptable.

In Chapter 15 of a future SER supplement (SSER), the staff will report its evaluation of the radiological consequences of accidents, using new χ/Q values.

2.3.5 Long-Term (Routine) Diffusion Estimates

The applicant deleted the χ/Qs and D/Q values and the respective calculation methodologies and relocated them to the Offsite Dose Calculation Manual (ODCM) for Watts Bar. The staff has completed its review of the Watts Bar ODCM under TAC M77553 and reported the results in a letter to the applicant dated July 26, 1994.

The changes in this section comply with relevant requirements of 10 CFR Part 100, "Reactor Site Criteria," and relevant guidance of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors," and are thus acceptable.

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

3.2 Classification of Structures, Systems, and Components

By FSAR Amendment 86, the applicant modified Table 3.2-2, "Summary of Criteria - Mechanical System Components," to indicate that its program for complying with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," is applicable to the emergency diesel fuel oil system transfer pumps and 7-day fuel oil tanks. Since these components had previously been included in the applicant's quality assurance program, the revision is editorial and acceptable.

The applicant modified Table 3.2-2a, "Classification of Systems Having Major Design Concerns Related to a Primary Safety Function," to reflect the use of previously approved (SSER 12, Appendix Z) alternative examining technique for inaccessible portions of the essential raw cooling water system. This is thus an editorial change and is acceptable.

The applicant added the main generator lube oil system to Table 3.2-2b, "Classification of Systems Not Having Major Design Concerns Related to a Primary Safety Function." The main generator lube oil system had been inadvertently omitted from Table 3.2-2b. This change corrects the error and is acceptable.

The applicant added American Society of Mechanical Engineers (ASME) Code Case N-514 to Table 3.2-7, "Code Cases and Provisions of Later Code Editions and Addenda Used by TVA for Design and Fabrication." Code Case N-514 gives an alternative analysis for the low-temperature overpressure protection (LTOP) system. The staff is reviewing use of Code Case N-514 under TAC M89048; it will report the results in Section 5.3.2 of a future SSER.

The applicant added ASME Code Section III, 1980 Edition, Summer 1980 Addenda, Subsection NCA 1273 to Table 3.2-7. Subsection NCA 1273 contains an exception that "orifice plates not exceeding 1/2 in. nominal thickness which are clamped between flanges and used only in flow measuring service" are not "considered to be a piping subassembly, part, appurtenance, component, or material in accordance with the rules of the Section." The staff reviewed this revision and found it acceptable.

The applicant added ASME Code Section III, 1977 Edition, Subsection NB 2538.4 to Table 3.2-7. Subsection NB 2538.4 allows areas that were ground to remove oxide scale or other mechanically caused depressions to be exempted from penetrant testing or magnetic particle testing. The staff reviewed this revision and found it acceptable.

The staff tracked its review by TACs M89217 and M89218.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.2 Internally Generated Missiles (Inside Containment)

In Section 3.5.1.2 of the SER, the staff identified a number of credible missile sources from high-energy systems inside containment. One source of missiles identified was the pressure sensors in the reactor coolant system (RCS) and associated instrument wells. As a matter of clarification, the only missile source associated with the RCS pressure instrumentation is the pressurizer instrument well. This was clarified in FSAR Amendment 79 with additional information provided by the applicant in a letter dated August 18, 1994. The staff has reviewed the FSAR as amended and the additional information, and concludes that the evaluation and conclusions in the SER are still valid.

This review was tracked by TACs M88488 and M88489.

3.5.1.3 Turbine Missiles

By FSAR Amendments 79 and 86, the applicant provided revised information on this issue. The staff's evaluation follows.

Although large steam turbines and their auxiliaries are not safety-related systems, failures that occur in these turbines can produce large, high-energy missiles. If such missiles were to strike and damage plant safety-related systems, they could render the safety systems unavailable to perform their function. Consequently, General Design Criterion 4 of Appendix A to 10 CFR Part 50 requires, in part, that structures, systems, and components important to safety be appropriately protected against the effects of missiles that might result from such failures. The staff's evaluation of the effects of turbine missiles is based on Regulatory Guide (RG) 1.115, "Protection Against Low-Trajectory Turbine Missiles," and Standard Review Plan (SRP) Sections 2.2.3, 10.2, 10.2.3, and 3.5.1.3. According to RG 1.115, the probability of unacceptable damage from turbine missiles should be less than or equal to 1.0E-7 per year. To satisfy this damage probability, the staff has recommended that utilities satisfy specific turbine-missile generation probabilities for favorably oriented turbines such as the Watts Bar turbines (see Table 3.1).

The main steam turbines (model number BB 281) were manufactured by Westinghouse. The turbine set consists of a double-flow, high-pressure turbine and three double-flow, low-pressure (LP) turbines with a rated speed of 1800 revolutions per minute. The LP discs and shaft are made of nickel-chromium-molybdenum-vanadium alloy steel. The outer cylinder and the two inner cylinders are fabricated of American Society for Testing and Materials (ASTM) 515-GR65 material. There are 10 shrunk-on discs in an LP rotor. The turbines in both units are favorably oriented relative to their respective reactor buildings.

Table 3.1 Turbine System Reliability Criteria

Turbine missile generation probability per year, P_1, for favorably oriented turbine	Required licensee action
(A) $P_1 < 1.0E-4$	This is the general, minimum reliability requirement for loading and bringing the turbine on line.
(B) $1.0E-4 < P_1 < 1.0E-3$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service.
(C) $1.0E-3 < P_1 < 1.0E-2$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion before returning the turbine to service.
(D) $1.0E-2 < P_1$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion before returning the turbine to service.

The turbine speed and load are controlled by the electrohydraulic control (EHC) system, controlling the main stop, governing, intercept, and reheat stop valves. Turbine overspeed protection is provided by an electrical overspeed governor, backed up by a mechanical overspeed governor. The overspeed protection controller will close the governor and intercept valves when the speed exceeds 103 percent of rated speed. If the controller fails to function and the turbine speed increases to 110 percent of rated speed, the mechanical overspeed mechanism will close all steam valves. This speed is still within the design basis because the rotors are designed and shop tested at 120 percent of rated speed. The turbines are also protected by trip logic in case of low condenser vacuum, excessive shaft vibration, abnormal thrust bearing wear, low bearing oil pressure, or transients. Additional devices protect against high temperature and pressure in the exhaust hood.

The applicant's turbine missile analyses were based on the Westinghouse methodology. Before 1980, missile generation caused by destructive speed and fatigue was considered in the Westinghouse methodology. When stress-corrosion cracks were found in LP turbine discs in the keyways and bores in late 1979, Westinghouse revised its methodology to include missile generation resulting from stress-corrosion cracking. The staff approved the revised methodology in the 1980s (letter, C. E. Rossi to J. A. Martin (Westinghouse), February 2, 1987, "Approval for Referencing of Licensing Topical Reports WSTG-1-P, WSTG-2-P, and WSTG-3-P").

The probability of missile generation from stress corrosion of an LP turbine rotor disc is related to the inspection intervals, critical crack size, maximum crack growth rate, and initial crack size. On the basis of the staff-approved methodology, the applicant determined a missile generation probability of 7.0E-5 per year for each LP turbine resulting from stress-corrosion cracking. This probability is more conservative (several orders of magnitude higher) than the probabilities resulting from destructive speed and fatigue. The staff determined that the probability of 7.0E-5 per year is acceptable because it is within the staff-required probability of 1.0E-4 per year for a favorably oriented turbine.

In the destructive speed analysis, the applicant calculated the bursting speed of each shrunk-on disc, assuming that the disc will fail when the average tangential stress equals the maximum tensile strength (adjusted for temperature) of the disc materials. Disc No. 2 is the most highly stressed disc with a calculated failure speed of 190 percent of rated speed. In the analysis, the applicant considered a disc fractured in 90°, 120°, and 180° segments and the results showed that the 90° fragments had the greatest impact as external missiles. The applicant considered the following structures and systems for possible missile targets: two reactor buildings, the main control room, the spent fuel pool, main steam valve rooms, and the essential raw cooling water (ERCW) system. In the original SER, the staff reviewed the possible missile targets and concluded that the ERCW conduit run is the only safety-related system located within the low-trajectory turbine missile strike zones. However, the ERCW conduit run is adequately protected by the concrete turbine pedestal, turbine deck floor, and turbine building walls.

In FSAR Section 10.2.3.6.1, the applicant committed to inservice inspection of LP turbine discs based on Westinghouse recommendations that conform to NRC criteria. Westinghouse will use an ultrasonic inservice inspection method to examine disc bore and keyway surfaces. The inspection intervals will vary from 3.34 to 4.65 operating years for the initial rotors on the basis of Westinghouse recommendations. When the initial rotors are replaced or refurbished, the discs will be inspected either every 5 operating years or at an interval recommended by Westinghouse on the basis of the NRC criteria, whichever results in the shorter inspection interval. If significant corrosion or cracks are found in the discs, the applicant committed to consult with Westinghouse to adjust the inspection schedule to shorter intervals.

The staff concludes that the impact of potential turbine missiles on public health and safety is insignificant because (1) the applicant has demonstrated that the missile generation probability of the LP turbine is acceptably low and within the NRC-specified criteria and (2) the applicant has committed to the inspection schedules for the low-pressure turbine that were recommended by the turbine manufacturer and were calculated on the basis of an NRC-approved methodology.

The staff tracked its review by TACs M88488, M88489, M89217, and M89218.

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

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Watts Bar Units 1 and 2 were designed to withstand a postulated unisolable main steamline break (MSLB) in the main steam valve vault (MSVV) rooms, which are located outside and adjacent to the containment. Each unit has two (north and south) MSVV rooms housing electrical equipment that is required to operate in the resulting high temperatures in the event of such a break. The initial Westinghouse mass and energy data did not take into account the effects of superheated steam after the steam generator tube is uncovered. The inclusion of superheated steam results in a higher peak MSVV room temperature than originally predicted by the initial Westinghouse analysis. The higher temperature could affect the environmental qualification of Class 1E electrical equipment and valve vault structural steel. This was identified as Outstanding Issue 24 in SSER 7. By letter dated November 30, 1992, the applicant submitted the results of a new evaluation that takes into account the increased environmental temperature in the MSVV rooms that could occur as a result of the release of superheated steam. By letter dated July 13, 1993, the staff requested additional information related to the assumptions used in the latest analysis. By letter dated March 28, 1994, the applicant submitted this additional clarifying information.

In the new evaluation, the applicant uses the same methodology as that used to resolve the same issue at the Sequoyah Nuclear Plant. The staff's detailed evaluation and acceptance of this methodology can be found in Section 3.2.2 of NUREG-1232, Volume 2, "Safety Evaluation Report on Tennessee Valley Authority: Sequoyah Nuclear Performance Plan," dated May 1988. In its letter of November 30, 1992, the applicant identified plant-specific information to show that the safety-related equipment (equipment necessary to mitigate the consequences of the MSLB) in the MSVV rooms was qualified and would perform its safety function.

The staff reviewed the information specific to Watts Bar and determined that the evaluation and conclusions reached in the Sequoyah Safety Evaluation Report (NUREG-1232, Supplement 2) are also applicable to the Watts Bar MSLB analysis. Therefore, the MSLB analysis for the MSVV rooms, including the effects of superheat, is acceptable. Outstanding Issue 24 is thus resolved.

3.6.2 Determination of Break Location Dynamic Effects Associated With the Postulated Rupture of Piping

During the review of FSAR Amendment 64, an issue was raised concerning construction of response spectra (RSs) for the steel containment vessel (SCV) resulting from the compartment pressure transients caused by pipe breaks. RSs obtained from the SCV are used to evaluate design adequacy of the items, such as piping, equipment, and cable trays, that are attached to the SCV. In response to two requests for additional information (March 25 and October 25, 1993, the applicant discussed the adequacy of the RSs in letters dated May 22 and December 30, 1993. The staff's evaluation of the issue follows. The scope of the evaluation is limited to the applicant's analysis related to the development of RSs, assuming that the input (pressure-time histories) applied to the SCV is acceptable.

The SCV for Watts Bar is a freestanding steel structure consisting of a cylindrical wall, a hemispherical dome, and a reinforced-concrete basemat with a liner plate on the top of the basemat. The cylindrical wall is 114 feet high from the base to the spring line of the dome and has an inside diameter of 115 feet. The cylinder wall thickness varies from 1-3/8 inch at the bottom to 1-1/2 inch at the spring line. The dome thickness varies from 1-3/8 inch at the spring line to 15/16 inch at the apex. Circumferential stiffeners are welded to the cylindrical shell at approximately 10-foot centers. Vertical stiffeners are placed at 5° intervals around the cylindrical shell between the first and the second ring stiffeners at the base.

The applicant based its structural analyses on a finite element method representing the SCV as an axisymmetric thin shell. The computer code ANSYS was used for the modeling, and the element selected for the shell was STIF 61. The applicant modeled the circumferential stiffeners explicitly. However, the vertical stiffeners were accounted for by adjusting the properties of the corresponding axisymmetric element. The element properties were represented as orthotropic to account for the effect of vertical stiffeners. The applicant increased element density to account for the masses of the attached personnel lock, equipment hatch, other equipment, piping system, and cable trays.

Loading on the SCV is pressure loading from a postulated pipe break. For constructing an RS, the applicant used the pressure transients resulting from a high-energy pipe break. In Section 3.6.3 of SSER 5, the staff approved the applicant's use of leak-before-break technology for the primary loop piping. Thus, in accordance with General Design Criterion 4 of Appendix A to 10 CFR Part 50, the applicant was allowed to eliminate the local dynamic effects associated with the double-ended guillotine breaks in the primary loop piping. The applicant used other breaks, such as the main steamline break, to develop a new set of design-basis-accident (DBA) SCV RSs. Internal pressures applied to the containment vessel were obtained from the compartment pressure-time history determined by Westinghouse. The time-dependent non-axisymmetric pressure on the SCV was decomposed into an equivalent loading in terms of Fourier series components. A total of 15 load cases corresponding to different pipe break scenarios were considered.

The applicant performed a series of linear transient dynamic analyses of the SCV for each of the 15 postulated accident cases. Shell displacements at various points were obtained by direct integration of the shell equation with the ANSYS code. Structural damping of 1 percent was used. These displacements were the bases for the construction of the RSs. The displacements were then differentiated twice to obtain acceleration-time histories that were used to generate RSs at 10 selected elevations and at 16 selected azimuths around the circumference of the shell. All spectral peaks were broadened by 15 percent to account for variation in material properties and modeling approximations.

The applicant stated that the RSs resulting from a high-energy line break were not used in the structural design of the vessel itself, but rather in the analysis of attachments to the SCV; thus, the design basis for the SCV continues to be the original DBA as discussed in Section 3.8.2.4.4 of the FSAR; that is, the pressure and temperature effects from previously postulated double-ended guillotine breaks of the primary loop piping were used as the design-basis loading for the SCV design.

As stated before, the analysis of the Watts Bar SCV for the development of RSs was based on the ANSYS code. The ANSYS code is widely used in the industry, and extensive technical verification programs exist to ensure that the results from the code are consistent with available data, such as closed form solutions in various text books. This is reflected in the ANSYS Verification Manual. However, verification data on the dynamic characteristics of thin shell are not readily available. For this reason, the staff requested additional information for the ANSYS verification.

In response to the staff request, the applicant stated in its December 30, 1993, letter that "the ANSYS code includes verification for analysis of thin shell structures. The SCV analysis methodology has relied only on documented features of the ANSYS computer code. The ANSYS code was verified and maintained in accordance with approved QA [quality assurance] procedures for software development and verification." However, during its review of the ANSYS Verification Manual, Revision 5.0, in the applicant's offices, the staff found that only cases for static loads were discussed for shell structures. Test Cases 13 and 20 in the manual pertained to cylindrical shells under static pressure. The staff was unable to find any analysis of shells under dynamic loads.

Subsequently, the applicant informed the staff that Swanson Analysis System, Inc., developer of the ANSYS code, had told the applicant that it had verified shell dynamic characteristics for the element SHELL 61. It is believed that STIF 61, the element used by the applicant, was renamed as SHELL 61 in Revision 5.0. The ANSYS Verification Manual, reviewed by the staff, addresses only plate natural frequency in its element SHELL 61. It does not discuss shell natural frequencies associated with the element STIF 61, nor any other elements.

In a conference call, the applicant informed the staff that it could not obtain any experimental data that simulated the Watts Bar SCV analysis. Because of a need to gain sufficient confidence in the applicant's analysis, the staff performed several confirmatory calculations for natural frequencies of simplified shells. First, natural frequencies of a circular cylindrical shell were calculated using the ALGOR code ("Finite Element Analysis System," February 1994, Revision 4, ALGOR, Inc. Pittsburgh, PA 15238-2932). The staff selected a circular cylindrical shell because experimental data as well as a theoretical derivation of shell natural frequency could be found in a National Aeronautics and Space Administration (NASA) technical publication, NASA SP-288 (Arthur W. Leissa, "Vibration of Shell," 1973). The theoretical derivation, however, was based on a conventional force-moment method. Experimental data discussed in NASA SP-288 all pertained to small shells, 6 inches in diameter and 0.01 inch in thickness. Data in the report were exclusively on natural frequencies of thin shells. The ALGOR code, like the ANSYS code, is a general-purpose finite element code (sold to the general public). The staff used plate elements to simulate shell structure. Comparison of the results with the experimental data reported in NASA SP-288 indicated that the ALGOR code is capable of predicting the data reasonably well, particularly for a shell with a fixed-fixed boundary condition.

For a cylindrical shell with one end fixed and the other end free (fixed-free boundary condition), there were some disagreements between experimentally obtained data in NASA SP-288 and calculated frequencies with the ALGOR code, especially at lower natural frequencies. To investigate the implication of this discrepancy with regard to ANSYS's capability to reasonably predict experimental data, the staff modified the shell geometry used in the ALGOR analysis. A hemispherical dome was added to the cylindrical shell to simulate the shape of the Watts Bar SCV without changing the basic shell dimension, such as diameter and thickness of the small prototype shell used for obtaining the experimental data in NASA SP-288. This was done to determine how the fundamental natural frequency of the SCV-type shell compared with those of a fixed-fixed and a fixed-free cylindrical shell already calculated. The staff expected that the fundamental natural frequency of a cylindrical shell with a dome would lie somewhere between the two.

Fundamental natural frequency of the newly modeled SCV-shaped shell was determined to be closer to the fixed-fixed cylindrical shell. This indicated that a finite element determination of natural frequency for a circular cylindrical shell with hemispherical dome would reasonably predict the experimental data. With this information, the staff next calculated natural frequencies for the full-size Watts Bar SCV. Respective fundamental natural frequencies with the ALGOR code and the ANSYS code were 5.18 Hz and 5.94 Hz. Considering that the ALGOR model is approximate in mesh size for performing a confirmatory analysis and also that it is a shell with a constant thickness rather than a variable thickness as was the case for the Watts Bar SCV, the results from the two analyses were found to fit reasonably well. From these independent evaluations, the staff concluded that, despite the lack of specific verification of the ANSYS code for shell dynamics, the applicant's analysis would produce reasonable and credible results.

The staff selected natural frequency calculation as an object of the evaluation because natural frequency is a key measure of a shell's dynamic property and stiffness. Also, the experimental data reported in NASA SP-288 pertained exclusively to the natural frequencies of shells. Displacement time history obtained by the applicant's direct integration of the shell dynamic equation also largely depended on shell stiffness. In addition, the applicant selected the size of the time steps for the integration of SCV response explicitly on the basis of the shell's natural frequencies.

As stated earlier, the applicant used the ANSYS code, which is similar to the ALGOR code; these are both general-purpose finite element codes available to the general public. The ANSYS code is more widely used in industry than the ALGOR code, and has been used longer. As a result, the staff believes that a high degree of reliability would exist with regard to QA, such as arithmetic logic and coding of the theory.

From the preceding discussion, the staff finds reasonable assurance that equipment attached to the SCV as designed, by the use of RSs developed by the applicant, would maintain its structural integrity under the load. The staff bases its findings on reasonable comparison of the ALGOR code analysis results with experiments in terms of natural frequencies, close similarity between the methodologies used in developing the ALGOR and the ANSYS codes, and a favorable comparison of the SCV fundamental natural frequency obtained from the analyses performed using the two codes. The staff, therefore, concludes that the issue related to the SCV RSs for the Watts Bar SCV has been resolved. The staff limits its endorsement of the applicant's use of the SCV shell model using the ANSYS code to the development of RSs to be used for the design of equipment.

On the basis of this discussion, the staff concludes that the applicant's methodology for obtaining shell dynamic displacements due to postulated pipe breaks, and construction of response spectra for the Watts Bar equipment and supports are acceptable.

The staff tracked its efforts by TACs M79692 and M79693.

3.8 Design of Seismic Category I Structures

3.8.3 Other Seismic Category I Structures

The staff's review of FSAR Section 3.8.4.5, as revised by Amendment 79, revealed that the watertight equipment hatch cover does not have a table showing allowable stresses. Instead, a brief statement is made for normal and limiting conditions. In contrast to the watertight equipment hatch covers, the railroad access door (FSAR Table 3.8.4-5) has five types of load combinations. In a letter dated August 18, 1994, the applicant stated that "While the statement contained within Section 3.8.4.5.3 is correct as stated, the amount of information provided is not consistent with similar components delineated in the FSAR. By FSAR Amendment 88, the applicant added a table to explicate materials, specific load cases, and associated allowable stresses." The staff finds the applicant's resolution acceptable.

This review was tracked by TACs M88488 and M88489.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

In FSAR Amendment 86, the applicant modified Section 3.9.2.1, "Preoperational Vibration and Dynamic Effects Testing on Piping," to add a sentence that states, "Sample and instrument lines beyond the root valves are normally not included." The applicant reasoned that most of the energy for vibration is contained in the process piping for ASME Class 1, 2, and 3 systems. Consequently, the sample and instrument lines, beyond the root valves, are not expected to vibrate significantly and, therefore, are not included in the applicant's preoperational piping dynamic effects test program. The staff accepts the applicant's justification.

The applicant made the following revisions to Table 3.9-17, "Active Valves for Primary Fluid Systems," to accurately reflect the actual plant configuration:

- FCV-74-21 was relabeled, changing it from a check valve to a gate valve.
- FCV-74-24 was added.
- FCV-74-33 was indicated as a gate valve.
- FCV-74-35 was added.
- RFV-77-2875 was added.

Because these revisions do not involve a change to the physical plant, they are acceptable.

The applicant modified Table 3.9-20, "Relief Valves in Class 2 Auxiliary Systems," by changing the chemical and volume control system (CVCS) positive displacement charging pump discharge valve setpoint from 2680 psig to 2735 psig. The original valve setpoint was 2735 psig. In a previous amendment, the applicant erroneously concluded that the valve setpoint was incorrect and had to be changed to 2680 psig. The applicant then found that the original valve setpoint of 2735 psig was correct and restored this number in Amendment 86. The staff reviewed this revision and found it acceptable.

The staff tracked its efforts by TACs M89217 and M89218.

3.9.6 Inservice Testing of Pumps and Valves (Unit 1)

As required by 10 CFR 50.55a, inservice testing (IST) of certain ASME Code Class 1, 2, and 3 pumps and valves should be performed in accordance with Section XI of the ASME Code and applicable addenda, except where alternatives have been authorized or relief has been requested by the applicant and granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the applicant must demonstrate that (1) the proposed alternatives provide an acceptable level of quality and safety, (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (3) conformance is impractical for its facility. The staff's guidance in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," gives alternatives to the Code requirements determined acceptable to the staff. Alternatives that conform with the guidance in GL 89-04 may be implemented without additional NRC approval. The staff reviewed the applicant's relief requests against GL 89-04 to determine conformance. Any concerns identified by the staff's review are discussed in the following sections.

The 1989 Edition of the Code, Section XI, Subsections IWP and IWV, state that the rules for inservice testing of pumps and valves should conform to the requirements in ASME/American National Standards Institute (ANSI) Operations and Maintenance Standards Part 6 (OM-6), "Inservice Testing of Pumps in Light-Water Reactor Power Plants," and Part 10 (OM-10), "Inservice Testing of Valves in Light-Water Reactor Power Plants." As clarified in 10 CFR 50.55a(b)(viii), the correct addenda and edition of the ASME/ANSI OM standard are the OMa-1988 Addenda to the OM-1987 Edition. The Watts Bar Nuclear Plant, Unit 1, IST Program is based on the requirements in the 1989 Edition of the Code. The program covers the first 10-year interval that starts on the date the plant begins commercial operation.

This safety evaluation applies to the first 10-year interval IST Program that begins on the date that the unit commences commercial operation. The original IST Program was submitted on March 27, 1985, supplemented by submittals dated April 12, April 29, and October 3, 1985, and May 1, 1986. Revision 1 was submitted on March 15, 1994. Revision 2 was submitted on July 22, 1994, to incorporate information related to Revision 1 discussed with the staff, and to incorporate information from the unit's preoperational testing program. The safety evaluation that follows addresses the relief requests included in Revision 2 of the Watts Bar Nuclear Plant, Unit 1, Inservice Testing Program for pumps and valves.

The staff tracked its efforts by TAC M74801.

3.9.6.1 Pump Test Program

The IST Program covers 8 ASME Code Class 2 pumps, 16 Code Class 3 pumps, and 10 non-Code class pumps. Relief requests for the Code class pumps are subject to review and approval by the Commission in accordance with 10 CFR 50.55a. Relief requests for the non-Code class pumps are not required to be reviewed and approved by the Commission in accordance with 10 CFR 50.55a, as the scope of Section 50.55a is limited to certain safety-related pumps that are Code Class 1, 2, or 3. In GL 89-04, Position 11, the staff stated that it was acceptable to include non-Code class pumps in the IST Program. When such pumps are included in the IST Program, testing should conform to the requirements of ASME Code Section XI to the extent practicable. Impractical conditions should be described and documented in the IST Program, but do not require NRC approval before implementation.

<u>General Pump Relief Request PV-01, Vibration Limits for Smooth-Running Pumps</u>

For velocity measurements, OM-6, Table 3A, requires that for centrifugal and vertical line shaft pumps operating at >600 rpm, the "alert" range be the lesser of >2.5 V_{ref} to 6 V_{ref} or 0.325 to 0.70 inch per second and the "required action" level be the lesser of >6 V_{ref} or 0.70 inch per second. The applicant has requested relief from these limits for all pumps in the IST Program that operate at low levels of vibration, referred to as "smooth-running" pumps.

The applicant states:

> The OM-6 requirements do not provide for pumps which have extremely low levels of vibration. For example the WBN [Watts Bar Nuclear Plant] 1B-B Safety Injection pump outboard bearing vibration is approximately 0.014 in./sec. Based on the OM-6 ranges, this reference value would result in entry into the Alert range at 0.035 in./sec and into the Required Action Range at 0.084 in./sec. By the standards listed below, these vibration levels are considered acceptable. Based on current vibration data, the application of the OM-6 ranges would result in a significant percentage of the WBN pumps entering the Alert range with vibration levels below 0.1 in./sec. The required increased frequency testing would accelerate the normal wear process and ultimately lead to increased maintenance activity and reduced availability.

A review of three widely accepted sets of guidelines for absolute vibration limits provides the results in Table 3.2.

The applicant states:

> Establish a minimum reference vibration threshold level of 0.10 in./sec peak velocity for centrifugal and vertical line shaft pumps operating $\geq$ 600 rpm. Alert and Required Action Levels for baseline vibration levels at or below 0.10 in./sec peak velocity will be 0.25 and 0.6 in./sec respectively. Components with measured vibration levels less than 0.10 in/sec peak velocity during testing will be acceptable, regardless of relative change from the baseline levels. Alert and Required Action levels for baseline vibration levels above 0.10 in./sec peak velocity will be as described in Table 3a [of

OM-6]. Alert and Required Action levels for Reciprocating pumps and for Centrifugal and Vertical Line Shaft pumps operating at < 600 rpm are not affected by this relief request and will be as described in Table 3a [of OM-6].

Table 3.2 Guidelines for Absolute Vibration Limits

Vibration Level	Quality Judgment
OM, Part 6	
> 0.325 in./sec	Alert range
> 0.700 in./sec	Required action
ANSI[1] S2.41 (ISO[2] 2372)	
0 - 0.10 in./sec	Good
0.10 - 0.25 in./sec	Satisfactory
0.25 - 0.62 in./sec	Unsatisfactory
> 0.62 in./sec	Unacceptable
IRD[3] General Machinery Vibration Severity Chart	
0 - 0.08 in./sec	Good
0.08 - 0.16 in./sec	Fair
0.16 - 0.31 in./sec	Slightly rough
0.31 - 0.63 in./sec	Rough
> 0.63 in./sec	Very rough

[1]American National Standards Institute

[2]International Standards Organization

[3]Vibration monitoring equipment vendor

The code requires the establishment of reference values for vibration measurements. It gives acceptance criteria for vibration in both relative and absolute terms. A well-balanced pump may exhibit very low vibration levels after installation or maintenance. The industry and the ASME Operations and Maintenance (O&M) Committee are concerned about using low reference values for smoothly running pumps, as this might lead to determinations of inoperability at very low vibration levels. The ASME O&M Task Group on Vibration is actively considering the reference value issue for smoothly running pumps. Assigning minimum reference values, such as 0.1 in./sec, has been discussed in committee meetings; however, there is not yet an approved change to the Code. It is obvious that a multiple of a very small reference value of vibration (such as 0.01 in./sec) could result in requiring action at what is generally considered a very low level of vibration (e.g., 6 × 0.01 in./sec =

0.06 in./ sec). That level is well below the absolute required action limit of 0.7 in./ sec and the alert level of 0.325 in./sec. Many new pumps vibrate in excess of 0.06 in./sec. An Electric Power Research Institute (EPRI) Report, EPRI GS-7406, "Vibration Sensor Mounting Guideline," reported variations of up to 0.05 in./sec in vibration readings made by several people with a hand-held probe.

The applicant proposes to assign a minimum reference value of 0.1 in./sec for any pump with a measured reference value of $\leq$0.1 in./sec. This value of vibration velocity is generally indicative of a pump in excellent operating condition, particularly pumps with speeds greater than 600 rpm. Values of pump vibration velocity that are 2.5 times higher than this reference value (in the Alert range) are generally representative of pumps that are still in good operating condition. Values of pump vibration velocity that are 6 times higher than this reference value (in the Required Action range) are generally representative of pumps that are in a "rough" or "unsatisfactory" condition, but not an inoperable condition (see Table 3.2 above). The assignment of a minimum vibration velocity reference value of 0.1 in./sec should allow an adequate assessment of pump condition for an interim period until the code committee establishes appropriate guidance for setting minimum reference values for smoothly running pumps. Requiring the applicant to assign very low reference values (<0.1 in./sec), which are representative of the actual cyclic vibrational forces on the pump and are the result of good practices, may lead to unneeded testing and maintenance on pumps that are in good operating condition and do not threaten plant safety. Imposing the code requirements for assigning the reference values would impose a hardship or unusual difficulty on the applicant and would not be offset by a compensating increase in the level of quality and safety for most pumps.

However, there could be cases, such as for small pumps, where the proposed reference values were inappropriate. Before assigning the proposed minimum reference value, the applicant should review the application and the manufacturers' recommendations to ensure that the proposed minimum reference value is appropriate for each applicable pump.

Another issue to consider is the mobility characteristic of the structure at the point at which the vibration measurement is taken. Mobility is a measure of the ease with which a structure can be set in motion by a force. The following formula shows the relationship of force and mobility to vibration: Force × mobility = vibration.

Given a certain level of cyclic vibrational force, if vibration is measured on an area of a pump with relatively low mobility, the measured value will be low. That is, the magnitude of the measured value would be influenced (in this case, suppressed) by the low mobility of the machine at the measurement point. In such a case, a low reference value of vibration velocity would be appropriate. The applicant should consider using the actual measured vibration level if low mobility is a problem.

The proposed alternative is authorized for an interim period pursuant to 10 CFR 50.55a(a)(3)(ii), based on (1) the hardship or unusual difficulty that would result if the requirements were imposed, (2) the assurance of acceptable operating conditions with the proposed monitoring criteria, and (3) the consideration that the requirements would not provide a compensating increase in the level of quality and safety for the pumps that operate in the low range of vibration levels. The authorization is predicated on the provision that, before assigning 0.10 in./sec as a minimum reference value, the applicant should review each case, including any manufacturers' recommendations on acceptable vibration levels, to ensure that the proposed minimum reference value is appropriate. Once the O&M Committee reaches consensus and changes the code to include vibration guidance for smoothly running pumps, the applicant must adopt the guidance or develop and justify a reasonable alternative to the code. If the O&M Committee changes the code in a manner that is consistent with the requested alternative, no further action will be required for the alternative to be acceptable on a continuing basis.

Relief Request PV-02

Relief Request PV-02 was deleted in Revision 2 of the IST Program.

Relief Request PV-03, Boric Acid Transfer Pumps

The boric acid transfer pumps supply boric acid to the suction of the charging pumps for emergency boration of the primary system. Paragraph 4.6.1.1 of OM-6 specifies that the instrument accuracy for flow instrumentation be within the limits of Table 1 of OM-6, which lists an accuracy for flow rate measurement of 2 percent. The accuracy is specified as a percentage of full scale for individual analogue instruments, a percentage of total loop accuracy for a combination of instruments, or over the calibrated range for digital instruments. The applicant sought relief from the accuracy requirements. The relief request also listed chilled-water-system pumps, which are non-code class. For the chilled-water-system pumps, NRC approval pursuant to 10 CFR 50.55a is not required, as stated above.

The applicant states:

> The only permanently installed flow instrumentation in the piping for the Boric Acid Transfer Pumps is in the line which supplies undiluted boric acid to the charging pump suction. Using this line during operation results in making a significant negative reactivity insertion. Temporarily installed instrumentation is available which will yield a 3% accuracy.

The applicant proposes to perform the pump test quarterly, using temporarily installed flow instrumentation with an accuracy of 3 percent.

The applicant indicates that the use of a line with currently installed instrumentation, which presumably complies with the requirements of the code, results in a negative reactivity insertion during power operations. Therefore, it would be impracticable to use these lines to perform quarterly testing while operating at power conditions. However, the proposed alternative does not include a test using instruments that comply with the Code on a cold shutdown or refueling outage frequency. Also, the applicant states that the accuracy of the temporarily installed flow instrumentation is 3 percent, without discussing whether the 3 percent is of the "reading" (in which case it could be within the accuracy of the code requirements on the basis of the 2-percent accuracy over a range of 3 times the reference value or less) or of the reference value for a range that meets the code requirement for 3 times the reference value or less. Generally, temporary test instruments are more accurate than permanently installed instruments (e.g., 0.5% accuracy over the range of the instrument). Flow instruments that use acoustics to measure the flow are generally digital and are calibrated to a percentage of reading with measurements outside a 2-percent accuracy. The applicant should clarify the issues of testing during cold shutdown or during refueling outages in addition to quarterly testing, particularly with regard to operating the pumps on minimum recirculation (see GL 89-04, Attachment 1, Position 9), and the issue of the accuracy of the temporary instrumentation. Requiring testing each quarter through the instrumented line would cause a negative reactivity insertion, which is not desirable and can create a safety concern for operation or cause a plant transient. Because immediate imposition of the code requirements would impose a hardship or unusual difficulty on the applicant without a compensating increase in the level of quality and safety that can be achieved by monitoring the pumps according to the proposed alternative for an interim period, approval can be authorized for the first operating cycle (i.e., until the first refueling outage). During the interim period, the applicant must further evaluate the adequacy of testing these pumps without an additional test during cold shutdowns or refueling outages through the line with installed flow instrumentation. This does not represent a restraint on licensing the plant, as the alternative testing during the interim represents an acceptable means of monitoring the condition of the pumps during the first cycle of operation. Another option that the applicant may want to consider is the "comprehensive pump testing," which is described in the OMc-1994 Addenda to the OM Code and is structured for standby pumps. Preoperational testing assures the capability of the pumps to perform their safety function.

The alternative to use temporary flow instrumentation for inservice testing of the boric acid transfer pumps is authorized for a period not to exceed the first refueling outage. The authorization is granted pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis of the hardship or unusual difficulty without a compensating increase in the level of quality and safety. During the interim period, the applicant must further assess the possibility of performing a supplemental test during cold shutdowns or refueling outages.

Relief Request PV-04, Screen Wash Pumps

The screen wash pumps supply water to flush accumulated debris from the surface of the traveling water screens in the suction to the essential raw cooling water system. Paragraph 5.2(d) [Relief Request PV-04 incorrectly references paragraph 5.3(d)] of OM-6 requires that flow rate be determined during inservice testing and compared to a reference value. The applicant requested relief from the requirement to measure flow for the screen wash pumps.

The applicant states:

> These pumps are not equipped with flow instrumentation. Piping configuration does not provide adequate straight runs of piping to install permanent or temporary clamp on type flow instrumentation. Flow is not the critical parameter for these pumps. The nature of their operation is to ensure that sufficient pressure is maintained at the spray nozzles during flushing operations of the traveling water screens to ensure that sufficient force is exerted on the debris accumulated on the screen to remove it. This can be verified by verifying the effectiveness of the flushing operation.

The applicant proposes to verify that the flow through the spray nozzles in the traveling water screens is adequate for covering the screen spray area and for flushing away debris on the screen. Pressure and vibration data will be collected and analyzed in accordance with OM-6.

According to FSAR Section 9.2.1, the screen wash pump motors take power from both normal and emergency sources to ensure the screens are clear, ensuring a continuous flow of essential raw cooling water (ERCW) under all conditions. The ERCW pumps, intake pumping station traveling screens and screen wash pumps, and associated piping and structures must remain operable during and after a safe-shutdown earthquake that might destroy non-seismic structures and equipment and the main river dams upstream and downstream of the plant site. The traveling screens remove debris from the reservoir water supply for the ERCW pumps. Each screen wash pump functions to clean the associated traveling screen to flush out accumulated debris and keep the screen clear so that an adequate flow of water to the ERCW pumps will continue. No specific flow rate is associated with the screen wash pumps. The pumps are adequately performing their safety function if the flow through the spray nozzles flushes the debris from the traveling screens. Therefore, each time the pumps are tested, the safety function would be demonstrated.

Monitoring differential pressure and vibration in accordance with OM-6 will provide a means of detecting degradation in the hydraulic or mechanical condition of the pumps, while ensuring that the flow is adequate to meet the design-basis function. It is impractical to comply with OM-6 requirements for measuring the flow because of limitations in the design of the pumps and piping. The screen wash pumps are vertical turbine-type pumps. The piping configuration does not have straight piping long enough for installing flow-measuring instrumentation or for using temporary clamp-on measuring devices. To impose the requirements, the plant design would have to be modified, or a new method of measuring flow that does not use instrumentation would be necessary. Both of these options would burden the applicant. Considering that the design function of the pumps is demonstrated during each inservice test, the proposed alternative offers an adequate level of assurance of the operational readiness of the pumps.

Relief is granted from measuring the flow during inservice testing of the screen wash pumps pursuant to 10 CFR 50.55a(f)(6)(i) on the basis of (1) the limitations in the piping design with regard to installing flow instrumentation or using temporary clamp-on flow-measuring devices, (2) the burden on the applicant if the requirements to measure flow were imposed, and (3) consideration of the adequate level of assurance of the operational readiness of the pumps that will be achieved by the alternative testing.

3.9.6.2 Valve Testing Program

The IST Program covers approximately 650 valves. It includes 34 "alternative frequency justifications," which address deferred testing for approximately 202 valves, and 9 relief requests, 5 of which state that the alternative is in conformance with Position 2 of Generic Letter 89-04. The staff discusses the six relief requests that conform with Position 2 at the end of this section (see page 3-23). It discusses anomalies in Section 3.9.6.4 (below). The staff's evaluation of the remaining four relief requests follows. Results of the staff's review of the 34 "alternative frequency justifications" are summarized in Section 3.9.6.3.

Relief Request PV-05, Containment Isolation Valves

The applicant requested relief from assigning permissible leakage rates for each containment isolation valve (CIV) or valve combination as specified by paragraph 4.2.2.3(e) of OM-10 as invoked by 10 CFR 50.55a(b)(2)(vii).

The applicant states:

> It is the total leakage from containment which is of significance in determining the effects of an accident and which is required by 10 CFR 50 Appendix J, not the leakage from an individual valve. Watts Bar has developed the following alternative to meet the total leakage requirements of Appendix J while still providing assurance that a single valve does not become the major source of leakage from containment.

The applicant proposes:

> CIVs are assigned conservative reference leak rates based upon the valve size and considering the total allowable containment penetration leakage, 0.6 L_a. The total of all of the reference leak rates is set to equal approximately 40% of 0.6 L_a. This provides a comfortable margin, even if all valves are leaking their respective reference leak rates. If a maximum permissible leak rate is not specified by the owner (licensee), OM-10 paragraph 4.2.2.3(e) requires a leak rate acceptance criteria equivalent to 0.3125 SCFH [standard cubic feet per hour] per inch valve size. The reference leak rate assigned to CIVs from the preceding methodology corresponds to an average of 0.06 SCFH per inch valve size. This is less than one fifth the OM-10 guidelines, a much more conservative number.
>
> During refueling outages maintenance is performed, as required, in an attempt to restore all CIVs to below their reference leak rates and as close to zero leakage as is reasonably achievable. This ensures the ability of the containment system to satisfy the integrated leak rate testing criteria and to provide adequate margin for valve degradation over the next fuel cycle. While every attempt is made to maintain CIVs at zero leakage or below their reference leak rates at all times, a valve leaking in excess of its reference value may remain operable and left "as is," provided that an evaluation finds it acceptable with 10 CFR 50 Appendix J. An example of such a situation would be a valve found to be leaking in excess of its reference leak rate in mid-fuel cycle, and for which all reasonable on-line maintenance efforts have been made. Such evaluation shall be based upon consideration of the effects on overall containment leakage and possible effects on adjacent piping and components, as well as consideration of time, cost, unit operations, and radiological exposure required for corrective measures. While the maximum permissible leak rate at this time would, by plant Technical Specifications, be limited to the current margin between overall containment leakage and 0.6 L_a, maximum single leakage is at all times administratively limited to a value that is as low as reasonably achievable and consistent with the evaluation by the 10 CFR 50 Appendix J program supervisory personnel or program engineer. Any

such valve would be repaired or replaced no later than the next refueling outage or even during the next cold shutdown of sufficient duration.

The above described methodology of setting and maintaining ultraconservative reference leak rates ensures system operability and provides reasonable assurance of valve leak tight integrity intended by the Code. At the same time flexibility is provided to prudently operate until the next refueling outage or lengthy cold shutdown when a valve exceeds its reference leak rate and all reasonable efforts have made to reduce its leakage.

Paragraph 4.2.2.3(e) of OM-10 requires that leakage rate measurements be compared with the permissible leakage rates specified by the plant owner for a specific valve or valve combination, and gives a formula for permissible leakage rates that are not specified by the owner. The applicant's proposal to assign target leak rates for CIVs, or valve combinations, is more conservative than that which would result if the formula in paragraph 4.2.2.3(e) were used. The total leakage of all the penetrations subject to Type C testing (local leak rate testing) in accordance with 10 CFR Part 50, Appendix J, would be limited to 0.6 L_a in accordance with Appendix J requirements. Paragraph 4.2.2.3(f) of OM-10 requires corrective action for valves or valve combinations with leakage rates exceeding the values established according to paragraph 4.2.2.3(e). Corrective actions may be either repair or replacement. The applicant proposes to perform an evaluation against the total leakage limit of 0.6 L_a in certain instances when repair or replacement may not be feasible. The alternative would ensure that the overall leakage remained within the limits of Appendix J and, therefore, would not be outside the design basis for the containment. This approach to controlling the leak rates is not in conformance with paragraph 4.2.2.3(f) if the target limits established according to paragraph 4.2.2.3(e) are exceeded; however, the target limits are based on fractions of the total allowed leakage limits. It is the total leakage limit that may not be exceeded. Therefore, the proposal ensures an acceptable level of quality and safety.

The alternative to include evaluation of the target leakage versus the total allowed leakage of 10 CFR Part 50, Appendix J (0.6 L_a) as an acceptable corrective action (along with repair or replacement) is authorized pursuant to 10 CFR 50.55a(a)(3)(i) because it provides an equivalent level of quality and safety for operating and maintaining the plant within accepted regulatory limits.

Relief Request PV-06, Safety or Relief Valves

The relief request is applicable to all safety or relief valves that provide overpressure protection in accordance with OM-1, "Requirements for Inservice Testing of Nuclear Power Plant Pressure Relief Devices," and that are tested at ambient conditions. Paragraphs 8.1.1.8, 8.1.2.8, and 8.1.3.7 of OM-1 specify that a minimum of 10 minutes shall elapse between successive openings. OM-10 references OM-1 for the requirements for inservice testing of safety and relief valves.

The applicant states:

> These steps require a ten minute delay between successive openings of valves. They [the steps listed above] were included in OM-1 in order to allow for thermal stabilization following an opening at an elevated temperature. When testing at ambient temperature this introduces an unnecessary delay in testing.

The applicant proposes to observe the 10-minute delay for tests performed at elevated temperature, but not to observe the hold time for thermal stabilization for tests performed at ambient conditions. The frequency of the proposed alternative will be as specified in OM-1.

The applicant states that the reasons for the 10-minute delay between successive openings of valves was to ensure thermal stabilization before successive tests. It is not clear that the intent of the 10-minute delay has such a basis. Thermal equilibrium is discussed in separate paragraphs of OM-1, with changes in the OMc-1994 Addenda to the 1990 Edition of the OM Code stating that "[v]erification of thermal equilibrium is not required for valves which are tested at ambient temperature using a test medium at ambient temperature." At the O&M Committee meeting in San Jose, California, in June 1994, the OM-1 Working Group discussed the 10-minute stabilization time between valve lifts. The discussion was not focused entirely on temperature stabilization. Utilities and valve test vendors have commented that the 10-minute period is longer than necessary and adds to the costs of set-pressure testing. A representative from the Westinghouse Service Center presented test data that demonstrate that a 5-minute period between valve lifts provides test results as accurate as a 10-minute period. The working group accepted the general principle to change the 10-minute period to a 5-minute period. There was no discussion of whether the stabilization period should be eliminated for valves tested at ambient conditions. Therefore, the proposed alternative cannot be approved. The applicant may elect to submit an inquiry to the O&M Committee suggesting that the stabilization period be eliminated, describing the reasons it believes this would be acceptable within the intent of the code.

The proposed alternative is not approved as requested. The staff informed the applicant of this decision by letter dated August 25, 1994. The applicant's justification does not necessarily agree with discussions by the O&M Working Group. The issue is more appropriately one that should be addressed by the O&M Committee, considering that the applicant's basis is stated as its interpretation of the requirements of OM-1, and is an issue that is already under discussion within the working group.

<u>Relief Request PV-09, Essential Raw Cooling Water System Valves</u>

The essential raw cooling water check valves open to admit air to the essential raw cooling water pump column to allow the water trapped in the pump column to drain down. This prevents motor overcurrent as the pump tries to accelerate the column of water on a pump start. These pumps are deep draft pumps, which extend a considerable distance from the river elevation to the pump discharge head. Valves also close to provide a flow boundary when the pump starts and water again reaches the pump discharge head. Paragraph 4.3.2.4 of OM-10 states that the obturator movement may be verified by observing a direct indicator such as changes in system pressure, flow rate, level, temperature, seat leakage testing, or other positive means. Position 1 of GL

89-04 states that a check valve's full stroke to the open position may be verified by passing the maximum required accident condition flow through the valve, that such a flow is considered a full stroke, and that any flow rate less than the design flow rate will be considered a partial-stroke exercise. According to Position 1, a valid full-stroke exercise by flow requires that the flow through the valve be known. The applicant requested relief from the full-stroke exercising for certain essential raw cooling water check valves, stating:

> There is no practical way to determine the flow rate through these small diameter valves during the venting of the pump column. The rules of OM Part 10 and the guidance of Generic Letter 89-04 were developed with liquid flow in mind and not compressible gaseous flow. Attempting to measure an air flow rate this small will result in very inaccurate and unrepeatable results. The techniques developed for field measurement of compressible gaseous flow lend themselves to the measurement of much larger quantities through a much larger ducting system.
>
> Additionally, the nature of the flow through these valves is such that it will not be at a steady state long enough to quantify. The flow will rapidly accelerate to a maximum, then steadily decrease as the driving force of the water column level above the river elevation decreases.
>
> The critical parameter for determining the performance of these column vent check valves is not the flow rate they will pass but rather the time which it takes to allow the column to vent. This ensures that the column will be vented before it can reasonably be expected that another start signal will be generated for the pumps. As long as the column has been vented, the motor does not have to attempt to accelerate a standing column of water.

The applicant proposes:

> Initially establish a reference value for the length of time required for the column to vent. Subsequent tests will then determine the time it takes for the column to vent and compare the time to the reference value. An increase of 50% or more will be considered indication that the valve is not opening sufficiently. The closing function of the valve will be demonstrated each pump test.

The capability of the subject check valves to perform their safety functions can be verified by monitoring the operating characteristics associated with pump venting each time the pump is tested. Even though the gas flow cannot be measured, the testing is essentially challenging the design functions of these valves each time the pumps are stopped (opening function) and started again (closing function). The intent of the statements in GL 89-04 regarding accident flow was to ensure that the tested check valves could, at a minimum, open to the point required to pass design flow. The subject valves perform their design function each quarter when the associated pumps are tested; therefore, the intent of the statements in GL 89-04 is met. By establishing a "reference value" for the time necessary to vent the pump column and then measuring the length of time to complete venting during each pump test (quarterly for inservice testing) will ensure that the function has been accomplished, indicating that the valves have performed adequately (opened sufficiently to vent the column). Establishing a limit on the venting time sets acceptance criteria that ensure corrective action will be taken if valve performance is in question. By monitoring that water is not discharged through the vent column after the water reaches the pump discharge head, the capability of the valves to close can be demonstrated during each pump test as well. Although the proposed method could be considered an "other positive means" in conformance with OM-10, the use of a reference value and a limiting increase in such value are not discussed in the code. However, the method will provide an equivalent level of quality and safety in ensuring that the valves are capable of performing their safety function to protect the pump motor from overcurrent and to close when the water column is filled. Requiring a test method that monitors the valves by some means of direct measurement would be a burden on the applicant, and would necessitate installation of instrumentation not used for any other purpose.

The applicant's alternative method is authorized pursuant to 10 CFR 50.55a(a)(3)(i) because of the equivalent level of quality and safety provided by the method for verifying the capability of the valves to perform their safety function. The burden of imposing another method was considered in the evaluation.

Relief Request PV-12, Component Cooling and Chemical and Volume Control System Valves

Component cooling system and chemical and volume control system vacuum relief breakers, 1-RFV-62-1079 (vacuum breaker for the chemical holdup tank) and 1-RFV-70-539-S (vacuum breaker for the Unit 1 component cooling surge tank), open to relieve vacuum in the respective tanks to prevent tank collapse. Paragraph 1.4.1.2 of OM-1 specifies that the test equipment used to determine the set pressure for safety and relief valves should have an overall combined accuracy within +2 percent to -1 percent at the pressure level of interest. Paragraph 8.1.2.2 of OM-1 specifies that there be a minimum accumulator volume below the valve inlet. The applicant requested relief from these requirements and proposed an alternative, stating:

> The valves installed as vacuum relief valves are very similar in design to a spring loaded check valve. Their operation is a function of the pressure exerted of [sic] the difference in pressure forces acting on the two sides of the disc or the "pallet" as the valve manufacturer calls it. The force which causes this valve to open corresponds to an approximate pressure differential of 0.15 psi. This is a very small differential pressure and would be very difficult to establish, control and measure to the accuracy required by OM-1. The manufacturer's recommended method of verifying proper operation of this valve is to measure via a force gage the additional force necessary to cause the valve pallet to move from the full closed condition with no differential pressure present across the valve. This force is to be within the absolute tolerance required by the reference paragraph of OM-1, the measurement is of the force required to open the valve and not the pressure at which it opens. Additionally, since the setpoint is verified without causing fluid to flow through the valve, the requirement for a minimum accumulator volume during testing is not applicable.

The applicant proposes:

> Establish the valve's setpoint and verify proper operation using the manufacturer's recommended technique of determining within 1% the force required to cause the pallet to move off its seat.

The frequency of the alternative testing will be as specified in OM-1.

OM-1 does not include testing methods specific to vacuum breakers of the same type as the subject valves. In attempting to apply the testing methods of OM-1 to the manufacturer's recommended method, the discrepancy between measuring pressure and measuring force arises. Because force measurements do not specifically meet the Code requirements, relief is necessary to use an alternative method.

The use of a force gauge will give a value that can be directly correlated to the setpoint of the vacuum breakers and the technique is accurate to within 1 percent of the measured force. Such a device is very much like an "assist device," which is allowed by OM-1, provided the accuracy requirements are met. Imposition of the requirements to establish a test method at the very small differential pressure necessary to test the valves in accordance with the code would be a hardship and unusual difficulty and would not ensure a higher level of quality and safety than that ensured by the manufacturer's recommended method. Therefore, the alternative can be authorized.

The alternative method of using a force gauge with a specified acceptable range rather than a direct pressure measurement gauge is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) because of the hardship and unusual difficulty without a compensating increase in the level of quality and safety that would ensue if the code requirements were imposed. The alternative method will provide assurance of the setpoint of the valves and meet the intent of the code-required test within an acceptable accuracy of measurement.

<u>Relief Requests Approved by GL 89-04</u>

The staff has identified a number of generic deficiencies that affect plant safety and have frequently appeared as IST programmatic weaknesses. These are addressed by GL 89-04. In GL 89-04, the staff delineated positions that describe deficiencies and explained alternatives to the ASME Code that it considers acceptable. If alternatives are implemented in accordance with the relevant position in the generic letter, the staff has determined that relief should be granted pursuant to 10 CFR 50.55a(g)(6)(i) (now (f)(6)(i) for IST) on the grounds that it is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest. In making this determination, the staff has considered the burden on the utility that would result if the requirements were imposed.

For relief granted pursuant to GL 89-04, the staff has reviewed the information submitted by the utility to determine whether the proposed alternative conforms to the relevant position in the generic letter. If an alternative conforms to a position of the generic letter, it is listed below and is approved. Any anomalies in the relief request are addressed in Section 3.9.6.4.

The relief requests that are approved as being consistent with Position 2 in GL 89-04 are as follows:

(1) Relief Request PV-07: Certain auxiliary feedwater and main steam check valves (auxiliary feedwater flow to the steam generators and steam flow to the auxiliary feedwater pump turbine driver) will be disassembled and inspected on a sampling basis in accordance with Position 2. The valves will be partial-stroke exercised during cold shutdowns.

(2) Relief Request PV-08: The accumulator discharge check valves and residual heat removal/safety injection combined check valves will be disassembled and inspected on a sampling basis in accordance with Position 2. The combined check valves will be partial-stroke exercised during cold shutdowns.

(3) Relief Request PV-10: Essential raw cooling water check valves that open to pass cooling water flow to diesel generators, and close to isolate trains, will be disassembled and inspected on a sampling basis in accordance with Position 2 to verify the capability of the valves to prevent reverse flow. The opening capability of the valves will be verified quarterly.

(4) Relief Request PV-11: Component cooling water check valves that are required to close to prevent overpressurization of piping from the last check valve back to the containment penetration will be disassembled and inspected on a sampling basis in accordance with Position 2.

(5) Relief Request PV-13: Containment spray check valves that are required to open to pass water from either the containment spray or the residual heat removal pumps to the containment spray or residual heat removal ring headers will be disassembled and inspected on a sampling basis in accordance with Position 2. The relief request states that the function is to open, but the alternative is described as verifying the backseating function. The applicant should correct this discrepancy.

(6) Relief Request PV-14: Auxiliary feedwater check valves that are required to open to pass flow to the steam generators and to close to prevent backflow will be disassembled and inspected on a sampling basis in accordance with Position 2 to verify the capability to close. The capability to open will be verified quarterly.

3.9.6.3 Review of "Alternative Frequency Justifications"

The applicant included 34 test deferrals in the IST Program covering approximately 202 valves. For each of the test deferrals, the applicant's justification appears to be reasonable, considering the safety-related functions and the effects on system and plant operation if testing was performed during power operations. The deferrals appear to comply with the requirements in paragraphs 4.2.1.2 and 4.3.2.2 of OM-10 for exercising valves. Further staff review of test deferrals is through inspection, as noted in response to Question 102 of the "Minutes of the Public Meetings on Generic Letter 89-04," dated October 25, 1989.

3.9.6.4 IST Program Anomalies

By letter dated August 25, 1994, the staff sent its comments to the applicant on the Watts Bar Unit 1 IST Program. It asked the applicant to respond to the action items in that letter before startup from the first refueling outage. The staff did not find any IST issue that would prevent issuance of an operating license for Watts Bar Unit 1.

The staff will track future actions by TAC M90252.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.1.1 Compliance With Appendix G, 10 CFR Part 50

In 1982, the staff reported in the Safety Evaluation Report (SER) that the applicant had complied with all the requirements of Appendix G to 10 CFR Part 50, except for the specific requirements of Paragraphs III.B.4, IV.A.1, and IV.B. The staff stated that pursuant to 10 CFR 50.12, exemptions from these specific requirements may be granted (note: the staff originally intended to grant the exemptions as part of the operating license). In addition, the staff reviewed the applicant's compliance with Paragraph I.A.

Since the SER was issued, the staff has amended Appendix G to 10 CFR Part 50. In 1982, Paragraph III.B.4 required testing personnel to be qualified by training and experience and to be able to perform tests in accordance with written procedures. At that time, Watts Bar had no written procedures. Currently, Appendix G no longer requires written procedures, since it is unlikely that the tests are conducted improperly. The tests are relatively routine in nature and are required to be conducted in accordance with the applicable industry standards (American Society of Mechanical Engineers, ASME, and American Society for Testing and Materials, ASTM).

In Paragraph IV.A.1 (Paragraph III.A in the current regulation), the staff required that reactor coolant pressure boundary (RCPB) materials be tested to the requirements of Paragraph NB 2330 of the ASME Boiler and Pressure Vessel Code. The applicant complies with these requirements except for Unit 1 reactor vessel base metal that is located outside the beltline region and the beltline intermediate-to-lower-shell-root weld. In Appendix G, the staff still requires that the RCPB materials be tested to the ASME Code, but permits applicants to submit fracture toughness data and analyses to demonstrate the equivalency of the test data to the requirements of this appendix. As discussed in the SER in 1982, the applicant has submitted supplemental material to demonstrate that its fracture toughness data for the base metal that is located outside the beltline region and the beltline intermediate-to-lower-shell-root weld, is equivalent to that required in Appendix G.

In Paragraph IV.B (Paragraph IV.A.1 in the current regulation), the staff required that reactor vessel beltline materials have unirradiated Charpy upper-shelf energy (USE) greater than 75 ft-lb. The applicant complied with this requirement for all beltline materials except for the Unit 1 intermediate shell forging. In Appendix G, the staff presently requires the unirradiated Charpy USE to be greater than 75 ft-lb and the irradiated value to be no less than 50 ft-lb, but permits lower values if the lower values provide margins of safety against fracture equivalent to those required in Appendix G of the ASME Code. In a letter dated October 15, 1993, the applicant submitted an analysis to demonstrate that the intermediate shell forging would provide margins of safety against fracture equivalent to Appendix G of the ASME Code. The staff has reviewed the applicant's analysis and concludes that the intermediate shell forging provides the margins of safety required in Appendix G of the ASME Code. See Section 5.3.1.1.1, which follows, for details.

In Paragraph I.A, the staff required that applicants demonstrate to the staff, on a case-by-case basis, the adequacy of the fracture toughness of any ferritic material that has a specified minimum yield stress over 50 ksi and is used in a pressure-retaining component of the RCPB. The applicant has used material with a minimum specified yield stress in excess of 50 ksi in the pressurizer and the steam generator RCPB components. In Appendix G, the staff permits the use of materials with minimum yield stress in excess of 50 ksi, if they are qualified by using methods equivalent to those described in Paragraph G-2110 of the ASME Code. Westinghouse Topical Report WCAP-9292, "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Materials and Applicable Weld Metals," contains fracture toughness data and analysis to satisfy the requirements of Paragraph G-2110 of the ASME Code. The applicant indicates that the conclusions in WCAP-9292 concerning SA-533 Grade A Class 2 and SA-508 Class 2a materials are applicable to Watts Bar. On the basis of its acceptance of WCAP-9292, the staff considers the applicant's use of SA-533 Grade A Class 2 and SA-508 Class 2a materials in the pressurizer and steam generator RCPB components acceptable.

On the basis of its evaluation, the staff finds that the applicant has complied with all the requirements in the current Appendix G, 10 CFR Part 50 without exemptions. Thus, the exemptions previously approved in the SER are no longer needed. The staff tracked these efforts by TACs M85712 and M85713.

5.3.1.1.1 Unit 1 Equivalent Margins Analysis

In letters dated July 7, 1992, and January 28, 1993, the applicant stated that the initial upper-shelf energy (USE) value for intermediate shell forging 05 in the Watts Bar Unit 1 reactor pressure vessel would be below the initial value of 75 ft-lb required in Appendix G to 10 CFR Part 50. In addition, the applicant projected that the end-of-life (EOL) USE for forging 05 would be less than 50 ft-lb. To resolve this issue, the applicant requested an exemption from Appendix G (January 28, 1993, letter). During a July 20, 1993, conference call, the staff informed the applicant that an exemption from Appendix G would not be needed if the applicant demonstrates that forging 05 could meet the margins of safety against fracture equivalent to those required in Appendix G. Subsequently, by letter dated October 15, 1993, the applicant withdrew the Appendix G exemption request and submitted a low upper-shelf equivalent-margins analysis for Watts Bar Unit 1, intermediate shell forging 05. This analysis is intended to demonstrate, via fracture mechanics, the existence of margins of safety equivalent to those required by Appendix G of ASME Code Section III for reactor pressure vessel (RPV) beltline materials with unirradiated USE values below the screening criteria of 75 ft-lb, and/or EOL USE falling below the NRC screening criterion of 50 ft-lb before the end of its service life as stated in 10 CFR Part 50, Appendix G. Westinghouse performed the analysis for the applicant and closely followed the approach it used for the bounding owners group analysis (WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors", September 1993) which the staff previously assessed (letter, April 21, 1994).

In subsequent teleconferences, the staff also requested that the applicant's analyses include (1) any unirradiated or irradiated J-R curve data from forging 05 or *J-R* curve data from material similar to forging 05 that was forged by Fried. Krupp Huttenwerke AG and heat treated by the Rotterdam Dockyard Company to a condition equivalent to forging 05 and (2) a description of the applicant's planned fluence management program and an assessment demonstrating that, with this plan, the reactor vessel materials will conform to the margins of safety required by Appendix G of the ASME Code.

The regulatory guidelines concerning upper-shelf safety margins appear in Appendix G to 10 CFR Part 50. The requirements state that the unirradiated upper-shelf energy (USE) at the start of vessel life shall be no less than 75 ft-lb, and that the vessel must maintain a USE level of no less than 50 ft-lb throughout its service life. If there is reason to believe that any beltline material might begin life with an unirradiated USE value below the 75 ft-lb criterion and/or might fall below the 50 ft-lb threshold before the end of life (EOL) date, an analysis demonstrating the existence of "margins of safety against fracture equivalent to those required by Appendix G of the ASME Code," must be submitted. The Director, Office of Nuclear Reactor Regulation, must approve that analysis.

The staff has accepted guidelines for the performance of equivalent-margins analyses. These guidelines are contained in ASME Code Case N-512, "Assessment of Reactor Vessels With Low Upper Shelf Charpy Impact Energy Levels" (Section XI, Division 1, ASME Boiler and Pressure Vessel Code, February 12, 1993), and Draft Regulatory Guide DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 Ft-Lb" (August 1993). DG-1023 has incorporated the criteria of ASME Code Case N-512 and gives supplemental information on material properties and transient selection. The code case and draft regulatory guide recommend that licensees and applicants compare the fracture resistance to the applied fracture driving force. These documents contain criteria for Service Levels A, B, C, and D that must be satisfied to demonstrate that the material complies with the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. Service Levels A and B are defined by the ASME Code as normal operating and upset conditions, respectively. Typically, the limiting condition for Levels A and B is the 100 °F/hour heatup/cooldown transient. Levels C and D are the emergency and faulted conditions, respectively. For the analysis of Watts Bar Unit 1, Westinghouse found the small steamline break to be the limiting transient for Level C, while the large-break loss-of-coolant accident and large steamline break were the limiting Level D transients.

Since a majority of licensees do not have fracture toughness information for their limiting beltline materials, Charpy V-notch (CVN) data are typically used to estimate the fracture toughness. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (May 1988), contains a procedure for estimating the decrease in CVN upper-shelf energy as a function of copper content and fluence. In NUREG/CR-5729 ("Multivariable Modeling of Pressure Vessel and Piping J-R Data," are empirically derived models for predicting the material fracture toughness (*J-R* curves) from CVN data or chemical composition and fluence. The models in NUREG/CR-5729 are applicable to the majority of RPV materials.

In the July 7, 1992, letter, the applicant reported an unirradiated USE of 62 ft-lb for A508 Cl 2 intermediate shell forging 05, along with a copper content of 0.17 percent and an EOL fluence of 1.9E+19. This information and Figure 2 of Regulatory Guide 1.99, Revision 2, indicate an EOL USE for forging 05 of 43 ft-lb. As both the unirradiated USE and projected EOL USE values fall below the criteria of 10 CFR Part 50, Appendix G, the applicant chose to demonstrate "equivalent margins of safety against fracture" at the 43-ft-lb level which, in turn, bounds the 62-ft-lb unirradiated USE. This is demonstrated by comparing the material fracture resistance (*J-R* curve) with the driving force for fracture ($J_{applied}$) as per the criteria of DG-1023 and Code Case N-512.

As an actual *J-R* curve was not available for forging 05, the Westinghouse report for Watts Bar Unit 1 used the correlations with CVN energy that are provided in NUREG/CR-5729 to determine the *J-R* curve. The correlation model selected was the Charpy model for the RPV combined database, which was based on data for both base metals and welds.

For Service Levels A and B, the highest stress at the T/4 location occurs at 390.5 °F during the cooldown from the normal operating temperature at 100 °F/hour (the maximum rate permitted by pressure-temperature limits). Hence, *J-R* curves for a temperature of 390.5 °F were used. For Levels C and D, the temperatures used for the *J-R* curves were from the appropriate transients (in the range of 400 °F — 500 °F). The approach to developing the *J-R* curve data is acceptable. However, use of the NUREG/CR-5729 Charpy model for RPV base metals is also acceptable to the staff, and is considered the more appropriate model for a forging material such as A508 Cl 2. The mean -2σ *J-R* curves at 390.5 °F for both models are compared in Figure 5.1. The *J-R* curve from the RPV base metals model is substantially elevated in comparison with the curve from the RPV combined database model. The resulting $J_{0.1}$ for the RPV base metals model was 854 in.-lb/in.^{2}, and that for the RPV combined database model was 599 in.-lb/in.^{2} Hence, the material fracture resistance calculated by Westinghouse for Watts Bar Unit 1 should conservatively estimate the fracture resistance behavior of forging 05. The actual fracture resistance of the forging 05 material will be determined from compact fracture toughness specimens which are included as part of the Watts Bar Unit 1 reactor vessel surveillance program.

The Westinghouse report for Watts Bar Unit 1 also described the A302 Grade B bounding *J-R* curve analysis from WCAP-13587, Revision 1. As the Watts Bar Unit 1 reactor pressure vessel does not contain A302 Grade B, and sufficient information exists to enable determination of an A508 Cl 2 *J-R* curve as described above, the staff did not consider the A302 Grade B analysis.

The Westinghouse report for Watts Bar Unit 1 used the 100 °F/hr heatup/cooldown case as bounding for Levels A and B. This is consistent with the evaluation performed in WCAP-13587, Revision 1. For Levels C and D, the Westinghouse analysis used both peak stress and the overall magnitude of the through-wall stress as the criteria to determine the bounding transients. The resulting temperature at the crack tip (10% of wall thickness + cladding) was 487 °F for Watts Bar Unit 1 (see letter from W. Hodges dated April 21, 1994). Using these criteria, the applicant judged that a small steamline break was the limiting Level C transient, and large loss-of-coolant accident and large steamline break were the limiting Level D transients.

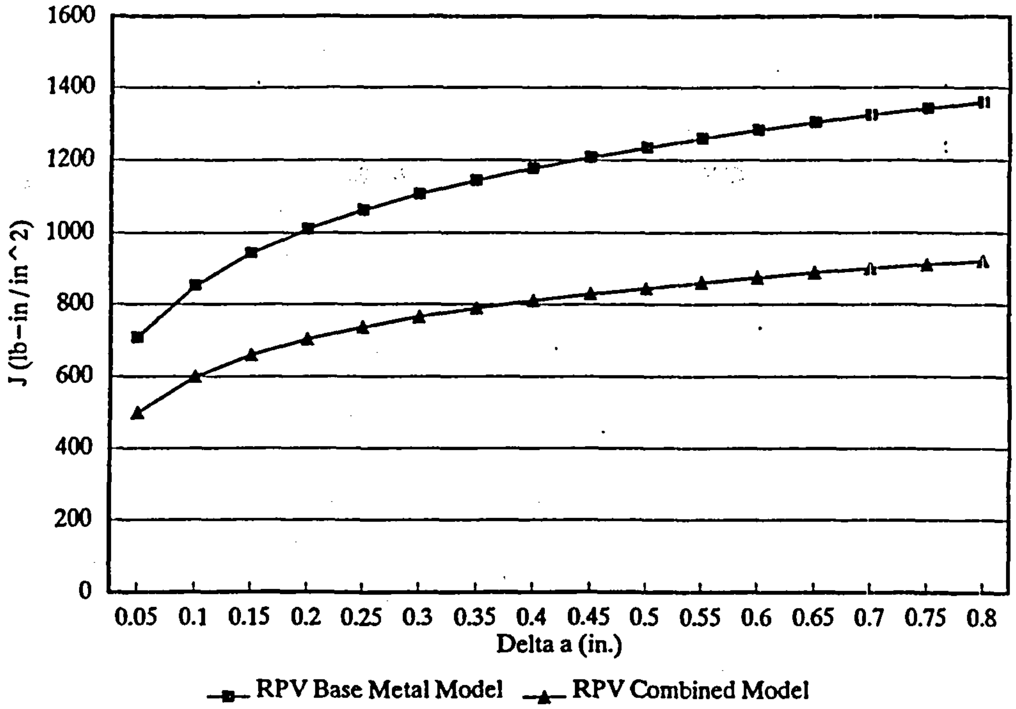

Figure 5.1 Watts Bar 1, Analysis of Intermediate Shell Forging 05 Model Comparison, Service Levels A and B

The Westinghouse analysis for Watts Bar Unit 1 employed the procedures of Code Case N-512 in determining the fracture driving force ($J_{applied}$) for Levels A and B. The staff independently evaluated the $J_{applied}$ calculations and was able to verify the Westinghouse calculations for $J_{applied}$ at 0.1 in. of crack extension. For the slope of the $J_{applied}$ curves, the Westinghouse analysis differed slightly from the staff's analysis but was within reasonable bounds of uncertainty for such calculations. For completeness, the Westinghouse analysis also evaluated two different equations used to compute the thermal component of the fracture driving force (K_{It}):

$$K_{It} = [(CR)/1000]\ t^{2.5}\ F_3$$

where CR = cooling rate in °F/hr
t = thickness in inches

$$F_3 = 0.584 + 2.647\ (a/t) - 6.294\ (a/t)^2 + 2.990\ (a/t)^3 \quad (1)$$

$$F_3 = 0.690 + 3.127\ (a/t) - 7.435\ (a/t)^2 + 3.532\ (a/t)^3 \quad (2)$$

The first equation is currently contained in Code Case N-512. The second equation is more conservative and was originally proposed as a replacement for the first equation by the ASME Code Section XI Working Group on Flaw Evaluation (WGFE). The WGFE has since accepted a third equation which will appear in the next revision of the code. The new equation yields K_{It} values between those determined from the previous two.

For Levels C and D, the pressure and temperature histories for all of the transients considered were put into a two-dimensional finite element model of the nuclear steam supply system using the WECAN computer code (Westinghouse Electric Computer Analysis (WECAN) Code, File 87-1J7-WESAD-R1, dated December 1987). The resulting stress distributions for the limiting transients, which included the contributions of the cladding to the thermal stress, were used to calculate $J_{applied}$ using the PCFAD computer code (Bloom, J.M., and D.R. Lee, "Users Guide for the Failure Assessment Diagram Computer Code FAD," Babcock and Wilcox, Rev. 4, April 1990). Consistent with the results of the bounding analyses performed for NUREG/CR-6023, "Generic Analyses for Evaluation of Low Charpy Upper Shelf Energy Effects on Safety Margins Against Fracture of Reactor Pressure Vessel Materials" (July 1993), the Westinghouse $J_{applied}$ values for Levels A and B were found to be controlling. The methodology employed by Westinghouse for calculating the fracture driving force for Watts Bar Unit 1 is acceptable.

The results of the Westinghouse equivalent margins analysis for Watts Bar Unit 1 and the independent analysis performed by the staff for Levels A and B are shown in Tables 5.1 and 5.2 and Figures 5.2 and 5.3. The tables compare the

Table 5.1 Westinghouse Equivalent Margins Analysis for Watts Bar Unit 1 for Levels A and B* Applied Driving Force ($J_{applied}$) and Material Resistance ($J_{material}$) for K_{It} Determined as per Equation 1.

Analysis	Applied		Material		Met criteria
	$J_{0.1}$	dJ/da	$J_{0.1}$	dJ/da/	
Westinghouse analysis using RPV combined database model	554	330	614	2330	Yes
Staff analysis using RPV combined database model	555	350	599	781	Yes
Staff analysis using RPV base metals model	555	350	854	*	Yes

NOTE: J units in in.-lb/n^2.

* Letter, TVA to NRC, October 15, 1993.

** The entire driving force curve lies significantly below the *J-R* curve as shown in Figures 5.2 and 5.3. If the driving force curve is elevated until an intersection is forced, the slope (*dJ/da*) of the *J-R* curve will be significantly greater than the slope of the applied curve.

Table 5.2 Westinghouse Equivalent Margins Analysis for Watts Bar Unit 1 for Levels A and B* Applied Driving Force ($J_{applied}$) and Material Resistance ($J_{material}$) for K_{It} Determined as per Equation 2.

Analysis	Applied		Material		Met criteria
	$J_{0.1}$	*dJ/da*	$J_{0.1}$	*dJ/da*	
Westinghouse analysis using RPV combined database model	590	345	614	2330	Yes
Staff analysis using RPV combined database model	590	375	599	473	Yes
Staff analysis using RPV base metals model	590	375	854	**	Yes

NOTE: *J* units are in in.-lb/n^2.

* Letter, TVA to NRC, October 15, 1993.

** The entire driving force curve lies significantly below the *J-R* curve as shown in Figures 5.2 and 5.3. If the driving force curve is elevated until an intersection is forced, the slope (*dJ/da*) of the *J-R* curve will be significantly greater than the slope of the applied curve.

Levels A and B applied driving force and material resistance values for both the Westinghouse and staff analyses. The comparison in Table 5.1 is for K_{It} determined as per Equation 1 and the comparison for Table 5.2 is for K_{It} determined as per Equation 2. The comparisons were restricted to Levels A and B since these were found to be controlling. The Code Case N-512 criteria require that $J_{material}$ (*J-R* curve) be greater than $J_{applied}$ at 0.1 in. of crack extension (Criterion 1) and that the slope of the *J-R* curve (*dJ/da*) be greater than the slope of the $J_{applied}$ curve at $J_{applied} = J_{material}$ (Criterion 2). For this determination, the specified *J-R* curve is a conservative representation (mean - 2σ) of the fracture behavior. The safety factor on pressure for $J_{applied}$ is 1.15 for Criterion 1 and 1.25 for Criterion 2.

As shown in the tables and figures, forging 05 complies with the Code Case N-512 criteria for all of the conditions evaluated. The comparisons in the tables apply to the axial flaw case, which was found to be limiting. The figures show the $J_{applied}$ curves only for the case of Equation 1.

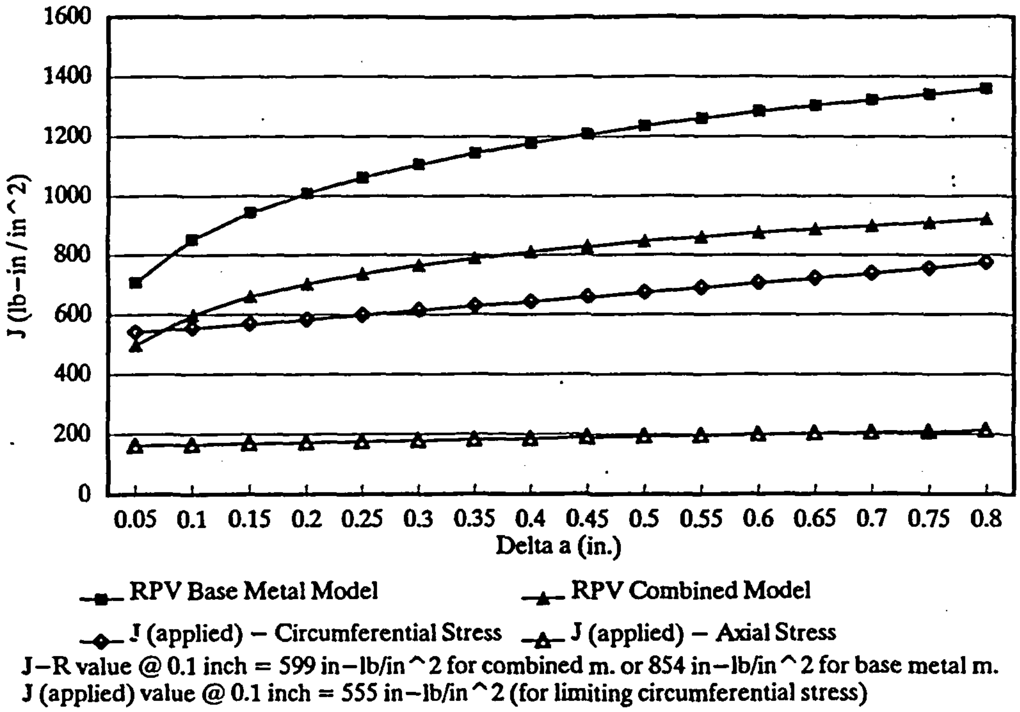

Figure 5.2 Watts Bar 1, Analysis of Intermediate Shell Forging 05 Establishment of Criterion #1, Service Leves A and B, SF=1.15

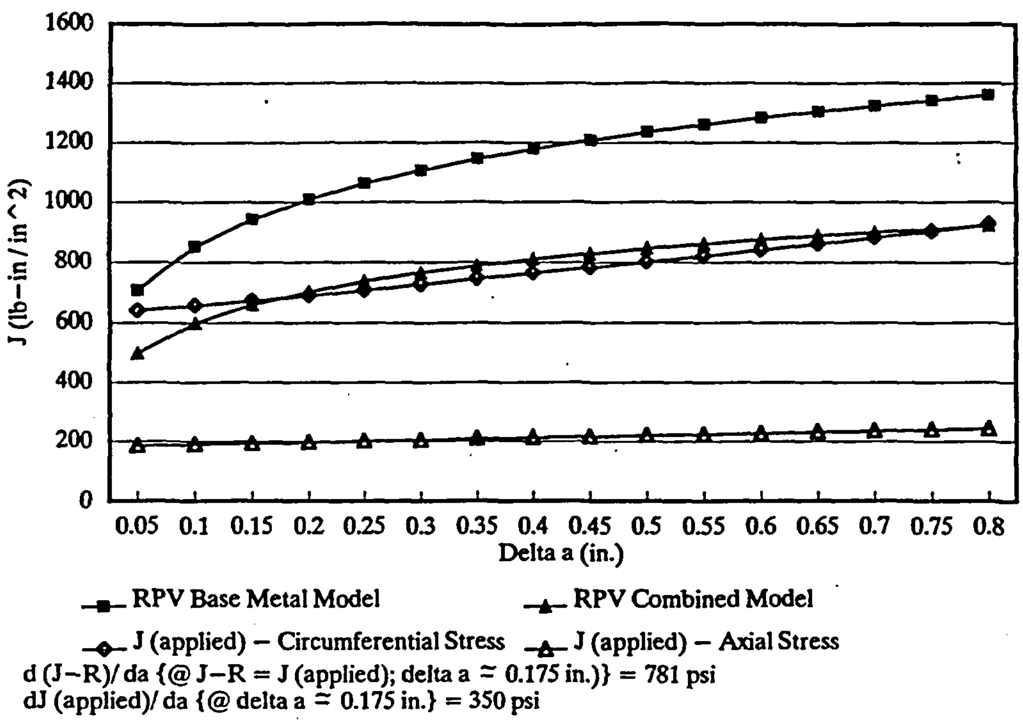

Figure 5.3 Watts Bar 1, Analysis of Intermediate Shell Forging 05 Establishment of Criterion #2, Service Leves A and B, SF=1.25

For completeness, the figures also show the $J_{applied}$ curves for the circumferential flaw case.

A discrepancy was discovered between the Westinghouse and the staff analyses regarding the Level A and B evaluation for Criterion 2 of Code Case N-512. On the basis of the levels cited, it appears possible that the Westinghouse evaluation determined the *J-R* curve slope at the intersection of the *J-R* curve with the applied curve with a safety factor of 1.15. This conflicts with the guidance of Code Case N-512 which sets the determination at the intersection of the curves where the safety factor on the applied curve is 1.25. The staff's analysis addressed this discrepancy and Criterion 2 was still satisfied for Watts Bar Unit 1 as shown in Tables 5.1 and 5.2.

In conclusion, the staff has evaluated the equivalent margins analysis submitted by the applicant. The staff's evaluation supports the following conclusions:

(1) The applicant used methodology, modeling procedures, and acceptance criteria which fall within the scope of Draft Regulatory Guide DG-1023 and ASME Code Case N-512.

(2) The applicant has demonstrated margins of safety for Service Levels A, B, C, and D equivalent to those required by the ASME Code, Appendix G, for intermediate shell forging 05 in the beltine of the Watts Bar Unit 1 reactor vessel.

(3) A discrepancy was discovered between the Westinghouse and the staff analyses regarding the Level A and B evaluation for ductile tearing stability (Criterion 2) of Code Case N-512. The staff's analysis addressed this discrepancy and Criterion 2 was still satisfied for Watts Bar Unit 1.

(4) The previously requested *J-R* curve data for forging 05 were not available. The staff understands that the applicant will submit this information when the first specimens are removed from the reactor vessel. The staff will track this action by TAC M89606. The actual fracture toughness data from these specimens will be used to verify the equivalent margins analysis.

The staff tracked its efforts by TACs M85712 and M85713.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In Table 6.1 of the original Safety Evaluation Report (SER, 1982), the staff listed a number of containment design characteristics. The Final Safety Analysis Report (FSAR), as currently updated to Amendment 87, provides minor variations to some of these parameters. For example, total containment volume was reported as 1,191,400 cubic feet in the SER, and weight of ice in the ice condenser was reported as 2.45E+6 pounds; the FSAR reports these values as 1,144,534 and 2.125E+6, respectively. The staff has reviewed these revised numbers, and concludes that none of them appears to affect conclusions reached in the SER or in Supplements 1-13. In addition, the staff approved the revised weight of the ice (2.125E+6 lb) in SSER 5.

This effort was tracked by TACs M89107 and M89108.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.5 Steam Generator Water Level Trip

Steam Generator Reference Leg Uninsulated Effects

The primary functions of the steam generator water level instrumentation are to initiate a reactor trip during a main feedwater line break (MFLB) event, and to indicate the need for operator action under other accident conditions per Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

By letter dated June 21, 1982, the applicant originally committed to insulate the steam generator (SG) reference leg water level instrumentation. The insulation was intended to provide further assurance that the reference leg water level instrumentation will be accurate under accident conditions. The staff accepted this commitment and reported its acceptance in Supplement 2 to the Safety Evaluation Report (SSER 2). In a subsequent letter dated July 27, 1994, the applicant withdrew its commitment to insulate the SG reference leg. The staff's review of that proposal follows.

The absence of insulation on the reference leg significantly decreases the accuracy of level instrumentation during an MFLB accident. Elimination of insulation was originally addressed in Westinghouse topical report WCAP-13462, "Summary Report - Process Protection System Eagle 21 Upgrade, NSLB, MSS and TTD Implementation - WBN Units 1 and 2," Revision 1, submitted by the applicant on May 23, 1994. In that submittal, the applicant concluded that without the insulation, the SG low-low level setpoint is 24-percent to 27-percent of instrument level span, which is approximately 10 percent higher than the originally determined setpoint for the insulated reference leg. This decrease in instrument accuracy will delay the low-low SG level reactor trip at an MFLB accident. The applicant states that, for the in-containment MFLB event, the high containment pressure setpoint will invariably be reached before the low-low SG level. The former signal will actuate safety injection, and the safety injection signal will, in turn, cause reactor trip.

In FSAR Section 15.4.2.2.2 (as updated to Amendment 86), the applicant evaluated an in-containment MFLB accident based, conservatively, on a reactor trip from the SG low-low water level. The applicant noted that the SG low-low level trip, with the decreased accuracy, will be generated 26 seconds after the feedwater line break; the high containment pressure signal will be generated in less than 1 second for the same accident. The staff has completed its review of the FSAR up to Amendment 87 (see statement in Chapter 1), and has found the analysis acceptable. This analysis is bounding since, in reality, the high containment pressure signal will actuate the series of safety functions associated with the in-containment MFLB event. Since the SG low-level signal is not depended upon, its delay due to lack of insulation on the SG reference leg is of no consequence to the in-containment MFLB accident.

The SG water level signal is also used by the post-accident monitoring system for control room indication. The applicant has classified this signal as Type A per Regulatory Guide (RG) 1.97 (the staff's evaluation was published in SSER 9). The primary function of SG water level as an RG 1.97 variable is to assist the operator in identifying and responding to an SG tube rupture accident. The applicant states that for that accident, reference leg insulation is not needed to assure accuracy of SG water level. The staff concurs with this assessment and, therefore, finds the applicant's proposal of not insulating the steam generator instrumentation reference leg acceptable.

This effort was tracked by TAC M90072.

7.3 Engineered Safety Features Actuation System

7.3.1 System Description

In Amendment 81 to FSAR Section 7.3.2.2.6, the applicant stated that the manual initiation of both steamline isolation and switchover from injection to recirculation following a loss-of-primary-coolant accident are performed at the component level only. Section 4.17 of the Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," requires that manual initiation be provided for each protective action at the system level, regardless of whether means are also provided to initiate the protective action at the component or channel level (Regulatory Guide (RG) 1.62, " Manual Initiation of Protective Actions").

The main steam isolation valves (MSIVs) are used to mitigate the consequences of steamline breaks. Protection logic automatically closes the valves when necessary. Four individual MSIV momentary control switches (one per loop) are mounted on the control board to isolate the main steamlines. The inadvertent manual closure of any single MSIV or the simultaneous closure of all MSIVs both create Condition II events. Remote individual closure of the MSIVs from the control room is required for operational reasons. However, the applicant stated that providing additional manual capabilities which can lead to the inadvertent closure of all MSIVs can lead to an unsafe condition. Therefore, the staff agrees with the applicant's justification for not providing manual initiation of steamline isolation at the system level as is required by IEEE Standard 279-1971.

During the injection mode, the residual heat removal (RHR) pumps take suction from the refueling water storage tank (RWST). On receipt of a low-level signal from the RWST in conjunction with a safety injection (SI) signal and a high-level signal from the containment sump, the RHR pumps realign to take suction from the containment sump. The automatic phase of switchover from injection (suction from the RWST) to recirculation (suction from the containment sump) consists of closing the valves admitting suction from the RWST to the RHR pumps as the valves providing suction from the containment sump begin to open. The operator manually performs the sequence of steps required to complete the switchover operation: opening and closing several valves, starting and stopping pumps, and verifying proper realignments. The Watts Bar design does not provide for manual operation at the system level for the switchover function. The applicant stated that the inadvertent initiation of switchover from injection to recirculation could lead to an undesirable plant transient. In addition, the operator has sufficient time to manually complete the required actions at the component level following the initiation of the switchover. Therefore, the staff agrees with the applicant's justification for not providing manual initiation for the switchover operation at the system level, as required by IEEE Standard 279-1971. By FSAR Amendment 88, the applicant has added such justification for this deviation from IEEE Standard 279-1971 in Section 7.3.2.2.6 of the Watts Bar FSAR. This issue is, therefore, resolved. This effort was tracked by TACs M88698 and M88699.

7.5 Safety-Related Display Information

7.5.2 Post-Accident Monitoring System

In Appendix V of SSER 9, the staff evaluated the post-accident monitoring system. By letter dated May 9, 1994, and in Amendment 81 to the FSAR, the applicant justified additional modifications to the deviations from RG 1.97, Revision 2, concerning post-accident monitoring instrumentation. The staff has reviewed these deviations and concludes that the applicant either conforms to, or has adequately justified the deviations from, the guidance of RG 1.97, Revision 2, as follows:

(1) In RG 1.97, Revision 2, the staff recommends that the detectors for containment area radiation (high range) respond to gamma photons within any energy range from 60 KeV to 3 MeV with an energy response accuracy of 20 percent at any specific photon energy from 0.1 MeV to 1 MeV. The applicant deviates from this in that overall system accuracy for containment area radiation will be within a factor of two over the entire range.

    In RG 1.97, Revision 3, the staff recommends that the detectors respond to gamma photons within any range from 60 KeV to 3 KeV with a dose rate response accuracy within a factor of two over the entire range. The applicant conforms to the recommendations in Revision 3 of RG 1.97 and, therefore, the staff finds this deviation acceptable.

(2) For containment sump level (wide range), the staff recommends in Revision 2 of RG 1.97 a range from the bottom of the containment to the level equivalent to 600,000 gallons. The applicant proposed a containment sump level monitoring system that starts measuring at 6 inches above the containment floor (level tap is located at elevation 703′ 3-3/8″). The range of the instrument is 20 feet (723′ 3-3/8″).

    The total volume of water available to flood the containment after a loss-of-coolant accident (LOCA) is 844,000 gallons, which equals a maximum transient flood level of 720 feet 0 inch. The applicant also noted that for post-accident monitoring, the operator is aware that the zero level actually begins at 6 inches above the floor.

    The applicant stated that the recommended range is fully adequate to monitor the maximum flood level that would be experienced following a LOCA. Further, in Revision 3 of RG 1.97, the staff states that a plant-specific designation for wide-range containment sump level is appropriate where necessary. On this basis, the staff finds this deviation acceptable.

(3) The applicant originally classified main steamline radiation as a Type A, Category 1, variable. After further review of the plant design, the applicant determined that this variable had been overqualified. The applicant has now designated this variable as either Type C or E, Category 2, depending on the location of the instrument. In RG 1.97, Revision 2, the staff does not list which specific variables are to be designated as Type A (and, therefore, Category 1) because such variables are plant specific, and will depend on the required post-accident manual operations that are necessary for accident mitigation. Since main steamline radiation does not provide primary indication for initiation of manual actions following an accident, primary indication is not required to be Type A for Watts Bar. The staff, therefore, finds this change acceptable.

(4) In RG 1.97, Revision 2, the staff recommends that control rod position indication be a Type B, Category 3, variable to monitor for reactivity control. The applicant proposed that this variable be a Type D, Category 3.

The applicant stated that control rod position indication is an indirect variable, providing backup indication for monitoring reactivity control. Neutron flux (Category 1) is a direct variable that allows the operator to determine if reactivity is under control. On the basis of this explanation, the staff finds the Type D designation for control rod position indication acceptable.

(5) In RG 1.97, Revision 2, the staff recommends a range of 50 to 750 °F for quench tank (pressurizer relief tank) temperature. By letter dated August 31, 1990, the applicant proposed a range of 50 to 300 °F for this instrument. This deviation was previously approved by the staff based on a commitment by the applicant to expand the upper range limit to 400 °F. In Amendment 81 to the FSAR, the applicant reflected this deviation (No. 11) from RG 1.97, Revision 2, in accordance with this commitment. The staff finds that the applicant has satisfactorily fulfilled this commitment.

(6) In RG 1.97, Revision 2, the staff recommends a range of 1.0E-6 to 1.0E+3 microcuries/cubic centimeter (μCi/cc) for auxiliary building exhaust vent radiation level—noble gas release. By letter dated August 31, 1990, the applicant proposed the range should be 1.0E-6 to 1.0E-2 μCi/cc. The staff approved. In Amendment 81, a sentence in the original justification was rephrased from, "The receipt of a high radiation level reading on the Auxiliary Building vent monitor shall cause...." to "An accident causing Auxiliary Building radiation level to be high will cause...." This change is only a clarification which does not affect the staff's original approval of this deviation, and it remains acceptable.

(7) In RG 1.97, Revision 2, the staff recommends a range of 1.0E-6 to 1.0E+5 μCi/cc for the condenser vacuum pump exhaust vent (noble gas). The applicant stated that this instrumentation has a range of 3.2E-7 to 3.5E+3 μCi/cc.

The applicant stated that the only credible accident monitored by this variable is a steam generator tube rupture. In the Standard Review Plan (NUREG-0800), the staff recommends that the tube rupture accident be analyzed using the highest isotope concentrations allowed by the Watts Bar Technical Specifications (TSs). The specific activity of the reactor coolant is limited by the TSs to:

(a) less than or equal to 1 μCi per gram (gm) dose equivalent iodine-131, or,

(b) less than or equal to 100/E μCi/gm.

The dose equivalent I-131 is 4.95 times more restrictive that the 100/E limit. The 100/E limit is more conservative and is used for the calculation of instrument range. The highest concentration of mixed noble gas isotope that can be present under the 100/E limit is 9.89E+2 μCi/cc, as determined in TVA Calculation WBNAPS-048. This value is within the proposed range for the condenser vacuum pump exhaust radiation monitor under expected design-basis conditions. On this basis, the staff finds this deviation acceptable.

(8) In RG 1.97, Revision 2, the staff recommends sampling and onsite analysis capability for the reactor coolant system, as well as for other selected systems. The applicant identified deviations in the post-accident sampling system from RG 1.97, Revision 2, in which the staff recommends a range of 0 to 2000 cc per kilogram (kg) for the dissolved total gas, and a range of 0 to 20 parts per million (ppm) for the dissolved oxygen. The applicant stated that the dissolved total gas concentration can be monitored down to 100 cc/kg rather than zero, and the dissolved oxygen level detected down to 1 ppm rather than zero.

The staff previously reviewed and approved the applicant's post-accident sampling facility as part of the review of NUREG-0737, Item II.B.3 (see Section 9.3.2 of the SER, SSER 3, and SSER 5). The staff finds that indication of dissolved total gas and dissolved oxygen to the lowest levels in the current post-accident sampling system are sufficient to provide necessary information following an accident, and are, therefore, acceptable.

(9) In RG 1.97, Revision 2, the staff requires that proper redundancy exist for variables classified as Category 1. Furthermore, where the failure of one accident-monitoring channel results in ambiguous information that could lead operators to defeat or fail to accomplish a required safety action, additional information should be provided to allow the operators to deduce the actual plant condition. One channel of the Watts Bar core exit temperature (Type A, Category 1) indication is subject to direct failure as a result of a specific pipe break jet impingement and/or pipe whip impact on the cable/conduit routed near the safety injection accumulator cold-leg injection line in loop 1. However, should such a break occur, the affected channel is expected to fail open and not give erroneous indication that could confuse the operators.

Assuming the loss of one channel of core exit temperature due to this specific pipe break plus a single failure of the redundant channel, in RG 1.97, Revision 2, the staff recommends the addition of an identical channel or a diverse channel that bears a known relationship to core exit temperature. For this scenario, the applicant stated that the operators will be able to compensate for the lost channels and correctly assess the accident scenario by using the indications provided by reactor vessel level, reactor coolant system (RCS) pressure, RCS temperatures T-hot and T-cold, and containment pressure and temperature. Since the applicant can use other Category 1 indications to compensate for the loss of core exit temperature indication, the staff finds this alternative consistent with the criteria of RG 1.97, Revision 2 and, therefore, acceptable.

(10) For the reactor vessel level instrumentation, Revision 2 of RG 1.97 recommends a range from the bottom of the core to the top of the reactor vessel. For the static mode (pumps not running), the applicant proposes a range from 0 percent, representing reactor vessel empty, to 100 percent, representing reactor vessel full. However, for the dynamic mode (pumps running), the applicant deviates from the recommendation by proposing a range of 20 percent to 100 percent.

Revision 3 of RG 1.97 recommends that the reactor vessel level instrumentation cover a range from the bottom of the hot leg to the top of the vessel. The bottom of the hot leg is located above the 20 percent level indication from the bottom of the core to the top of the vessel. The Watts Bar reactor vessel level instrument range is thus greater than the required range recommended in Revision 3 of RG 1.97. The applicant conforms to the recommendations of Revision 3 of RG 1.97 and, therefore, the staff finds this deviation acceptable.

This effort was tracked by TACs M88698 and M88699.

7.7 Control Systems Not Required for Safety

7.7.8 Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)

The AMSAC signal was added as required by 10 CFR 50.62 as an automatic initiation signal to start the turbine-driven and motor-driven auxiliary feedwater (AFW) pumps. The staff's evaluation was previously published as Appendix W to SSER 9. The staff review of FSAR Amendment 81 found that this signal was not added to the logic diagram for the AFW system shown in FSAR Figure 7.3-3, Sheet 2. By Amendment 88, the applicant revised Figure 7.3-3, thereby resolving this issue. This effort was tracked by TACs M88698 and M88699.

7.8 NUREG-0737 Items

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

The staff's review of this item was published in the SER and SSER 5. By letter dated November 7, 1994, the applicant revised the original design by relocating the accelerometers (for valve position indication) to downstream of the relief valves from upstream. The purpose of the relocation is to environmentally qualify the accelerometers. This revision does not change the function of the position indication hardware, nor does it alter the staff's review of this issue previously published.

This effort was tracked by TAC M90992.

8 ELECTRICAL POWER SYSTEMS

The following sections are based on the staff's efforts tracked by TACs M89109 and M89110.

8.2 Offsite Electric Power Systems

8.2.2 Compliance With GDC 17

The material that follows revises the discussion in Supplement 13 to the Safety Evaluation Report (SER) (SSER 13) editorially, but does not change the original conclusions:

In SSER 13, the staff incorrectly described how offsite power was supplied to the onsite non-Class 1E distribution system. Offsite power is normally supplied to the onsite non-Class 1E distribution system from the main generator through a 22.5-to-6.9-kV transformer (unit station service transformer, USST) to the 6.9-kV unit boards and 6.9-kV RCP boards, and from the 161-to-6.9-kV transformers (common station service transformers A and B) to the 6.9-kV common boards. For any unit generator trip, offsite power to the unit boards and RCP boards is automatically transferred from the normal supply to the two common station service transformers A and B. Automatic transfers of the non-Class 1E 6.9-kV boards between these two transformers occur on undervoltage conditions.

8.2.2.2 Minimizing the Probability of Losing All AC Power

The material that follows revises the discussion in SSER 13 editorially, but does not change the original conclusions:

(3) Testing of the Automatic Transfer From the Normal to the Preferred Offsite Circuit

In SSER 13, the staff stated that the automatic transfer from the normal common station service transformers A and B to the preferred common station service transformers C and D would be eliminated by a design change notice (M-12051-A). This statement was incorrect in that the automatic transfer to be eliminated was from the normal sources (USSTs) for the 6.9-kV shutdown boards and not from common station service transformers A and B, which are only sources for the shutdown boards during shutdown conditions.

(5) Separation Between Offsite Power Transformers and Preferred Offsite Circuits

In SSER 13, the staff stated that the offsite circuits from the common station service transformers C and D are separately routed underground. This statement was only partially correct in that these offsite circuits are routed separately underground for part of the run before exiting to separate overhead cable trays and conduits for the rest of the run.

8.3 Onsite Power System

8.3.1 Onsite AC Power System Compliance With GDC 17

8.3.1.2 Low and/or Degraded Grid Voltage Condition

The material that follows revises the discussion in SSER 13.

(1) Allowable Technical Specification Limits for the Inverse Time Delay Relay

In SSER 13, the staff stated that Technical Specifications should require, for example, that the capability of the relays not to trip when subjected to a voltage of 75 percent for 30 seconds be demonstrated. The staff implied that this had been included in the draft Technical Specifications. This statement was wrong. Instead, the staff required that the setpoints and allowable values for the load-shed and diesel start relays be included in the plant's Technical Specifications to resolve the concerns.

8.3.1.10 No-Load Operation of the Diesel Generator

The material that follows revises the discussion in SSER 13 editorially, but does not change the original conclusions:

In SSER 13, the staff stated that the applicant, for all situations, has loads continuously available to the operator that exceed 50 percent of the continuous rated load for the diesel generator. This statement was incorrect in that only during non-accident conditions, such as during normal testing, does the operator have available loads to add that exceed 50 percent of continuous rating with only Unit 1 licensed.

8.3.1.12 The Capability and Independence of Offsite and Onsite Sources When Paralleled During Testing

As discussed in SSER 13, the staff in a March 28, 1994, letter transmitted questions to the applicant pertaining to an emergency diesel generator's (EDG's) response to a loss-of-offsite-power (LOOP) condition while it is paralleled to the grid. The staff's concerns are resolved as follows:

(1) The applicant in a February 7, 1994, letter stated that fault conditions associated with the normal offsite source to which the EDG is connected are indicative of a LOOP condition. The staff requested a discussion of the specific fault conditions and how they would serve as indicators of a LOOP.

In a letter dated June 29, 1994, the applicant stated that five specific fault conditions would directly trip both the output breaker for the EDG being tested and the normal supply breaker for the shutdown board. The five conditions which are either a direct indication that a LOOP event has occurred or is about to occur on the 161-kV supply to the shutdown board are:

- common station service transformer (CSST) differential
- CSST overcurrent
- CSST neutral overcurrent
- CSST sudden pressure

- tripping of the 161-kV feeder breaker at the Watts Bar Hydro Plant switchyard.

The first four conditions lead to tripping of the secondary side breakers for the transformer pair (CSSTs A and D or CSSTs B and C) tied to the same 161-kV line. A trip command is also sent to the associated 161-kV feeder breaker in the hydro switchyard. A fault condition that trips the 161-kV feeder breaker in the switchyard also trips the secondary side breakers of the associated CSST pair.

When an EDG is being tested parallel to the grid through the shutdown board's normal supply breaker, automatic transfer from the normal offsite power source to the alternate offsite source is administratively controlled. In the event that any of the preceding five fault conditions occur on the associated 161-kV supply, both the output breaker for the EDG being tested and the normal feeder breaker trip. Consequently, the associated shutdown board experiences a loss of voltage that is equivalent to a LOOP condition.

On the basis of the information provided, the staff's concern is resolved.

(2) In the February 7, 1994, letter, the applicant also stated that if the offsite source for the associated shutdown board was through the alternate feeder, a LOOP condition would not result in the output breaker of the EDG being tested directly tripping. In this scenario, the EDG overcurrent relays would prevent the EDG from being overloaded. The staff requested a discussion of whether these relays lock out and of any associated manual action in response to a lockout condition (if applicable).

In the letter dated June 29, 1994, the applicant stated that when an EDG is being tested in parallel with the alternate 161-kV source and if any of the fault conditions discussed in concern (1) above should occur on the alternate source, the associated CSST secondary side breaker would trip, but not the EDG output breaker or shutdown board supply breaker. When the secondary side breaker trips, the EDG remains connected to the shutdown board. The EDG overcurrent relays are enabled and would trip the EDG output breaker in the event of an overload.

The applicant stated that the overcurrent relays are Westinghouse Type SC relays which provide instantaneous pickup and dropout with a self-resetting feature. Since these relays do not lock out, there is no manual action required to reset these relays. These overcurrent relays are disabled unless the EDG is being tested with both its output breaker closed and either the normal, alternate, or maintenance supply breaker also closed. If an overload should occur while the EDG is being tested, the overload relays directly trip the EDG output breaker through contact logic which utilizes the shutdown board's normal, alternate, and maintenance supply breakers' auxiliary contacts. This response adequately resolves this issue.

8.3.2 Onsite DC System Compliance With GDC 17

8.3.2.5 Non-Safety Loads Powered From the DC Distribution System and Vital Inverters

8.3.2.5.1 Transfer of Loads Between Power Supplies Associated With the Same Load Group but Different Units

In SSER 13, the staff stated that review of a sample Watts Bar procedure (SOI-211-01) to control the use of the alternate feeders had raised several questions and that those questions had been transmitted to the applicant in a letter dated March 28, 1994. The staff's concerns are resolved as follows:

(1) On page 6 of Procedure SOI-211-01, a note stated that a Technical Specifications LCO (limiting condition for operation) action may be required if the alternate feeders for breaker control power are used. Since this was counter to the applicant's commitment to take positive action per the Technical Specifications, the staff asked the applicant to discuss this discrepancy.

In the letter dated June 29, 1994, the applicant stated that the note was revised to indicate that the use of the alternate dc control power does not require any action related to a technical specification. In followup discussions with the staff, the applicant stated the following:

- The batteries have adequate capacity to carry all the alternate loads.
- Breaker control power alignment is checked every 7 days.
- Use of alternate control power is coordinated with plant management.
- Only one or two loads are supplied from an alternate source.
- An alternate source is used only for an unusual condition (2 to 3 times in the plant's life).
- Breaker alignment is returned to normal as soon as the problem is fixed.

On the basis of this information, the staff's concern is resolved.

(2) The same note on page 6 of the procedure stated that the alternate source of control power for a Train A shutdown board would be Train B. Since this was counter to the applicant's September 13, 1991, statement that loads would only be transferred within the same train but different unit, the staff asked the applicant to discuss this discrepancy.

In the letter dated June 29, 1994, the applicant stated the note was wrong as both the normal and alternate dc control power sources for each 6.9-kV shutdown board are supplied from the same train as described in FSAR Section 8.3.2.2. Procedure SOI-211.01 has been revised to correct this error.

This response adequately resolves this issue.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDCs 2 and 4

8.3.3.1.4 Use of Waterproof Splices in Potentially Submersible Sections of Underground Duct Runs

In SSER 13, the staff stated that the Watts Bar design basis does not permit splices to be installed in raceways and that the applicant committed to make that statement in an amendment to the FSAR. Contrary to this, the applicant, in a response to issues raised in Inspection Reports 50-390/93-74 and 50-391/93-74, described two methods of splicing cables in open cable trays as allowed by Watts Bar Standard Drawing SD-E12.5.9. This is also in opposition to the staff's prohibition against splices in cable raceways which is centered on the prevention of fires caused by improper splicing. If splices are used in raceways that are part of the general raceway system, then an analysis justifying their use should be made and documented in the FSAR as recommended by Revision 1 to RG 1.75, "Physical Independence of Electric Systems." The staff transmitted this concern to the applicant in a March 28, 1994, letter. Resolution of the staff's concerns are as follows:

(1) Originally, in a request for additional information (RAI) dated June 20, 1991, the staff was concerned about the use of underground splices in potentially submersible sections of duct runs.

In a September 13, 1991, letter, the applicant stated that these underground splices were located in manholes of the underground duct runs that serve the same function as junction boxes in conduit runs. The manholes are enclosed structures with very limited space. Support and protection is provided in the manholes for the cables and cable splices. Redundant divisions are either installed in separate manholes or in a common manhole with a concrete barrier between the divisions. Sump pumps with level switches for automatic operation are located in the manholes to prevent water accumulation. Manholes will also be included in the plant's preventive maintenance program with an annual inspection of the sump pump operability and possible flooding. This information is contained in FSAR Section 8.3.1.2.3.

On the basis of this information, the staff's concern is resolved.

(2) Analysis/justification for the use of splices in the general raceway system

In a letter dated June 29, 1994, the applicant stated that RG 1.75 is not applicable to Watts Bar and that there has been no commitment to comply with that regulatory guide. In addition to the splices in manholes, the applicant discussed two specific instances of splicing in wireway extensions in the plant. Splices are used in the trenches (walkways) beneath the main control boards and on the outboard side of primary containment electrical penetrations in the reactor building annulus. The cable splices in the walkways are typically for low-energy cables and are used for terminations to the control boards. The walkways are sealed with firestops and any damage would be limited to one train. The cable splices for the penetration pigtails are located in junction boxes attached to the penetration nozzle for cables from conduits and in splice boxes for cables from cable trays. The splice boxes consist of a solid-bottom cable tray with a cover and a firestop seal located about 8 feet from the penetration nozzle which prevents a fire caused by a splice failure from spreading to other sections of the cable tray.

In other general areas of the plant, splices in cable trays are not permitted except for extraordinary situations where they must be reviewed and approved by the plant's site engineering organization. For the few situations that have been approved, splices in cable trays have been installed in accordance with either of two detailed methods shown on Watts Bar Standard Drawing SD-E12.5.9, "Cable Splicing of Installed Cables in Cable Trays." One method utilizes a rigid conduit sleeve placed over the splice with a fire seal at each end. For the other method, a solid metal barrier is installed between each spliced cable section and other cables in the tray with covers on the top and bottom and a fire seal at each end of the tray section containing the splice. Installation, testing, and documentation requirements are contained in TVA Electrical Engineering General Engineering Specification G-38, "Installation, Modification, and Maintenance of Insulated Cables Rated up to 15,000 Volts."

Although the staff's concern is resolved for the two specific instances discussed above and for any extraordinary situations in general plant areas on the basis of the preceding information, splices should not be used in certain important situations. The staff was recently informed by an NRC resident inspector at Watts Bar that numerous cable splices have been used throughout the emergency diesel generator output cable runs. These splices are located in cable trays and in manholes. The staff finds this unacceptable because of the relative importance of these cables. This is an open item pending further discussions with the applicant; it will be tracked by TACs M89109 and M89110.

8.3.3.2 Compliance With GDC 5

8.3.3.2.1 Sharing of DC Distribution Systems and Power Supplies Between Units 1 and 2

The material that follows replaces the first paragraph of this section contained in SSER 13, is editorial, and does not change the original conclusion:

In the SER, the staff stated that the Class 1E dc system for Unit 1 supplies power to vital buses I and II for Unit 2, and the Class 1E dc system for Unit 2 supplies power to vital buses III and IV for Unit 1. This was wrong; the Unit 1 Class 1E dc system supplies Unit 1 buses I and II and the Unit 2 Class 1E dc system supplies Unit 2 buses III and IV. The Class 1E dc systems are common to both units and the dc systems are shared in all modes of plant operation.

8.3.3.3 Physical Independence (Compliance With GDC 17)

(5) Separation Between Open Cable Trays and Conduits

In SSER 13, the staff stated that there were several differences between RG 1.75 and the Watts Bar General Design Criterion WB-DC-30-4, "Separation/Isolation," pertaining to the electrical separation for divisional open cable trays and conduits. The staff also noted that WB-DC-30-4 allows separation distances even smaller than those supported by the latest industry guidance contained in ANSI/IEEE Standard 384-1992, "Standard Criteria for Independence of Class 1E Equipment and Criteria." Because of these differences, the staff stated that the applicant's case-by-case justification (supported by analysis/test) for deviation from staff and industry guidance would be reviewed.

The staff transmitted its concern about possible inadequate separation between divisional open cable trays and conduits in a March 28, 1994, letter to the applicant. Subsequently, NRC Inspection Report 50-390/94-18 reported a situation in which a cable in free air was in contact with a conduit from the redundant train. In response, the applicant issued Design Change Notice S-29587-A to add specific requirements for free air cable-to-conduit separation to raceway separation drawings. A comparison between the applicant's requirements and those in industry guidance (IEEE Standard 384-1992) revealed that the applicant allowed a lower value for minimum vertical separation (1 inch versus 3 inches) for free air cable below a conduit.

Also Region II staff questioned the acceptability and adequacy of the applicant's analyses used to justify case-by-case separation deviations from WB-DC-30-4. Those analyses were based on the availability of backup redundancy for the affected component(s), the types of power sources involved, and the protective action of associated fault-detecting devices in lieu of actual test results.

In a followup conference call on May 12, 1994, the staff expanded upon the concerns to include free air cable-to-conduit separation and adequacy of case-by-case analysis. The applicant was asked to describe in detail all the electrical separation criteria in use at the plant and to justify (on the basis of test results wherever possible) each deviation from RG 1.75 and industry guidance and to add that description to the plant's FSAR.

In a July 29, 1994, response, the applicant provided a general discussion pertaining to Watts Bar's deviation from RG 1.75. Tests which had justified electrical separation at several other nuclear power plants were discussed and Watts Bar electrical separation requirements wre justified according to IEEE Standard 384-1974, "Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits," or applicable industry tests. Also proposed FSAR changes to describe separation distances between Class 1E open cable trays and conduits were submitted.

As a result of the staff's review of the July 29, 1994, letter, further information was requested in an August 22, 1994, letter. This remains an open issue pending further staff review of the information to be provided, and will be tracked by TACs M89109 and M89110.

8.3.3.5 Compliance With GDC 18

8.3.3.5.1 Compliance With Regulatory Guides 1.108 and 1.118

In SSER 13, the staff stated that the applicant deleted from the FSAR compliance statements for RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," which has been withdrawn by the staff and instead addressed compliance with RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," Revision 3. This was considered an open item pending staff review. Subsequent staff review raised the following concerns:

(1) Class 1E Standby Power System Testing

In Section 8.1.5.3 of FSAR Amendment 86, the applicant indicated that the Watts Bar electrical system design does not fully comply with Position C.2.2.6 of RG 1.9 (Revision 3). Position C.2.2.6 of RG 1.9 (Revision 3) recommends that a combined SIAS (safety-injection-actuation-signal) and LOOP (loss-of-offsite-power) test be performed (as part of preoperational and periodic testing programs) to demonstrate that the emergency diesel generator can satisfactorily respond to a LOOP in conjunction with SIAS in whatever sequence they might occur (e.g., loss-of-coolant accident (LOCA) followed by delayed LOOP or LOOP followed by LOCA). In clarification, the applicant stated:

> The design basis at WBN [Watts Bar] is a simultaneous LOOP/LOCA, not LOOP followed by LOCA. Although there are some design features to meet the effects of LOOP followed by LOCA, there is no analysis to demonstrate the design will meet the DG [diesel generator] voltage and frequency requirements.

On the basis of this clarification, the staff understood that an actual simulated LOCA followed by a LOOP or a LOOP followed by a LOCA test will not be performed. In place of an actual simulation, the staff understood that the operability of the additional design features installed to meet the effect of LOOP before or after LOCA will be demonstrated by a number of overlapping tests. These tests are to be included as part of the Watts Bar preoperational and periodic test programs.

In addition, the applicant submitted the following information as part of the Watts Bar Technical Specifications and as part of discussions with the staff:

- A simultaneous LOOP/LOCA event will be demonstrated by simulating an actual LOOP and LOCA signal.

- The capability of the diesel generator to start and operate at no load will be demonstrated by test.

- The operability of the logic or design features which perform the following functions will be included as part of preoperational and periodic test programs:

    - With the standby diesel generator operating at no load, Class 1E buses are deenergized, loads are shed from the buses, and the standby diesel generator energizes permanently connected loads.

    - After a LOOP followed by a delayed LOCA, loads already sequentially connected to Class 1E buses (which are not required for an accident) are disconnected.

    - After a LOOP followed by a delayed LOCA, loads already sequentially connected to the Class 1E buses (which are required for an accident) remain connected.

    - After a LOOP followed by a delayed LOCA, loads awaiting sequential loading that are required for an accident are either sequentially loaded as a result of the non-accident loading sequence or have their sequential timers reset to time zero from which they are then sequentially loaded in accordance with the accident sequence.

- The capability of the standby diesel generator to supply worst-case loading which may occur due to automatic load sequencing if there is a LOOP followed by a delayed LOCA will be demonstrated as part of pre-operational and periodic test programs. Program testing will include the simultaneous connection and disconnection of two diesel generator loads for each diesel generator. The two diesel generator loads combined will exceed the worst-case loading which may occur due to automatic load sequencing if there is a LOOP followed by a delayed LOCA.

Criteria III and XI of 10 CFR Part 50, Appendix B, require that (1) measures be provided for verifying or checking the adequacy of design by design reviews, by the use of alternative or simplified calculational methods, or by the performance of a suitable testing program and (2) a test program be established to ensure that systems and components perform satisfactorily and that the test program includes operational tests during nuclear power plant operation.

The staff concludes that a preoperational and periodic test program which includes the testing described above will adequately demonstrate the capability of the additional design features installed to meet the effect of LOOP before or after a LOCA. The proposed testing, therefore, conforms to the requirements of Criteria III and XI of 10 CFR Part 50, Appendix B (defined above) and is acceptable. On the basis of this information and the design commitment to comply with the recommendations of Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," Revision 2, and IEEE Standard 338-1977, "IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," documented in Section 8.1.5.2 of FSAR Amendment 86, the staff concludes that there is reasonable assurance that the additional design features installed to meet the effect of LOOP before or after a LOCA will be appropriately tested as part of the Watts Bar preoperational and periodic test programs and is, therefore, acceptable.

In addition, the staff has initiated a generic evaluation into the capacity and capability of safety systems to respond to a non-simultaneous LOOP/LOCA event. Results of this generic evaluation will be imposed as appropriate on the Watts Bar design as well as on other plant designs.

(2) Testing Diesel Generator Full Load Rejection Capability

In Section 8.1.5.3 of FSAR Amendment 86, the applicant indicated that the Watts Bar electrical system design does not comply with Position C.1.3 of RG 1.9 (Revision 3). Position C.1.3 of RG 1.9 (Revision 3) recommends that the predicted loads to be connected to the diesel generator should not exceed the continuous rating of the diesel generator unit. In clarification, the applicant stated:

> Revision 2 [R2] of RG 1.9 Position C2 required the predicted loads not to exceed the short time rating. This position has required the predicted loads not to exceed the continuous rating. WBN [Watts Bar] diesel generators load assignment was based on the RG 1.9 R2 limit.

On the basis of this indicated noncompliance and clarification, the staff understood that the load that would be rejected during testing (i.e., full-load rejection test) would continue to be equal to the diesel generator's short-time rating (4850 kW).

However, in Section 8.1.5.3 of FSAR Amendment 86, the applicant indicated that the Watts Bar electrical system design will be in full compliance with Position C.2.2.8 of RG 1.9 (Revision 3). Position C.2.2.8 of RG 1.9 (Revision 3) recommends that a "full-load rejection test" be performed (as part of preoperational and periodic testing programs) to demonstrate the Class 1E standby diesel generator's capability to reject a load equal to 90 to 100 percent of its continuous rating and verify that the voltage requirements are met and that the Class 1E standby diesel generator will not trip on overspeed. In addition, in Appendix 8D (Section 6.3 of IEEE Standard 387) of FSAR Amendment 86, the applicant indicated that the Watts Bar electrical system design does not fully comply with Section 6.3.4 of IEEE Standard 387-1984, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations." Section 6.3.4 requires that the load rejection test be conducted from the short-time rating.

On the basis of this indicated compliance with RG 1.9 and noncompliance with IEEE industry recommendations, the staff understood that the load that would be rejected during testing (i.e., full-load rejection test) and would be between 90 and 100 percent of the diesel generator's continuous rating (between 3960 and 4850 kW).

Because the electrical system load assignment is based on the diesel generator's short-time rating (Section 8.1.5.3 of FSAR Amendment 86 (page 8.1-12)), the staff was concerned that the capacity and capability of diesel generators to reject loads greater than 90 percent of their continuous rating (between 3960 and 4850 kW) may not be adequately demonstrated in accordance with the defined requirements of Criteria III and XI of 10 CFR Part 50, Appendix B.

In response to the inconsistency described above and the staff's concern, the applicant indicated the following:

- Automatically sequenced accident loads remain below the continuous rating of the diesel.

- Use of manual actuation in accordance with plant emergency operating procedures could result in loading to 101 percent of the diesel generator's continuous rating.

- Loading of non-safety loads during an accident are limited to the short-time rating of the diesel generator.

- The diesel generator's load rejection capability would be demonstrated as part of preoperational and periodic testing by using a load equal to between 90 and 100 percent of the diesel generator's continuous rating.

Criteria III and XI of 10 CFR Part 50, Appendix B, require that (1) measures be provided for verifying or checking the adequacy of design by design reviews, by the use of alternative or simplified calculational methods, or by the performance of a suitable testing program and (2) a test program be established to ensure that systems and components perform satisfactorily and that the test program include operational tests during nuclear power plant operation.

The staff concludes that the capacity and capability of the diesel generators to reject a complete loss of load without tripping, mechanical damage, or harmful overstresses will be demonstrated in accordance with the guidelines of Regulatory Guide 1.9 (Revision 3) as part of the preoperational and periodic test programs. The design, therefore, meets the requirements defined above of Criteria III and XI of 10 CFR Part 50, Appendix B, and is acceptable.

(3) Non-Class 1E Circuitry Used for Transmitting Signals Needed for Starting Diesel Generators

In Section 8.1.5.3 of FSAR Amendment 86, the applicant indicated that the Watts Bar electrical system design conforms to the intent of Position C.2.2.5 of RG 1.9 (Revision 3). Position C.2.2.5 of RG 1.9 (Revision 3) recommends that an "SIAS Test" be performed (as part of preoperational and periodic testing programs) to demonstrate that, on a safety injection actuation signal (SIAS), the Class 1E standby diesel generator starts on the auto-start signal from its standby conditions, attains the required voltage and frequency within acceptable limits and time, and operates on standby for at least 5 minutes. In clarification, the applicant stated:

> The diesel generators associated with the nuclear unit affected by the SI [safety injection] event are started by 1E circuits. However, the starting of the diesel generators of the non-SI unit is implemented with a non-1E circuit (common start circuit). The intent of this position is to have all the DGs started in case there is a loss of off-site power (LOOP). WBN [Watts Bar] meets this precautionary requirement with the common start circuit. In the event of a LOOP, the 1E LOOP circuits also start the DGs, independent of the common start circuit.

On the basis of this clarification, the staff understood that the circuitry used to supply the diesel generator start signal as a result of an accident is non-Class 1E for two of the four diesel generators. The start circuitry will be classified non-Class 1E for the two diesel generators on the unit that is not experiencing an accident and will be classified Class 1E for the two diesel generators on the unit that is experiencing the accident. The Class 1E circuitry for LOOP and SIAS start of the two diesel generators on the unit that has experienced the accident will be included as part of the Watts Bar preoperational and periodic test programs. The Class 1E circuitry for LOOP start of the two diesel generators on the unit that has not experienced an accident will be included as part of the Watts Bar preoperational and periodic test programs. In addition, the non-Class 1E circuitry for SIAS start of the two diesel generators in the unit that has not experienced an accident will be included as part of the Watts Bar preoperational and periodic test programs.

In addition, during telephone discussions with the staff, the applicant indicated sufficient electric power will be available to safety system loads so that they will be capable of performing their required safety function in accordance with the accident analysis for the Watts Bar plant for the following defined postulated condition:

- single failure of either one of two diesel generators on the unit that has experienced an accident,

- start and loading of accident loads per design to the remaining diesel generator on the unit that has experienced an accident,

- failure of the non-Class 1E accident start circuitry to the two diesel generators on the unit that has not experienced an accident,

- start per design of the two diesel generators on the unit that has not experienced an accident (by the Class 1E LOOP start circuitry) assuming a loss of offsite power for both units either at the same time as the accident or at some time before or after the accident, and

- loading of accident loads per design on the unit that has not experienced an accident (by the Class 1E loading circuitry) assuming a loss of off-site power for both units either at the same time as the accident or at some time before or after the accident.

General Design Criterion (GDC) 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E ac standby power supply (the diesel generator power supply) have sufficient capacity and capability to permit safety systems to perform their required safety functions.

For a plant system design which conforms to the accident analysis requirements defined above, the staff concludes that the onsite Class 1E standby power system will have sufficient capacity and capability to permit safety system loads to perform their required safety function. The design, therefore, conforms to the requirements defined above of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

In addition, the staff has initiated a generic evaluation into the capacity and capability of safety systems to respond to a nonsimultaneous LOOP/LOCA event. Results of this generic evaluation will be imposed as appropriate on the Watts Bar design as well as on other plants.

9 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.2 Process Sampling System

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

In Safety Evaluation Report (SER) Supplement 3 (SSER 3), the staff stated that the postaccident sampling system met all of the 11 criteria of Item II.B.3 of NUREG-0737 and was, therefore, acceptable. The staff stated, however, that before restart following the first refueling outage, the applicant will be required to submit a final procedure for estimating the degree of core damage. As stated in the original SER (1982), this will be assured by a license condition (proposed License Condition 19).

In SSER 5, the staff revised this requirement, reasoning that since there was a 5-year delay in licensing, the applicant should submit the procedure at an earlier date. In response, the applicant submitted the procedure by letter dated June 10, 1994. This resolves the staff's concerns and proposed License Condition 19 is deleted.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.5 Engineered Safety Features Ventilation Systems

In the SER, the staff evaluated the ventilation systems for the essential raw cooling water (ERCW) pumping station (intake structure). In that evaluation the staff made the following statement:

> At the essential raw cooling water (ERCW) pumping station, the ERCW pumps are cooled by natural convection; electrical and mechanical equipment are cooled mechanically. Failure of one of the two ventilation fans in a mechanical equipment room will not prevent the operation of any safety-related equipment. Failure of the electrical equipment room ventilation subsystem will not affect any safety-related components or functions.

Although the SER never mentions that the ventilation systems for the pumping station are safety related, it may be implied from the preceding SER statement that the ventilation fans for the mechanical equipment room are safety related because they meet the single-failure criterion. As a matter of clarification, none of the ventilation systems for the pumping station are safety related and the failure of both mechanical equipment room ventilation fans will not prevent the operation of any safety-related equipment. Because this is considered a clarification of the original SER, the conclusions reached by the staff in that SER are still valid and the systems are still acceptable. This effort was tracked by TAC M90253.

10 STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features

10.4.7 Condensate and Feedwater System

In the original Safety Evaluation Report (SER, 1982), the staff stated that a feedwater isolation signal was initiated by a high-high steam generator level, an engineered safety feature (ESF) (safety injection) actuation signal, or a reactor trip. In Amendment 82, the applicant revised Final Safety Analysis Report (FSAR) Section 10.4.7 to identify a new feedwater isolation signal and to clarify the isolation signal generated by a reactor trip. The applicant noted that a high-flood-level detection in either the south or north main steam valve (MSV) vault rooms also generates a main feedwater isolation signal. In addition, the applicant clarified that the main feedwater isolation would only be generated by a reactor trip if the reactor trip is coincident with a low reactor coolant average temperature (low T_{ave}) signal.

The new feedwater isolation signal initiated by high-flood-level detection in the MSV vault rooms was added to prevent the submergence of equipment governed by 10 CFR 50.49 (environmental qualification) located in the MSV vault room in the event of a double-ended main feedwater line break. The flood detectors consist of three safety-grade-level switches in each room with a 2-out-of-3 logic to provide channelized inputs to trains A and B for feedwater isolation. The staff is revising this section of the SER to make the description of feedwater isolation signals consistent with actual plant design. This revision does not affect the conclusions reached in Section 10.4.7 of the original SER.

During its review of FSAR Amendment 82, the staff noted an unrelated error in the SER. In the SER, the staff stated that the main feedwater regulation valves will close within 5 seconds of receipt of a feedwater isolation signal and that the main feedwater isolation valves will close within 6.5 seconds of receipt of the isolation signal. According to the FSAR, both the feedwater regulation valves and feedwater isolation valves will close within 6.5 seconds of initiation of the feedwater isolation signal. The staff could not determine the actual origin of this discrepancy, but assumes that it was probably related to actual valve stroke times versus time to close after generation of a feedwater isolation signal. The accident and containment analyses are based on a closure time 6.5 seconds from the initiation of the feedwater isolation signal. Therefore, the staff concludes that 6.5 seconds is acceptable for both valves and considers this a matter of clarification of the original SER. The conclusions reached in Section 10.4.7 of the original SER are, therefore, still valid.

The staff's efforts were tracked by TACs M88694 and M88695.

10.4.9 Auxiliary Feedwater System

In the SER, the staff stated that the motor-driven auxiliary feedwater (AFW) pumps were designed to deliver 470 gallons per minute (gpm) each and the turbine-driven AFW pump was designed to deliver 940 gpm to the steam generators. These flow rates were based on the original FSAR that identified the

AFW pump design flows as 500 gpm (30 gpm during recirculation) and 990 gpm (50 gpm during recirculation) for the motor- and turbine-driven AFW pumps, respectively. The staff also stated that either of these pumps could meet the minimum flow requirement of 470 gpm to two steam generators. Since the SER was published, the applicant has conducted a number of design reviews and AFW pump tests, and has modified the AFW pumps, resulting in new design-basis flow rates for the pumps and new minimum flow requirements.

Therefore, in FSAR Amendment 71 the applicant gave the new minimum flow requirement as 410 gpm delivered to two steam generators for a loss of main feedwater (LOFW) coupled with a loss of offsite power (LOOP). By letter dated March 28, 1994, the applicant responded to the staff's July 13, 1993, request for additional information relating to this reduced AFW flow and the revised accident analyses involving AFW flow. Further, in FSAR Amendment 82, the applicant stated that the manufacturer's revised design flow rates for the AFW pumps are 450 gpm for the motor-driven pumps and 790 gpm for the turbine-driven pump. These design flow rates include recirculation flow rates of 30 gpm and 50 gpm for the motor- and turbine-driven pumps, respectively.

The applicant stated that the AFW flow rate reduction resulted primarily from the findings of several design reviews. These reviews noted inconsistencies between AFW design parameters established by the applicant and assumptions that were used by Westinghouse in accident analyses. AFW pump testing also prompted a reduction in the flow rates that were assumed for accident analyses purposes. The applicant, therefore, reevaluated the minimum AFW flow requirement for relevant accident analyses at Watts Bar and determined that the revised design flow rates were adequate for all design-basis events. The applicant also modified the pumps to address the capacity problems revealed during testing. The modifications are intended to ensure sufficient operating margin to offset any future deterioration from the effects of wear and aging.

The proposed changes to the minimum required and design flow rates are supported by the reanalysis of design-basis events, including Chapter 15 accident and transient analyses. The staff, therefore, concludes that the revised AFW flow rates are acceptable and that the staff's evaluation and conclusions reports in Section 10.4.9 of the original SER remain valid.

The staff's efforts were tracked by TACs M88694 and M88695.

12 RADIATION PROTECTION

By Amendments 72, 84 and 88, the applicant revised the Final Safety Analysis Report (FSAR) principally to conform with the revised 10 CFR Part 20, "Standards for Protection Against Radiation", which has a mandatory implementation date of January 1, 1994. The staff reviewed these amendments against the requirements of the January 1, 1993, revision to 10 CFR Part 20; the regulatory guides issued since January 1, 1993, that provide guidance for meeting these revised requirements; the criteria in the Standard Review Plan (SRP) Section 12 (NUREG-0800); and the staff's conclusions in the original Watts Bar Safety Evaluation Report (SER, 1982).

FSAR Amendment 88 fully resolves previous staff comments raised by the staff's review of Amendments 72 and 84. Details of the staff's review are delineated in the sections that follow, revising or supplementing the staff's evaluations in the SER or its supplements (SSERs). The staff's comments on Amendment 72 were made publicly available (memo, P. S. Tam to Docket File, January 31, 1994). The staff presented its comments on Amendment 84 to the applicant in a meeting on May 20, 1994 (meeting summary by P. S. Tam, May 26, 1994). The staff's efforts were tracked by TACs M90253 and M90254.

12.2 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

The applicant has revised the discussion of ALARA design and operational considerations in this section to clarify that the total effective dose equivalent (TEDE) for each individual will be maintained ALARA. As revised, FSAR Section 12.1 is consistent with the requirements in 10 CFR 20.1101 and 20.1702 and is, therefore, acceptable to the staff.

12.3 Radiation Sources

In FSAR Amendment 84, the applicant revised the descriptions of the radioactive sources expected to result from normal plant operations, anticipated operational occurrences, and accident conditions. The expected radioactive content of plant components presented in FSAR Tables 12.2-1 through 12.2-22 has been updated. Also, the expected radioactive airborne concentrations presented in FSAR Tables 12.2-19 through 12.2-22 have been revised to indicate the fraction of the derived airborne concentration (DAC) limits listed in 10 CFR Part 20, Appendix B, Table 1, Column 3. The descriptions of plant radioactive sources, as revised, conform to the acceptance criteria in SRP Section 12.2 and are, therefore, acceptable to the staff.

12.4 Radiation Protection Design Features

In Amendment 84, the applicant deleted FSAR Tables 12.3-1 and 12.3-2 which, in light of the requirements in the revised 10 CFR Part 20, contained erroneous information. FSAR Tables 12.3-3, 12.3-4, 12.3-5, and 12.3-6 have been updated to reflect as-built information. FSAR Figures 12.3-1 through 12.3-19 have also been updated to reflect as-built information. These sections, as amended, comply with the acceptance criteria in the SRP and are acceptable to the staff.

In Amendment 88, the applicant revised FSAR Section 12.3.2.2, "Design Description," to specify the radiation dose rate design criteria for the placement and configuration of plant system valves (i.e., local or remote operation). This section as amended is consistent with the staff's conclusion that Watts Bar can be operated within the dose limits of 10 CFR Part 20 and that radiation doses can be maintained ALARA. Therefore, these changes are acceptable to the staff.

12.5 Dose Assessment

In FSAR Amendment 88, the applicant revised the discussion of the estimate of personnel internal exposures to address the new 10 CFR Part 20 requirements concerning the prospective determination for monitoring internal exposures and the consideration for maintaining the TEDE ALARA when using respiratory protective equipment. This section as amended conforms to the staff's guidance in Regulatory Guide 8.7, "Occupational Radiation Exposure Records System," Revision 1, and provides reasonable assurance that the requirements of 10 CFR 20.1502 and 20.1703 will be met.

In Amendment 84, the applicant revised FSAR Tables 12.4-1 and 12.4-2 as well as FSAR Figure 12.4-1 to update the predicted maximum annual doses resulting from plant operation. The discussion of the radiation dose rates at the boundary of the restricted area was also amended to address the new, lower, dose limit for members of the public (100 mrem/year). This section as amended provides reasonable assurance that the radiation doses resulting from plant operations will not exceed the limits in 10 CFR 20.1301.

12.6 Health Physics Program

The applicant revised FSAR Section 12.5 to reflect several programmatic changes that have been made to address the new requirements in the revised 10 CFR Part 20. The discussion of the Respiratory Protection Program has been revised to describe the available equipment, and the considerations integrated into the decision for their use, necessary to ensure that TEDE is maintained ALARA as required by 10 CFR Part 20, Subpart H. The programmatic considerations for the prospective determination of the need to monitor internal and external radiation doses is also provided by these amendments. Dosimeters used for monitoring external radiation doses will be processed by a laboratory accredited under the National Voluntary Laboratory Accreditation Program as required by 10 CFR 20.1501(c). Internal doses will be estimated using DA-hr. tracking and bioassay. A whole-body counter with a radionuclide detection capability consistent with the criteria in American National Standards Institute Standard ANSI-N13.30 is provided. FSAR Amendment 88 specifies that the onsite Radiological Controls Program be conducted in accordance with the guidance in Regulatory Guides (RGs) 8.34*, 8.35*, and 8.36* to ensure that doses from planned special exposures, doses to minors, and doses received by pregnant women are within the limits established by 10 CFR Part 20, Subpart C.

In Amendment 88, the applicant revised FSAR Section 12.5 to describe the controls for access to high-radiation areas that will be used in lieu of those

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*"Monitoring Criteria and Methods To Calculate Occupational Radiation Doses" (RG 8.34), "Planned Special Exposures" (RG 8.35), and "Radiation Doses to the Embryo/Fetus" (RG 8.36).

required in 10 CFR 20.1601(a), in accordance with the provision in 10 CFR 20.1601(c). The posting of and the additional measures for controlling access to very high radiation areas, as required by 10 CFR 20.1902(c) and 20.1602, respectively, are also described. The access controls for high and very high radiation areas described are consistent with the guidance in Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Power Plants," and are acceptable to the staff.

12.7 NUREG-0737 Items

12.7.1 Plant Shielding (II.B.2)

In Amendment 88, the applicant revised the discussion of shielding for accident conditions in FSAR Section 12.3.2.2 to clarify which areas of the plant have been provided shielding to ensure access under accident conditions. All of the applicable vital areas identified in NUREG-0737 Item II.B.2 are discussed. This change does not affect the staff's previous conclusion that Watts Bar conforms to the positions in NUREG-0737 Item II.B.2, and is therefore, acceptable to the staff.

14 INITIAL TEST PROGRAM

In Supplement 12 to the Safety Evaluation Report (SSER 12), the staff reported its evaluation of the preoperational test program, leaving open a number of issues to be tracked by TACs M82644 and M82645. The staff's efforts, reported in the following sections, were tracked by TACs M88937, M88938, M90253, and M90254.

By Final Safety Evaluation Report (FSAR) Amendments 84 and 86, the applicant proposed comprehensive changes to address or resolve these issues. Additionally, in FSAR Section 14.2.7, "Conformance of Test Programs With Regulatory Guides," the applicant has proposed taking specific exceptions to various provisions of, or has proposed to rescind its previous commitments to, various regulatory guides relevant to the Initial Test Program (ITP). Finally, by FSAR Amendment 88, the applicant submitted changes to the FSAR to resolve most of the staff's concerns. The staff's review is reported below.

14.2 Preoperational Tests

Item 1

In Section 14.2.7 of FSAR Amendments 84, 86, and 88, the applicant has proposed taking additional exceptions or alternate approaches to Regulatory Position (RP) C.1 of Regulatory Guide (RG) 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," as described below.

(a) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(a), the applicant takes exception to testing the pressure boundary integrity of the reactor coolant system in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.a.4). The applicant stated that a cold hydrostatic test for the Unit 1 reactor coolant system was performed in support of the original preoperational test program in 1981. The basis for the exception was submitted to the staff in letters dated April 16 and July 2, 1993. In these letters, the applicant stated that instead of performing a second complete reactor coolant system (RCS) cold hydrostatic test (CHT) in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, it will hydrostatically test the piping segments that were not N-stamped and that were modified or repaired since the initial CHT was performed in 1981.

Also, the applicant committed to performing WBN Surveillance Instruction 4.05.0, "Reactor Coolant System Leakage Test," in a revised manner, during hot functional testing (HFT). The results of these tests will be approved by the Joint Test Group (JTG) and retained as plant records in accordance with FSAR Section 14.2.6.

In a letter to the applicant dated July 30, 1993 (tracked by TAC M86347), the staff concluded that the applicant's commitments to perform hydrostatic and leakage tests, as discussed above, were acceptable to resolve the RCS-related concerns. Therefore, the staff finds that these commitments constitute an acceptable alternative for Watts Bar to comply with the pertinent provisions of RG 1.68. This item is closed.

(b) In FSAR Amendment 88, Section 14.2.7, Subparagraph 4.A.(1)(a,1), the applicant has taken exception to performing chemical control system (boration) operability tests in accordance with the guidance in RG 1.68 (Appendix A, Subparagraphs 1.b.2 and 1.n.12). The applicant stated that boron will not actually be introduced to plant systems during preoperational testing, and proposed to simulate boron system operations using demineralized water. Boron will, however, be introduced to the operational system as part of surveillance testing in preparation for the power ascension phase. In addition, the applicant deleted verification of reactor coolant boron concentration adjustment as a test objective and acceptance criteria from FSAR Table 14.2-1, Sheets 18 and 19, "Chemical and Volume Control System Test Summary."

The staff finds the applicant's justification for not verifying proper boron concentration adjustment in the reactor coolant system during preoperational testing unacceptable. The applicant should reinstate its commitment to performing boration in accordance with the guidance in RG 1.68 (Appendix A, Subparagraphs 1.b.2 and 1.n.12), or should submit the necessary technical justification or analysis to demonstrate that simulating boron system operations using demineralized water confirms the ability of the system to batch, store, and transfer boric acid in accordance with the design-basis requirements described in FSAR Section 9.3.8. This item is open and will be tracked by TACs M90253 and M90254.

(c) In FSAR Amendment 88, Section 14.2.7, the applicant has proposed performing component testing, as described in a July 14, 1994, letter, in lieu of performing preoperational testing in accordance with the guidance in the corresponding subparagraphs in Appendix A to RG 1.68 for the systems, or portions of systems, described below:

(1) In Subparagraph 4.A.(1)(c,1), the applicant has taken exception to preoperational testing of containment postaccident heat removal systems related to the non-safety-related portions of the ice condenser system in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.h.3); this issue correlates with Item 10(g) below.

(2) In Subparagraph 4.A.(1)(c,2), the applicant has taken exception to preoperational testing of seismic instrumentation in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.j.10); this issue correlates with Item 10(e) below.

(3) In Subparagraph 4.A.(1)(f,1), the applicant has taken exception to preoperational testing of solid-waste handling systems related to the non-safety-related, solid-waste processing system in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.l.3); this issue correlates with Item 10(c) below.

(4) In Subparagraph 4.A.(1)(h,1), the applicant has taken exception to preoperational testing of communications systems in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.n.13); this issue correlates with Item 10(d) below.

(5) In Subparagraph 4.A.(1)(h,2), the applicant has taken exception to preoperational testing of the heating, cooling, and ventilation systems serving the intake pump station in accordance with the

guidance in RG 1.68 (Appendix A, Subparagraph 1.n.14), based on its non-safety-related nature; this issue correlates with Item 10(h) below.

The staff finds that the proposed testing program elements, controls, and commitments described by the applicant in its July 14, 1994, letter provide an acceptable approach to demonstrate satisfactory operability of the affected systems, or portions thereof, and to confirm the adequacy of their design and performance criteria. Therefore, this item is closed.

(d) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(g), the applicant has taken exception to testing spent fuel pit cooling, including antisiphon devices, high-radiation, and low-water-level alarm tests, in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.m.1). Subsequently, in Amendment 88, Section 14.2.7, Subparagraph 4.A.(1)(g), the applicant stated that because new fuel is currently being stored in the spent fuel pool, only the refueling water purification subsystem will be tested (as described in its July 14, 1994, letter) during the preoperational phase. The applicant added that the balance-of-system (safety-related portions) testing would be conducted during the power ascension test program.

The staff finds that the testing approach proposed by the applicant for both the safety-related and non-safety-related portions of the system complies with the provisions of RG 1.68 and is, therefore, acceptable. This item is closed.

(e) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(h), the applicant takes exception to testing static loads at 125 percent of rated load on three of the four Unit 1 fuel-handling devices (spent fuel pit bridge crane, refueling machine, and 125-ton auxiliary building crane main hook, including both polar crane hooks) in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.m.4). The applicant's justification for this exception is that, except for the auxiliary hook of the 125-ton auxiliary building crane, (1) all the fuel-handling equipment was previously tested at 125 percent of rated capacity and (2) this equipment has not undergone extensive repairs or modifications that would warrant such testing. Also, the applicant had committed to performing the requisite 125-percent-rated capacity test of the 125-ton auxiliary building crane auxiliary hook. In Amendment 88, Section 14.2.7, Subparagraph 4.A.(1)(h), the applicant proposed that cranes not associated with the movement of spent fuel be operationally tested by a combination of acceptance tests and component level testing as described in the applicant's July 14, 1994, letter. The balance of equipment used for handling of spent fuel would be tested under FSAR Table 14.2-1, Sheets 74 and 75, "Fuel Handling Equipment Test Summary"; this issue correlates with Item 10(f) below.

The staff finds that the approach proposed by the applicant gives adequate assurance that the structural integrity of the subject equipment will be verified, and thus complies with the provisions of RG 1.68 (Appendix A, Subparagraph 1.m.4). This item is closed.

(f) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(k), the applicant is taking exception to measuring reactor coolant system (RCS) differential pressure across the fully loaded core and measuring RCS core flow in accordance with the guidance in RG 1.68 (Appendix A, Subparagraphs 2.f and 5.m). The applicant considers that these are prototype tests and that proper measurements of these parameters are shown indirectly through performance of other tests that verify operating temperature and RCS flow.

The staff finds that not measuring an RCS differential pressure across the fully loaded core is acceptable for plants using calculation models and designs identical to prototype plants. This approach conforms to the guidance in RG 1.68 (Appendix A, Subparagraphs 2.f and 5.m) and the acceptance criteria of SRP Section 14.2. Therefore, this item is closed.

(g) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(m), the applicant is taking exception to demonstrating the operability of residual or decay heat removal systems, including atmospheric steam dump valves and turbine bypass valves, in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 4.q), and to demonstrating that process and effluent radiation monitoring systems are responding correctly by performing independent laboratory or other analyses in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 5.z). Subsequently, in FSAR Amendment 88, Section 14.2.7, Subparagraph 4.A.(1)(m), the applicant removed its exception to the provisions of RG 1.68 (Appendix A, Subparagraph 4.q), and confirmed, instead, its exception to having to demonstrate that laboratory analyses of samples from the process and/or effluent systems verify responses of installed process and effluent radiation monitors in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 4.g). The applicant states that the proper response of process and effluent radiation monitors is demonstrated by the plant calibration program and during preoperational testing for the process and effluent radiation monitoring system as described in FSAR Table 14.2-1, Sheet 31, "Process and Effluent Radiation Monitoring System Test Summary."

The staff finds the proposed approach is in agreement with the guidance of RG 1.68, for Appendix A, Subparagraph 5.z, and the acceptance criteria of SRP Section 14.2 and is, therefore, acceptable. This item is closed.

(h) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(q), the applicant takes exception to demonstrating that core thermal and nuclear parameters are in accordance with predictions with a single, high-worth rod fully inserted, during return and following return of the rod to its bank position, in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 5.f). The applicant states that no appreciable new data would be obtained from performing this test, as previous plants have proved the design bases for typical cores.

The staff finds the proposed approach is in agreement with the guidance and acceptance criteria of SRP Section 14.2 and is, therefore, acceptable. This item is closed.

(i) In FSAR Amendment 84, Section 14.2.7, Subparagraph 4.A.(1)(i), the applicant is taking exception to performing preoperational or acceptance testing of heat tracing and freeze protection systems in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 1.n.(18)). The applicant's basis for this exception is that such systems only warrant component-level testing because of their non-safety-related simple functions.

The staff finds that component-level testing will appropriately demonstrate the operability of these non-safety-related systems and this exception is, therefore, acceptable. This item is closed.

Item 2

In FSAR Amendment 84, Section 14.2.7, Subparagraphs 4.A.(2) through 4.A.(5), the applicant has stated that certain provisions of RG 1.68, Revision 2 regarding the power-ascension phase (1) do not apply to the design of Watts Bar, (2) are not specifically tested or reviewed in power ascension, (3) are satisfied in the preoperational phase, or (4) are satisfied at plant conditions other than those specified.

The staff recognizes that certain specific provisions of RG 1.68 are only applicable to boiling-water reactor designs. Furthermore, the staff acknowledges that the information in the subject paragraphs has been included by the applicant for purposes of clarification only and, therefore, is not to be used to effect changes or amendments to the conduct or performance of ITP testing at Watts Bar, in a manner not previously described in Amendments 84, 86, and 88 to FSAR Section 14.2.7, Subparagraph 4.A.(1). This item is closed.

Item 3

In SSER 12, the staff requested that additional information be provided in FSAR Section 14.2.7, "Conformance of Test Programs With Regulatory Guides," to clarify the applicant's compliance with RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems" (April 1982).

(a) In FSAR Amendment 74 to Sections 9.3.1.4 and 14.2.7.7, the applicant described changes to address its commitments to RG 1.68.3 and noted an exception to Regulatory Position (RP) C.8 for testing a sudden loss of instrument air pressure. The applicant referenced correspondence from R.C. Lewis (NRC) to H.G. Parris (TVA), dated February 28, 1984, as providing NRC concurrence for this exception.

    In SSER 12, the staff noted that this response appeared to be in conflict with the applicant's correspondence dated July 12, 1990, in response to NRC Generic Letter (GL) 88-14. Enclosure 2 of the applicant's correspondence indicated that the preoperational test scoping document has been revised to require testing the safety-related valves supplied by the auxiliary control air system for both rapid and gradual loss of air in accordance with RG 1.80, "Preoperational Testing of Instrument Air Systems." In Attachments 1 and 2 to TVA letter dated February 28, 1994, the applicant stated that Amendment 84 to FSAR Chapter 14 would not include clarifications on commitments in this area as further evaluation was necessary. In Amendment 88 to FSAR Section 14.2.7.7, and as confirmed in TVA letter dated August 23, 1994, the applicant reasserted its intent of complying with RG 1.68.3, with the following exception to Regulatory Position (RP) C.8:

    (1) Auxiliary control air system loads will be tested on an individual basis to verify their response to a sudden loss of system pressure.

The applicant claims that the staff's concurrence with this exception is reflected in the NRC letter dated February 28, 1984.

(2) Control air system loads (safety-related only) will be tested to verify their response to a loss of system pressure. The test will be performed on an individual load basis. Non-safety-related air-operated loads will be tested on a component basis to verify proper response to a loss of air pressure.

In the February 28, 1984, letter, the staff had concluded that performing the sudden-loss-of-air test as described in RP C.8 of RG 1.80, "Preoperational Testing of Instrument Air Systems," was not practicable due to limitations inherent in the Watts Bar system design and, therefore, the applicant's exception to the RP was accepted. Subsequently, in Amendment 74 to FSAR Chapter 14, the applicant rescinded its commitment to RG 1.80 and committed to performing the requisite preoperational testing of instrument air systems in accordance with the guidance in RG 1.68.3 while retaining its exception to performing the sudden-loss-of-air testing.

The staff finds that the performance of (i) the gradual-loss-of-air test in accordance with the guidance in RP C.8 of RG 1.68.3, and as described in Table 14.2-1, Sheets 86 and 87, "Compressed Air System Test Summary," and (ii) the sudden-loss-of-air test, as proposed by the applicant on the auxiliary control air system and control air system loads, constitute an acceptable approach to meet the objectives of the initial test program guidance in RG 1.68, and is in conformance with the SRP, Section 14.2. Therefore, this item is closed.

(b) In Amendment 74 to FSAR Section 14.2.7.7, the applicant had taken an exception to RP C.11 to not demonstrate operability of compressed-air system loads under increased pressure conditions as the applicant considered that this system had adequate safety features, as described in FSAR Section 9.3.1.3, to prevent such occurrences. The applicant stated that the safety evaluation indicated that the design is adequate to prevent system overpressure as described in FSAR Section 9.3.1.3. Additionally, the applicant stated that the maximum pressure rating of the most limiting component of the system piping, valves, and equipment (up to the end-user pressure regulator) was determined to be at least 20 psi higher than the system safety valve setpoint plus 10-percent accumulation, thus ensuring that there is adequate protection against overpressurization.

The staff finds this response and the change incorporated in Amendment 84 to FSAR Chapter 14 acceptable. This item is closed.

<u>Item 4</u>

In Amendment 74 to FSAR Section 14.2.7.9, the applicant documented its commitment to the provisions of RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1 (August 1977). Subsequently, in Amendment 84 to FSAR Section 14.2.7.9, the applicant rescinded its commitments to RG 1.108 and committed to perform preoperational testing of onsite diesel generators, designed to provide emergency power to emergency safety features under loss-of-offsite-power conditions, in accordance with the provisions of RG 1.9, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants", Revision 3 (July 1993).

The staff finds this commitment to be in agreement with the guidance and acceptance criteria of the applicable sections of the SRP and, therefore, acceptable. This item is closed.

Item 5

The staff had requested that the applicant include an exception for not using RG 1.139, "Guidance for Residual Heat Removal," issued for comment May 1978, in the development or conduct of the Watts Bar Initial Test Program. The applicant based this exception to this RG on the Safety Evaluation Report (NUREG-0847) and subsequent acceptance of the applicant's assessment of applicability of the Diablo Canyon natural circulation test to Watts Bar by the NRC in Section 5.4.3 of SSER 10.

The applicant included an exception to RG 1.139 in Amendment 84 to FSAR Chapter 14 incorporating the justification noted above. Therefore, this item is closed.

Item 6

The staff had requested that the applicant provide the technical basis for deleting the failed fuel detector (FFD) system and the technical means that would be utilized to monitor fuel cladding integrity as recommended by RG 1.68 (Appendix A, Subparagraph 1.j.(12)). The applicant responded that the gross failed fuel detection system (GFFDS) is installed at Watts Bar but that, as discussed in FSAR Section 9.3.5, no credit is taken for its use in identifying conditions of fuel failure. The applicant added that the GFFDS performs no safety-related function and is not designed to satisfy any specific safety criteria. The applicant also indicated that fuel cladding damage is prevented by the sampling requirements and operational limits in the Technical Specifications and that fuel cladding integrity is ensured through monitoring performed by subcooling margin, incore thermocouples, and reactor vessel level instrumentation. In addition, in Amendment 88 to FSAR Section 14.2.7.4.A(d), the applicant has also taken exception to verifying proper operation of the failed fuel (FF) detection system (25% FF, 100% FF) in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 5.q).

The staff concurs with the applicant's assessment of the functional and design criteria requirements of the FFD system at Watts Bar. The staff confirmed that the applicant has no commitments on the docket to include a system that satisfies the functions of an FFD system as described in RG 1.68 (Appendix A, Subparagraph 1.j.(12)) or that would need to have its proper operation verified in accordance with the guidance in RG 1.68 (Appendix A, Subparagraph 5.q). This item is closed.

Item 7

The staff had asked the applicant to modify the justification in FSAR Section 14.2.7 for the exception to power coefficient testing at 100-percent power to include performance of core reactivity balance testing at low power levels.

Additionally, the applicant was asked to provide or reference the appropriate test abstract where core reactivity balance testing is performed at low power levels. The applicant responded that overall core reactivity will be measured during low-power physics testing and at approximately 100-percent power to demonstrate the adequacy of core design reactivity coefficients. The appropriate test abstract for core reactivity balance testing at low power levels was included in FSAR Table 14.2-2, Sheet 26, "Integrated Engineered Safety Features Actuation System Test Summary."

The staff finds this response and the changes incorporated in Amendment 84 to FSAR Chapter 14 acceptable. This item is closed.

Item 8

In SSER 12, the staff had requested that the Power Ascension Phase test summaries or FSAR Section 14.2.3.4 or both be modified to describe the initial system conditions, including configuration, components that should or should not be operating, and other pertinent conditions that might affect the operation of the system, and to specify the bases for determining acceptable system and component performance. In Amendment 84 to FSAR Chapter 14, the applicant's response refers to the inclusion of the appropriate Power Ascension Phase test summaries in Table 14.2-2.

The staff reviewed the modified Power Ascension Phase test program abstracts to assess the technical adequacy of the acceptance criteria in demonstrating the important-to-safety functional requirements of the system, and to verify that such acceptance criteria are traceable to the appropriate source documents. The staff found that the modified Power Ascension Phase test program individual abstracts incorporated in Amendment 84 to FSAR Chapter 14, Table 14.2-2, provide adequate bases for determining acceptable system parameters and performance characteristics, except for the following, for which the applicant determined that performance acceptance criteria were not required:

(a) "Rod Control System Test Summary," Sheet 10 of 37

(b) "Incore Movable Detectors Test Summary," Sheet 12 of 37

(c) "Calibration of Steam and Feedwater Flow Instrumentation at Power Test Summary," Sheet 21 of 37

Subsequently, in Amendment 88 to FSAR Chapter 14, Table 14.2-2, the applicant revised these test summaries to incorporate the applicable performance acceptance criteria. This item is closed.

Item 9

In Amendment 74 to FSAR Section 14.2.7, the applicant took exception to RG 1.68, Revision 2 (Appendix A, Subparagraph 1.j.(22)), which involves instrumentation that can be used to monitor plant parameters during the course of postulated accidents. On the basis of the staff's review, documented in SSER 12, two instrument tests appeared to be incomplete. The staff requested that the applicant incorporate proper control room indication and alarm function tests and acceptance criteria for the containment pressure instrumentation, and water level instrumentation, into the appropriate preoperational test abstract(s). The staff's request was based on the following:

(a) The functional parameters and performance characteristics of the containment wide-range pressure indicators system, as described in FSAR Sections 6.2.4, 7.3.1, 7.3.2, and FSAR Table 7.5-1, were found to be encompassed within the preoperational test abstracts described in Sheet 59, "Reactor Pressure Boundary Leakage Detection System Test Summary," and Sheet 83, "Containment Isolation System Test Summary." The test method described in Sheet 59 verified proper calibration of the instrumentation and annunciation of containment pressure. However, the referenced acceptance criterion (FSAR Section 5.2.7) did not describe containment pressure detection or indication, and the test method described in Sheet 83 did not address functionality of control room instrumentation.

(b) The functional parameters and performance characteristics of the containment water level monitors system, as described in FSAR Sections 6.3.2, 6.3.4, and FSAR Table 7.5-1, were found to be encompassed within the preoperational test abstracts described in Sheet 22, "Safety Injection System Test Summary," and Sheet 83 "Containment Isolation System Test Summary." However, the test methods described in Sheets 22 and 83 did not specifically reference the containment water level monitoring system.

In FSAR Amendment 84, the applicant stated that the requisite testing requirements had been included in Table 14.2-1, Sheets 23, 59, 82 and 83. The staff confirmed that these test abstracts contained the appropriate test requirements; however, the staff noted that this response appeared to be in conflict with the exception continued to be taken by the applicant in Section 14.2.7, Subparagraph 4.A.(1)(e). In particular, the exception for instrumentation within the scope of RG 1.68 (Appendix A, Subparagraph 1.j.(22)), "additional testing in the form of a preoperational test is not warranted."

Subsequently, in Amendment 88 to FSAR Chapter 14, the applicant revised Section 14.2.7, Subparagraph 4.A.(1)(e) to delete the reference to the previous exception to RG 1.68 (Appendix A, Subparagraph 1.j.(22)). Therefore, this item is closed.

Item 10

In its letters of July 14 and August 19, 1994, the applicant proposed removing, from the description of the preoperational test program at Watts Bar, certain test abstracts or portions of test abstracts previously included in FSAR Table 14.2-1 by Amendment 74. As an alternative to the preoperational testing provisions of RG 1.68, the applicant proposed performing a combination of acceptance test instructions (ATIs), component tests (CTs), and special performance tests (SPTs) on the systems within the scope of these test abstracts, or on portions thereof, as described in the revised or modified test summaries in the August 19, 1994, letter.

The affected test abstracts or summaries follow:

(a) Secondary Process Sampling Test Summary (previously "Process Sampling System Test Summary," Table 14.2-1, Sheets 9 and 10, in Amendment 74 to FSAR Chapter 14).

(b) Refueling Water Purification Test Summary (previously a portion of "Spent Fuel Cooling System Test Summary," Table 14.2-1, Sheets 15 and 16, in Amendment 74 to FSAR Chapter 14). As proposed by the applicant, testing of the balance of the spent-fuel cooling system is to be addressed under the power ascension phase of Amendment 88 to FSAR Chapter 14, Table 14.2-2, Sheet 11, "Spent Fuel Pool Cooling System Test Summary."

(c) Solid Waste Processing Test Summary (previously "Solid Waste Processing System Test Summary," Table 14.2-1, Sheet 29, in Amendment 74 to FSAR Chapter 14). The applicant notes that FSAR Section 11.5 states that this system no longer processes dry active waste.

(d) Communications System Test Summary (previously "Communications System Test Summary," Table 14.2-1, Sheet 56, in Amendment 74 to FSAR Chapter 14).

(e) Seismic Instrumentation Test Summary (previously "Seismic Instrumentation Test Summary," Table 14.2-1, Sheet 56, in Amendment 74 to FSAR Chapter 14).

(f) Fuel Handling and Vessel Servicing Equipment Test Summary (previously "Fuel Handling and Vessel Servicing Equipment Test Summary," Table 14.2-1, Sheets 74 and 75, in Amendment 74 to FSAR Chapter 14). The applicant notes that equipment used for handling of spent fuel is tested under "Fuel Handling Equipment Test Summary," Table 14.2-1, Sheets 74 and 75, in Amendment 88 to FSAR Chapter 14.

(g) Ice Condenser Test Summary (previously "Ice Condenser System Test Summary," Table 14.2-1, Sheet 88, in Amendment 74 to FSAR Chapter 14). The applicant notes that the ice condenser is a safety-related system that has some non-safety-related features. These non-safety-related features are to be tested as described in the applicant's August 19, 1994, letter. The safety-related portions of the system will be tested under "Ice Condenser System Test Summary," Table 14.2-1, Sheet 88, in Amendment 88 to FSAR Chapter 14.

(h) Intake Pump Station Ventilation Test Summary (previously "Intake Pump Station Ventilation System Test Summary," Table 14.2-1, Sheet 90, in Amendment 74 to FSAR Chapter 14).

The staff finds that the proposed testing program elements, controls, and commitments described by the applicant in the August 19, 1994, letter, provide an acceptable approach to demonstrate satisfactory operability of the affected systems, or portions thereof, and to confirm the adequacy of their design and performance criteria. Therefore, this item is closed.

Item 11

In Amendment 88 to FSAR Chapter 14, Table 14.2-1, "Preoperational Test Summaries," Sheet 48 of 90, "AC Power Distribution System Test Summary," the applicant deleted the requirement to verify, under "Test Method," the capability of each common station service transformer (CSST) to carry the load required to supply engineered safety feature (ESF) loads of one unit under loss-of-coolant-accident (LOCA) conditions in addition to power required for shutdown of the non-accident unit in accordance with the guidance in RG 1.68 of Appendix A, Subparagraph 1.g.(1). This requirement is related to the design bases of Watts Bar Units 1 and 2.

Although the applicant has not formally withdrawn its license application for Unit 2, the applicant is presently concentrating all its efforts toward obtaining the operating license (OL) for Unit 1 only. Therefore, under the current scenario, it would be sufficient for the applicant to demonstrate the capability of each CSST to carry the load required to supply ESF loads of one unit (Unit 1) under LOCA conditions to comply with the provisions of RG 1.68. Before an OL can be issued for Unit 2, however, the applicant would have to demonstrate the capability of each CSST to carry the load required to supply ESF loads of one unit under LOCA conditions in addition to power required for shutting down of the non-accident unit.

Therefore, the applicant should reinstate the deleted test objective and demonstrate the CSST capability related to both units, or alternatively, should commit to performing the requisite testing, subject to the conditions described above, for Unit 1 only. This item is open and will be tracked by TACs M90253 and M90254.

Item 12

In Amendment 88 to FSAR Chapter 14, the applicant modified the test abstract for the "Anticipated Transient Without Scram Mitigation System Actuation Circuitry [AMSAC] Test Summary," Table 14.2-1, Sheet 85 of 90, to exclude specific design and related system logic text from Step 2 of its "Acceptance Criteria." Specifically, requirements to verify that with AMSAC armed at greater than or equal to (1) 40-percent simulated power, the steam generator low-low-level logic setpoint is greater than 12 percent of narrow range level and less than the RPS low-low-level trip setpoint, and (2) 80-percent simulated power, the steam generator low-low-level logic setpoint is at least 25 percent of narrow gauge level and less than the RPS low-low-level trip setpoint, were deleted.

In Amendment 72, the applicant modified FSAR Chapter 7 to implement design changes in the reactor protection system (RPS) at Watts Bar to upgrade certain portions with microprocessor-based technology in order to improve the reliability and accuracy of process data signals, to simplify calibration testing and maintenance, and to accommodate future additions to improve control and monitoring instrumentation as new technology is developed. In SSER 13, the staff accepted the proposed design changes. Subsequently, the implemented design changes permitted the elimination of certain design features or capabilities (described above) related to the AMSAC system.

The staff concurs with the applicant's assertion that the text in FSAR Table 14.2-1, Sheet 85 of 90, was deleted to reflect current system design and such revision is, therefore, acceptable. This item is closed.

Item 13

In Amendment 88 to FSAR Chapter 14, Table 14.2-1, "Preoperational Test Summaries," Sheet 11 of 90, "Post Accident Sampling System Test Summary," the applicant deleted the requirement to confirm, under "Acceptance Criteria," the capability of safely transporting all samples for onsite analysis, or to a transfer point for offsite analysis, and have them analyzed within the required timespan as described in FSAR Section 9.3.2.6.

Following the accident at Three Mile Island, Unit 2, the NRC required licensees to have the capability of promptly obtaining (within 3 hours or less from the time a decision is made to obtain the sample) a sample of the reactor coolant and containment atmosphere sampling line systems under accident conditions without incurring a radiation exposure to any individual in excess of 3 rem to the whole body or 18-3/4 rem to the extremities. This requirement was designated as Item II.B.3, "Postaccident Sampling Capability," in NUREG-0737, "Clarification of TMI Action Plan Requirements." In Amendment 87 to FSAR Section 9.3.2.6.1, the applicant states that the postaccident sampling subsystem (PASS) is designed to meet the intent of and provide for acquiring, analyzing, and disposing of samples, as described in Section II.B.3 of NUREG-0737, and to keep personnel exposures within the limits of General Design Criterion (GDC) 19.

The staff finds the applicant's proposal of not having to demonstrate this capability during preoperational testing unacceptable. The applicant should reinstate the text deleted from Table 14.2-1, Sheet 11 of 90, and perform the requisite testing, or should provide clarification on how the applicant intends to demonstrate that requirements in Section II.B.3 of NUREG-0737 are satisfied. This item is open and will be tracked by TACs M90253 and M90254.

14.2.3 Conclusion

The staff performed its review on the basis of the information provided by the applicant in the FSAR as updated by Amendments 84, 86, and 88, and the applicants's letters dated February 28, April 2, July 14, July 20, and August 19, 1994. The staff's safety evaluation in this supplement discusses in detail (1) the items that are open, pending receipt and review of the applicant's responses, and (2) the bases for the resolution of issues that had been previously identified in SSER 12. For areas not discussed above, the staff finds the Watts Bar Units 1 and 2 Initial Test Program description contained in FSAR Chapter 14, as updated through Amendment 88, to be generally comprehensive and to encompass the major phases of the testing program requirements prescribed by various guidance documents. Unresolved issues will continue to be tracked by TACs M90253 and M90254.

15 ACCIDENT ANALYSIS

15.1 General Discussion

In Safety Evaluation Report (SER) Supplement 13 (SSER 13), Chapter 7, the staff approved the applicant's replacement of the original analogue Foxboro processor control system with Westinghouse Eagle-21 digital process protection equipment. Westinghouse discussed the effect of the replacement on the Chapter 15 accident analyses for the upgraded protection features presented below and in Westinghouse report WCAP-13462, Revision 1, "Summary Report - Process Protection System Eagle-21 Upgrade, NSLB, MSS and TTD Implementation for Watts Bar Units 1 & 2," June 1993 (transmitted by TVA letter dated May 23, 1994). The digital electronics of the Eagle-21 system do not affect the non-LOCA (loss-of-coolant accident) safety analyses done without the Eagle-21 upgrade, because the time delays and inaccuracies with the Eagle-21 system are not greater than those previously assumed for the analogue system. The upgrades, however, affect the protection system modeling which was used in some of the original analyses of the licensing-basis non-LOCA transients. In Table 2.1 of WCAP-13462, Revision 1, Westinghouse lists the affected non-LOCA accidents. Two of the accidents, loss of normal feedwater (see Section 15.3.3 below) and main feedwater pipe rupture (see Section 15.3.2 below), whose analyses were affected by the upgrade, were reanalyzed.

15.2 Normal Operation and Anticipated Transients

15.2.1 Loss-of-Cooling Transients

The trip time delay (TTD) functional upgrade was incorporated as part of the Eagle-21 process protection system for low-low level steam generator reactor trip. The staff approved this upgrade in Section 7.2.1.1 of SSER 13. See Sections 15.3.2 and 15.3.3 (below) for the staff's combined evaluation of all transients affected by the TTD.

The staff's effort was tracked by TAC M81063.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.4 Inadvertent Boron Dilution

In FSAR Amendment 80, the applicant used a reactivity insertion rate of 0.6 pcm (% millirho)/sec in the most recent analysis of the accident associated with uncontrolled boron dilution during full-power operation with reactor control (i.e., control rods) in manual. By letter dated June 30, 1994, the applicant stated that this insertion rate replaced the previous value of 2.0 pcm/sec because it results in a lower (more limiting) value for the departure from nucleate boiling ratio (DNBR). The staff accepts the applicant's reanalysis.

This review was tracked by TACs M88644 and M88645.

15.2.4.6 Rod Cluster Control Assembly Ejection

As stated in the original Safety Evaluation Report (SER, 1982), the applicant's original acceptance criteria for the rod ejection accident for gross damage of fuel were a maximum cladding temperature of 2700 °F and 200 calories per gram in the hottest pellet. In FSAR Amendment 80, the maximum temperature was changed to 3000 °F. The change is acceptable because the NRC acceptance criterion of 280 calories per gram (Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"), as stated in the SER, continues to be met and there is no specific NRC criterion on cladding temperature for the rod ejection accident.

The staff's efforts were tracked by TACs M88644 and M88645.

15.3 Limiting Accidents

15.3.2/15.3.3 Steamline Break/Feedwater System Pipe Break

The elimination of the low-flow feedwater reactor trip via the median signal selector (MSS) is part of the Eagle-21 upgrade. The staff found this upgrade acceptable in SSER 13, Section 7.2.1.1. Elimination of the low-feedwater reactor trip does not require a reanalysis of the non-LOCA safety analyses because this trip was never assumed to be the primary functioning reactor protection.

The applicant reanalyzed the safety analyses for transients affected by the new steamline protection feature (see SSER 13, Section 7.2.1.1 for design of this feature). In addition, the new steamline-break protection logic was modeled in a reanalysis of two ruptures: the main feedline and a steamline break outside containment (WCAP-11053, Proprietary Class 2, "Steamline Break, Outside Containment Mass and Energy Release Analysis, Watts Bar," March 29, 1985 and WCAP-13462, Revision 1). In SSER 13, Section 7.2.1.1, the staff found the new steamline protection feature for the Watts Bar Unit 1 acceptable. The staff reviewed the applicant's reanalyses and found that the conclusions in the SER remain valid.

The trip time delay (TTD) functional upgrade was incorporated as part of the Eagle-21 process protection system for low-low level steam generator reactor trip. The staff had approved this functional upgrade in SSER 13, Section 7.2.1.1. The applicant used the approved methodology of WCAP-11325-P-A, Rev. 1 ("Steam Generator Low Water Level Protection System Modifications To Reduce Feedwater-Related Trips," February 1988) to perform Watts Bar-specific analysis of the loss of normal feedwater transient. This analysis serves as the basis for limits on (1) 1/*N* logic time delays calculated in the part-power loss of normal feedwater analyses and (2) 2/*N* logic time delays that were lower than the 2/*N* logic time delays calculated in the part-power loss-of-normal-feedwater analyses. This analysis was used to establish limits for the steam generator, low-low water level, signal delay times and trip setpoints. For all the cases, the auxiliary feedwater heat removal capability was determined to be sufficient to remove the decay heat so that the pressurizer does not fill. This ensures that all applicable Condition II safety analysis acceptance criteria are met. Therefore, the staff finds the TTD upgrade to be acceptable.

A number of other events that credit the steam generator low-low water level trip were reanalyzed to ensure that with the time delays calculated above, the current licensing-basis events presented in the FSAR remain as the limiting transients. These transients were reanalyzed as documented in WCAP-13462, Revision 1, as part of the Watts Bar Reduced Auxiliary Feedwater Program. The following transients were reanalyzed:

(1) full-power loss of normal feedwater
(2) full-power major rupture of a main feedwater pipe
(3) steamline break outside containment

The applicant found that implementation of the TTD for Watts Bar introduces no time delays at indicated power levels greater than 50 percent and the TTD feature is consistent with the Eagle-21 system design. Therefore, in its review of the three analyses in WCAP-13462, Rev. 1, the staff concludes that implementation of the TTD is acceptable, and that there is no need to revise the staff's findings and conclusions that appeared in the SER.

The staff's efforts were tracked by TAC M81063.

15.3.4/15.3.5 Reactor Coolant Pump Rotor Seizure/Reactor Pump Shaft Break

By FSAR Amendment 80, the applicant combined these two sections because the accidents were of a similar nature. The revised analysis uses LOFTRAN rather than PHOENIX to calculate the loop flow transients. The use of LOFTRAN for this purpose is acceptable, as originally stated in the SER. The new maximum cladding temperature is 1795 °F and the maximum reactor coolant pressure is 2642 psia, as originally communicated to the staff as part of TVA's submittal on VANTAGE 5H fuel on August 24, 1992. The staff published its review of VANTAGE 5H in SSER 13, finding the use of VANTAGE 5H fuel design, including the new analysis results transmitted by the August 24, 1992 letter, acceptable. The applicant's new analysis does not change the staff's original conclusion in the SER.

The staff's efforts were tracked by TACs M88644 and M88645.

15.4 <u>Radiological Consequences of Accidents</u>

15.4.3 Steam Generator Tube Rupture

In SSER 12, the staff completed most of its review on this subject, with operator action time left as the only open issue. Furthermore, by letter dated April 13, 1994, the applicant submitted revised information as Revision 1 of topical report WCAP-13575, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for Watts Bar Nuclear Plant Units 1/2."

The applicant revised the steam generator tube rupture (SGTR) analysis to include a conservative bounding value of 10 percent for blowdown through the main steam safety valves (MSSVs) prior to reseating during the SGTR accident. The SGTR analysis methodology that is described in Revision 1 of WCAP-13575 is unchanged from the original analysis. The revised analysis indicates that the steam generator with the ruptured tube will not overfill for an SGTR for Watts Bar Units 1 and 2, even with a 10-percent MSSV blowdown. The conclusions in the staff's original safety evaluation are still valid.

By the April 21, 1994, letter, the applicant also submitted the results from simulator runs to address operator response times during an SGTR. The purpose of the simulator runs is to demonstrate the capability of plant operators to respond to the most limiting SGTR accident scenario, a failed open level control valve resulting in steam generator overfill, within the time limits assumed in the accident analysis. This demonstration should take place on the plant-specific simulator and may be included as part of the operator training program.

By letter dated June 28, 1994, the staff informed TVA that it would consider TVA to have acceptably demonstrated the capability to respond to the most limiting SGTR accident when a minimum of 80 percent of the licensed operators acceptably mitigated the event during the simulator runs within the operator response times assumed in the SGTR accident analysis. The staff noted that Operations Crews 1, 2, and 3 would be considered by the staff to represent 80 percent of the applicant's projected operating staff. By letter dated April 21, 1994, the applicant successfully demonstrated the SGTR operator response times for Crews 1 and 2. By letter dated August 15, 1994, the applicant submitted the SGTR operator response times for Crew 3.

The staff's evaluation (letter from C.E. Rossi (NRC) to A.E. Ladieu (Westinghouse), WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," March 30, 1987) of the Westinghouse Owner's Group report WCAP-10698 stipulates plant-specific criteria for assessing operator response times in the event of an SGTR. The following criteria were employed to evaluate the information provided by the applicant regarding operator response times during an SGTR at Watts Bar:

- Criterion 1 - Provide simulator and emergency operating procedure training related to a potential SGTR.

  The applicant stated in a letter dated April 13, 1993, that onsite simulator and emergency operating procedure (EOP) training relevant to an SGTR are provided. The staff finds that the applicant has satisfied Criterion 1.

- Criterion 2 - Utilizing typical control room staff as participants in demonstration runs, show that the operator response times in the SGTR accident analysis are realistic and achievable by the plant operators.

  By the April 21 and August 15, 1994, letters, the applicant provided the assumed and demonstrated operator response times for the most limiting SGTR accident scenario. The information representing at least 80 percent of Watts Bar operators indicates that the demonstrated times are bounded by the assumed times for the SGTR accident analysis. On the basis of this information, the staff finds that the applicant has satisfied Criterion 2.

- Criterion 3 - Complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention, using design-basis assumptions regarding available equipment and its impact on operator response times.

  The data submitted by the applicant show that the operator response times successfully demonstrate the response to the limiting SGTR accident scenario. The demonstrated times are bounded by the assumed times in SGTR accident analysis. On the basis of this information, the staff finds that the licensee has satisfied Criterion 3.

- Criterion 4 - If the EOPs specify SG [steam generator] sampling as a means of identifying the SG with the ruptured tube, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

  By the April 13, 1993 letter, the applicant indicated that EOPs require sampling of the secondary coolant for laboratory analysis as one of the steps in the process of evaluating a potential SGTR event. The applicant estimated that analyzing the secondary coolant would take about 20 to 30 minutes. The applicant noted that the analysis is only used to confirm that an SGTR event has occurred and is not used as a basis for initiating a response to the SGTR event. The applicant explained that the Watts Bar EOPs are written so that SGTR mitigating actions will not be delayed while awaiting the results of the laboratory sample. On the basis of this information, the staff finds that Criterion 4 is satisfied.

The staff has reviewed the applicant's submittals regarding operator response times during an SGTR, and concludes that the applicant has satisfactorily verified the times assumed in the SGTR analysis for the Watts Bar Nuclear Power Plant.

The preceding review fully resolves the staff's concerns on this subject, and proposed License Condition 41 is deleted.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

In SSER 4, the staff responded to the ACRS letter of August 16, 1982, which, among other things, addressed the mortar-lined emergency raw cooling water (ERCW) piping. By a letter dated June 15, 1984, TVA committed to (1) leave a mortar-lined pipe sample in the river at TVA's Singleton Materials Engineering Laboratory, (2) perform an atomic absorption test every 5 years on a sample of the mortar lining to quantitatively determine the calcium ion content, and (3) take steps to investigate the condition of the lining should the tests indicate a loss of 40 percent or greater calcium ion loss. The calcium carbonate content was then determined to be approximately 30 parts per million (ppm), and the calcium ion concentration was approximately 20 ppm.

By a letter dated June 9, 1994, TVA informed the staff that since TVA has sold Singleton Materials Engineering Laboratory, TVA can no longer leave samples there. However, TVA proposed to move the samples to the cooling tower basin, where the character of the water is much more representative of that inside the ERCW piping. The staff agrees with TVA's assessment that the water in the cooling tower basin is more representative than the water at Singleton, which is some distance from Watts Bar. Therefore, the relocation of the sample does not change the substance of the three commitments. The staff's original evaluation is unaffected.

This evaluation was tracked by TAC M89762.

APPENDIX A

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

NRC Letters and Summaries

February 28, 1984	Letter, R. C. Lewis to H. G. Parris (TVA), regarding "Preoperational Testing of Watts Bar Instrument Air Systems, Docket No. 50-390."
July 13, 1993	Letter, P. S. Tam to M. O. Medford (TVA), regarding auxiliary feedwater flow.
July 30, 1993	Letter P. S. Tam to M. O. Medford (TVA), regarding "Watts Bar Unit 1 Hydrostatic Test Commitment Clarification (TAC M86347)."
January 31, 1994	Memorandum, P. S. Tam to Docket File, commenting on FSAR Amendment 72.
February 2, 1994	Letter, P. S. Tam to M. O. Medford (TVA), informing of preliminary review results on use of Thermo-Lag fire-retardant material.
February 2, 1994	Letter, F. J. Hebdon to M. O. Medford (TVA), discussing schedule to develop the Unit 1 Technical Specifications.
February 14, 1994	Letter, T. E. Murley to A. Harris (TVA), informing of NRC policy regarding participation in meetings by members of the public.
February 14, 1994	Letter, P. S. Tam to M. O. Medford (TVA), transmitting review results of Revision 3 of the Cable Issues Corrective Action Program.
February 14, 1994	Letter, F. J. Hebdon to M. O. Medford (TVA), informing of completion of the staff's review of TVA's response to Generic Letter 89-10, Supplement 5, "Inaccuracy of Motor-Operated Valve Diagnostic Equipment."
February 25, 1994	Letter, S. A. Varga to I. P. Dickinson (TVA), stating that the NRC's concern regards safety and environment aspects of Watts Bar, not the financial aspect.

March 1, 1994	Letter, P. S. Tam to M. O. Medford (TVA), transmitting final safety evaluation on use of U-bolts as pipe clamps.
March 4, 1994	Letter, P. S. Tam to M. O. Medford (TVA), accepting proposed date for TVA to respond to Generic Letter 88-20, Supplement 4, regarding individual plant examination of external events.
March 9, 1994	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information to update the staff's Final Environmental Statement.
March 17, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), accepting Revision 3 of the Q-list Corrective Action Program.
March 25, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information regarding TVA's changed commitment on Generic Letter 89-13, "Service Water Systems."
March 28, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information regarding FSAR Chapter 8.
April 5, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting review results of Revision 5 of the Replacement Items Corrective Action Program.
April 6, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on the fire-protection program.
April 6, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on FSAR Chapter 11.
April 21, 1994	Letter, W. Hodges to W. Rasin (NUMARC), transmitting safety evaluation of WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors."
April 25, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of completion of Revision 6 of QA Record Corrective Action Program.
April 25, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), informing of recent appeals court decision that may have some effect on the applicant's future request to extend the construction permit.
May 2, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting copies of SSER 13.
May 3, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on FSAR Chapter 3, as revised by Amendment 79.

May 3, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting results of review of the Fourth and Fifth Annual Reports of Employee Concerns Special Program.
May 5, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of completion of Revision 3 of the Instrument Line Corrective Action Program.
May 11, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of completion of review of the applicant's response to Generic Letter 92-10, Revision 1, regarding reactor vessel structural integrity.
May 12, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on FSAR Chapter 15, as revised by Amendment 80.
May 26, 1994	Memo, P. S. Tam to O. D. Kingsley (TVA), summarizing meeting on May 20, 1994.
June 13, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), finding the applicant's commitment on Generic Letter 89-13, regarding service water systems, acceptable.
June 21, 1994	Letter, S. A. Varga to O. D. Kingsley (TVA), requesting additional information on environmental issues.
June 28, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on operator response time during a steam generator tube rupture accident.
July 5, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), informing that a seismic response issue is resolved.
July 26, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), reporting results of ODCM review.
August 22, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), asking for additional information on cable separation.
August 25, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting staff comments on the inservice testing program for pumps and valves.

<u>TVA Letters</u>

April 12, 1985	Letter, J. A. Domer to NRC, transmitting additional information on the pumps and valves inservice testing program.
April 29, 1985	Letter, D. E. McCloud to NRC, transmitting additional information on the pumps and valves inservice testing program.

October 3, 1985	Letter, R. H. Shell to NRC, transmitting additional information on the pumps and valves inservice testing program.
May 1, 1986	Letter, R. Gridley to NRC, transmitting additional information on the pumps and valves inservice testing program.
January 28, 1993	Letter, W. J. Museler to NRC, requesting exemption from requirements of 10 CFR Part 50, Appendix G, IV.A.1. for Watts Bar Unit 1.
April 13, 1993	Letter, W. J. Museler to NRC sending information on operator response time to steam generator tube rupture.
May 22, 1993	Letter, W. J. Museler to NRC, submitting "Responses to Staff RAI—Design Bases Accident Spectra for the Steel Containment Vessel."
February 2, 1994	Letter, W. J. Museler to NRC, responding to staff questions regarding U-bolt pipe supports.
February 7, 1994	Letter, W. J. Museler to NRC providing additional information on FSAR Chapter 8.
February 18, 1994	Letter, W. J. Museler to NRC, transmitting updated information about the bypassed and inoperable status indication system.
February 18, 1994	Letter, W. J. Museler to NRC, transmitting final report on ampacity derating testing for 3M fire barrier.
February 25, 1994	Letter, B. S. Schofield to NRC, transmitting revised pages to the TVA Nuclear Quality Assurance Plan, Revision 4.
February 28, 1994	Letter, W. J. Museler to NRC, transmitting FSAR Amendment 84.
March 3, 1994	Letter, W. J. Museler to NRC, transmitting additional information on TVA's response to Generic Letter 89-10 regarding motor-operated valves.
March 4, 1994	Letter, W. J. Museler to NRC, amending previous commitment on Generic Letter 89-13 regarding service water systems.
March 8, 1994	Letter, W. J. Museler to NRC, transmitting Revision 7 of the Design Baseline and Verification Program Corrective Action Program.
March 10, 1994	Letter, W. J. Museler to NRC, transmitting the current version of the pressure temperature limits report

	based on ASME Code Case N-514, and requesting an exemption from 10 CFR 50.60.
March 11, 1994	Letter, W. J. Museler to NRC, responding to Bulletin 88-09 regarding thimble tube thinning.
March 11, 1994	Letter, W. J. Museler to NRC, transmitting Revision 3 of the Instrument Lines Corrective Action Program.
March 11, 1994	Letter, W. J. Museler to NRC, transmitting drawings listed in FSAR Table 1.7-1.
March 15, 1994	Letter, W. J. Museler to NRC, transmitting revised physical security plan.
March 15, 1994	Letter, W. J. Museler to NRC, transmitting Revision 1 to the Inservice Testing Program for Pumps and Valves.
March 15, 1994	Letter, W. J. Museler to NRC, transmitting Revision 5 of the Replacement Items Corrective Active Action Program.
March 15, 1994	Letter, W. J. Museler to NRC, submitting Revision 1 of the Inservice Testing Program.
March 28, 1994	Letter, W. J. Museler to NRC, transmitting information regarding postulated main steamline break outside containment.
March 29, 1994	Letter, W. J. Museler to NRC, transmitting updated information on Bulletin 88-08 regarding thermal stress on piping connected to the reactor coolant system.
April 2, 1994	Letter, W. J. Museler to NRC, transmitting FSAR Amendment 87.
April 2, 1994	Letter, W. J. Museler to NRC, transmitting FSAR Amendment 86 and 87.
April 6, 1994	Letter, W. J. Museler to NRC, transmitting Revision 6 of the Quality Assurance Records Corrective Action Program.
April 13, 1994	Letter, W. J. Museler to NRC, submitting additional information steam generator tube rupture analysis.
April 16, 1994	Letter, W. J. Museler to NRC, transmitting information on work control during hot functional testing.
April 21, 1994	Letter, W. J. Museler to NRC, transmitting security personnel training and qualification plan.
April 21, 1994	Letter, W. J. Museler to NRC, transmitting Revision I of the steam generator tube rupture analysis report.

April 23, 1994	Letter, W. J. Museler to NRC, informing of recent changes to emergency core cooling system evaluation model.
April 23, 1994	Letter, W. J. Museler to NRC, transmitting Revision 3 of the Offsite Dose Calculation Manual.
April 26, 1994	Letter, W. J. Museler to NRC, transmitting annual radiological environmental monitoring report.
April 27, 1994	Letter, W. J. Museler to NRC, transmitting final report on Quality Assurance Records Corrective Action Program.
May 2, 1994	Letter, W. J. Museler to NRC, transmitting Revision 1 of the probabilistic risk assessment individual plant examination report.
May 9, 1994	Letter, W. J. Museler to NRC, transmitting supplemental response regarding compliance with Regulatory Guide 1.97, Revision 2.
May 18, 1994	Letter, M. O. Medford to NRC, responding to the staff's request on environmental issues update.
May 19, 1994	Letter, D. E. Nunn to NRC, revising response to NUREG-0737, Item II.F.1.1.
May 23, 1994	Letter, D. E. Nunn to NRC, revising previous response to Generic Letter 89-13 regarding service water systems.
May 23, 1994	Letter, D. E. Nunn to NRC, transmitting additional information pertaining to the Eagle-21 process protection system.
May 23, 1994	Letter, D. E. Nunn to NRC, transmitting WCAP-13462, Revision 1, containing additional information regarding Eagle-21 process protection system.
May 24, 1994	Letter, D. E. Nunn to NRC, transmitting Process Control Program for radioactive wastes.
May 24, 1994	Letter, D. E. Nunn to NRC, transmitting additional response to NUREG-0737, Item II.B.2 regarding design of plant shielding and environmental qualification of equipment.
June 9, 1994	Letter, D. E. Nunn to NRC, informing about relocation of samples.
June 10, 1994	Letter, D. E. Nunn to NRC, transmitting plant-specific procedure for estimating degree of core damage.

June 13, 1994	Letter, D. E. Nunn to NRC, transmitting additional response to Generic Letter 92-01 regarding reactor vessel structural integrity.
June 17, 1994	Letter, D. E. Nunn to NRC, sending information regarding 3M fire retardant materials.
June 25, 1994	Letter, D. E. Nunn to NRC, transmitting additional test plans for Thermo-Lag fire-retardant material testing.
June 29, 1994	Letter, D. E. Nunn to NRC, responding to the staff's request for additional information on FSAR Chapter 8.
June 30, 1994	Letter, D. E. Nunn to NRC, responding to the staff's request for additional information on FSAR Amendment 80.
June 30, 1994	Letter, D. E. Nunn to NRC, sending additional information on severe accident mitigation design alternatives.
July 1, 1994	Letter, D. E. Nunn to NRC, responding to the staff's questions on the fire-protection program.
July 5, 1994	Letter, D. E. Nunn to NRC, sending results of gap measurement performed per Bulletin 88-11, regarding pressurizer surge line thermal stratification.
July 14, 1994	Letter, D. E. Nunn to NRC, transmitting draft change pages for FSAR Chapter 14.
July 19, 1994	Letter, D. E. Nunn to NRC, responding to the staff's request for additional information regarding NUREG-0737, Item II.D.1 regarding safety and relief valve testing.
July 20, 1994	Letter, D. E. Nunn to NRC, sending additional information regarding FSAR Chapter 14.
July 22, 1994	Letter, D. E. Nunn to NRC, submitting Revision 2 of the Inservice Testing Program.
July 22, 1994	Letter, D. E. Nunn to NRC, transmitting Revision 2 of the Inservice Testing Program for Pumps and Valves.
July 22, 1994	Letter, D. E. Nunn to NRC, sending additional information regarding the Cable Issues Corrective Action Program.
July 27, 1994	Letter, D. E. Nunn to NRC, withdrawing commitment to insulate the steam generator reference log.

July 29, 1994	Letter, D. E. Nunn to NRC, sending additional information regarding electrical separation to supplement FSAR Chapter 8.
July 29, 1994	Letter, D. E. Nunn to NRC, transmitting proprietary and nonproprietary copies of "Westinghouse Setpoint Methodology for Protection Systems, Watts Bar Units 1 and 2."
August 15, 1994	Letter, D. E. Nunn to NRC, regarding SGTR operator response time.
August 18, 1994	Letter, D. E. Nunn to NRC, sending information on operator response time.
August 18, 1994	Letter, D. E. Nunn to NRC, sending additional information on FSAR Amendment 79.
August 19, 1994	Letter, D. E. Nunn to NRC, transmitting "Watts Bar Nuclear Plant (WBN) — Final Safety Analysis Report (FSAR) Chapter 14, 'Initial Test Program'."
August 19, 1994	Letter, D. E. Nunn to NRC, transmitting "Watts Bar Nuclear Plant (WBN) — Final Safety Analysis Report (FSAR) — Amendment 88."
August 23, 1994	Letter, D. E. Nunn to NRC, clarifying commitment on instrument air supply system.

APPENDIX E

PRINCIPAL CONTRIBUTORS

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Same as 8. above.

10. SUPPLEMENTARY NOTES

Docket Nos. 50-390 and 50-391

11. ABSTRACT *(200 words or less)*

Supplement No. 14 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 with (1) additional information submitted by the applicant since Supplement No. 13 was issued, and (2) matters that the staff had under review when Supplement No. 13 was issued.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
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