October 25, 1993

Docket Nos. 50-390
50-391

See SER's

Dr. Mark O. Medford, Vice President
Technical Support
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: WATTS BAR NUCLEAR PLANT - ISSUANCE OF SAFETY EVALUATION REPORT, SUPPLEMENT 12 (TAC NOS. M63597, M74801, M76742, M76973, M77138, M77139, M77569, M79717, M80346, M83837, M83838, M84234, M84235, M84776, M84777, M85488, M85489, M85802, M85803, M86037, M86038)

The U.S. Nuclear Regulatory Commission staff has completed Safety Evaluation Report, Supplement 12, related to the operation of Watts Bar Nuclear Plant, Units 1 and 2 (NUREG-0847, Supplement 12). Twenty (20) copies of the report are enclosed for your use. Also enclosed is a copy of a related notice of availability which has been sent to the Office of the Federal Register for publication. Any comments should be addressed to Peter Tam, Project Manager for Watts Bar.

Sincerely,

Original signed by
Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

AA3

Enclosures:
As Stated

cc w/enclosures:
See next page *see previous concurrence

OFC	PDII4/LA*	PDII4/PM*	OGC*	PDII4/D*	ADRII*	DRPE/D
NAME	BClayton	PTam		FHebdon	GLainas	SVarga
DATE	10/14/93	10/15/93	10/15/93	10/05/93	10/06/93	10/6/93

DOCUMENT NAME: SSERLETT

DFOI

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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Availability of Safety Evaluation
Report Supplement

cc w/enclosure:
See next page

OFC	PDII4/LA	PDII4/PM	OGC *	PDII4/D	ADRII	DRPE/D
NAME	BClayton	PTam		FHebdon	GLainas	SVarga
DATE	9/14/93	10/5/93	10/ /93	10/5/93	10/6/93	10/ /93

DOCUMENT NAME: SSERLETT

* See separate concurrence

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NUREG–0847
Supplement No. 12

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

October 1993

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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), and Supplement No. 11 (April 1993), 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.

CONTENTS

CONTENTS (continued)

ABBREVIATIONS

ADGB	additional diesel generator building
AEH	analog electrohydraulic
AFW	auxiliary feedwater
ANI	authorized nuclear inspection
ANII	authorized nuclear inservice inspection
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BISI	bypassed and inoperable status indication
BIT	boron injection tank
BTP	branch technical position
CAP	corrective action program
CFR	Code of Federal Regulations
CLOC	closed loops outside containment
CNPP	Corporate Nuclear Performance Plan
COLR	Core Operating Limits Report
CR	control room
CRDR	control room design review
CSB	Containment Systems Branch
DCCS	TVA Document Control Change System
DCRM	Document Control and Records Management
DG	diesel generator
EAL	emergency action level
ECC	emergency core cooling
ECCS	emergency core cooling system
EOP	emergency operating procedure
EPG	emergency procedures guideline
ERCW	essential raw cooling water
ESF	engineered safety feature
FFD	failed fuel detector
FR	function restoration
FSAR	Final Safety Analysis Report
GDC	general design criterion
GL	generic letter
HVAC	heating, ventilation, and air conditioning
IDI	integrated design inspection
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IPEEE	individual plant evaluation on external events
IST	inservice testing
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LWR	light-water reactor
MELB	moderate energy line break
MSIV	main steam isolation valve
MSLB	main steamline break
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
OIR	open item report
OPC	overspeed protection controller
PCT	peak cladding temperature
PDM	Pittsburgh–Des Moines Steel Corp.
PORV	pilot-operated relief valve
PRT	pressurizer relief tank
PSI	preservice inspection
PTS	pressurized thermal shock
QA	quality assurance
RAI	request for additional information
RCS	reactor coolant system
RCSV	reactor coolant system vent
RCVS	reactor coolant vent system
RG	regulatory guide
RHR	residual heat removal
RP	regulatory position
RTD	resistance temperature detector
RWST	refueling water storage tank
SBLOCA	small-break loss-of-coolant accident
SER	Safety Evaluation Report
SGTR	steam generator tube rupture
SI	safety injection
SMAW	shielded metal arc weld
SP	special program
SRP	Standard Review Plan

SSE	safe-shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SSI	soil-structure interaction
TAC	technical assignment control
TI	temporary instruction
TMI	Three Mile Island
TRM	Technical Requirements Manual
TVA	Tennessee Valley Authority
UHI	upper head injection
WBN	Watts Bar Nuclear Plant
WBNPP	Watts Bar Nuclear Performance Plan
WISP	Workload Information and Scheduling System
WOG	Westinghouse Owners Group

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by 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), and Supplement No. 11 (SSER 11, April 1993). As of this date, the staff has completed review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 71 (up to Amendment 74 for Chapter 14).

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

In addition to the guidance of 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 is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.[1] Appendix E is a list of principal contributors to this supplement. A supplement to Appendix Z is included. Appendices C, D, and F–Y are not changed by this SSER.

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

Mr. Peter S. Tam
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SSERs that followed. This section 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.

[1] Availability of all material cited is described on the inside front cover of this report.

Issue[2]	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)	Updated (SSER 5)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC M71919)	Resolved (SSER 9)	3.10

[2] 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
	(b) Environmental (TAC M63591)	Under review (SER)	3.11
(5)	Preservice inspection program (TAC M63627)	Resolved for Unit 1 (SSER 10)	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)	Under review (SSER 3)	8.2
(12)	Fire-protection program (TAC M63648)	Under review (SER)	9.5.1
(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)	Updated (SSER 5)	17
(18)	Seismic classification of cable trays and conduit (TAC R00508, R00516)	Resolved (SSER 8)	3.2.1, 3.10
(19)	Seismic design concerns (TAC 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

Issue	Status	Section
(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 (TAC M79718, M80345)		
(a) Feedwater check valve slam	Under review (SSER 6)	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)	Under review (SSER 7)	3.11
(25) Health Physics Program (TAC M63647)	Resolved (SSER 10)	12
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TAC 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

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. Issue 43 was added in SSER 6. This section 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

Issue	Status	Section
(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
(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 (TAC 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)	Updated (SSER 7)	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC M63649)	Resolved (SSER 7)	8.3.1.6

Issue	Status	Section
(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)	Under review (SSER 3)	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)	Under review (SER)	8.3.3.1.1
(43) Safety parameter display system (TAC M73723, M73724)	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 identified 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) (TAC M77132, M77133)	Resolved (SSER 10)	4.4.8

Proposed Condition	Status	Section
(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 (TAC 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)	Under review (SSER 3)	8.3.2.2
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC 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)	Updated (SSER 5)	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

Proposed Condition	Status	Section
(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) (TAC 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)	Under review (SER)	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
(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 (TAC M63657, M83973)	Resolved (SSER 10)	13.6.4
(39) Control of heavy loads (NUREG–0612) (TAC M77560)	Under review (SSER 3)	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)	Updated (SSER 12)	15.4.3
(42) Loose-parts monitoring system (TAC M77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC M73723, M73724)	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)

(3) Fracture toughness requirements (Section 5.3.1.1, SER) (TAC M85712 and M85713)

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).

Implementation status: Full implementation expected by April 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); to come.

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

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 March 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) Design Baseline and Verification Program (TAC M63594; TI 2512/019)

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

Implementation status: Full implementation expected by March 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); 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 March 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); 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 April 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); 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 March 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); 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 March 1994.

NRC inspections: 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 March 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); 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 March 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); 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).

Implementation status: Full implementation expected by March 1994.

NRC inspections: Inspection Reports 50–390, 391/90–14 (August 3, 1990); 50–390, 391/90–23 (November 19, 1990); 50–390, 391/91–02 (March 6, 1991); 50–390, 391/91–03 (April 15, 1991); 50–390, 391/91–26 (December 6, 1991); 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 pre-operational 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.

Implementation status: Full implementation expected by April 1994.

NRC inspections: Inspection Reports 50–390, 391/90–06 (April 25, 1990); 50–390, 391/90–08 (September 13, 1990); 50–390, 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); to come.

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

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG- 1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), January 23, 1991; to come.

Implementation status: Full implementation expected by November 1993.

NRC inspections: 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); to come.

(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.

Implementation status: Full implementation expected by April 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) Seismic Analysis (TAC R00514; TI 2512/030)

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) Civil Calculation Program (TAC R00514)

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: Final calculations transmitted by letter, W. J. Museler (TVA) to NRC, July 27, 1992.

NRC audits: 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. 0. Medford (TVA), May 26 and December 18, 1992 and July 3, 1993; 50–390, 391/93–07 (February 19, 1993); to come.

(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 November 1993.

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. 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); 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: Full implementation expected by October 1993.

NRC inspections: Inspection Report 50–350, 391/93–56 (September 20, 1993); to come.

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

Program review status: Complete: NUREG–1232, Vol. 4; Section 18.1 and Appendix L of SSER 6.

Implementation status: Full implementation expected by March 1994.

NRC inspections: 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 by April 1994.

NRC inspections: 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 August 1993.

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 to come.

NRC inspections: 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); to come.

(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 March 1994.

NRC inspections: 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 March 1994.

NRC inspections: 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).

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

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

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

In a letter of June 22, 1992 (transmitting Westinghouse Topical Report WCAP-12773), and supplemented by a letter of March 26, 1993, Tennessee Valley Authority (TVA) proposed to eliminate pressurizer surge line rupture as a design basis for Watts Bar Units 1 and 2. The request was based on a leak-before-break (LBB) analysis of the pressurizer surge line, as permitted by General Design Criterion 4 (GDC 4) of 10 CFR Part 50 (Appendix A).

GDC 4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures in high-energy piping in nuclear power units. Implementation of the LBB technology permits the removal of pipe whip restraints and jet impingement barriers as well as other related changes in operating plants. The acceptable technical procedures and criteria of the LBB evaluation are defined in NUREG-1061 and summarized, in part, as follows:

> The forces and moments of pressure, deadweight, thermal expansion, and earthquake associated with normal operation and safe-shutdown earthquake (SSE) should be considered. The location(s) at which the highest stresses coincident with poorest material properties for base metals, weldments, and safe ends should be identified; a through-wall flaw should be postulated at those location(s). The flaw size should be large enough so that the leakage is assured of detection with at least a margin of 10 using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.
>
> Operating experience should be provided to show that the pipe will not experience stress corrosion cracking, fatigue, or water hammer. The operating history should include system operational procedures; system or component modification; water chemistry parameters, limits, and controls; resistance of piping material to various forms of stress corrosion; and performance of the pipe under cyclic loadings.
>
> The materials data provided should include types of materials and material specifications; stress-strain curves and J-R curves (not required if limit load analysis is used in the stability analysis;) long-term effects such as thermal aging; and other limitations to materials data (e.g., J maximum, and maximum crack growth). The piping materials must be free from brittle cleavage-type failure over the full range of the system operating temperature.
>
> The postulated leakage flaw should be shown to be stable under normal plus SSE loads for long periods of time; that is, crack growth is minimal during an earthquake. A flaw stability analysis should be done to show that the leakage flaw is stable under larger loads (at least 1.4 times the normal plus SSE loads). However, the margin of 1.4 may be reduced to 1.0 if the individual normal and seismic loads are summed absolutely.
>
> Under normal plus SSE loads, there should be a safety margin of at least 2 between the leakage-size flaw and the critical-size flaw, to account for the uncertainties inherent in the analyses and leakage detection capability.

The staff performed independent calculations to check TVA's results on leakage rate and flaw stability on the pressurizer surge line of Watts Bar Units 1 and 2 and finds that this line complies with GDC 4. The following evaluation has been transmitted to the applicant by letter (P. S. Tam (NRC) to M. O. Medford (TVA), April 28, 1993):

> The pressurizer surge line in Watts Bar Units 1 and 2 has a nominal diameter of 14 inches with a minimum wall thickness of 1.250 inch. The piping material is austenitic wrought stainless steel A-376/TP304.
>
> TVA used combined normal and faulted loadings in the flaw stability analysis to assess margins against pipe rupture at postulated faulted load conditions. The normal operating loads include pressure, deadweight, and thermal expansion; the faulted loads include seismic, stratification temperatures as high as 320 °F (160 °C), heatup/cooldown case, and the forced cooldown case. In the worst loading case for the stability analysis, all individual normal,

faulted, and seismic loads were summed absolutely. The highest stress locations are at the shielded metal arc weld (SMAW) close to the hot-leg nozzle.

Watts Bar Units 1 and 2 have reactor coolant system (RCS) pressure boundary leak detection systems which satisfy the guidelines of Regulatory Guide 1.45 so that a leakage of one gallon per minute (gpm) in one hour can be detected. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems. The staff determined that the margin is a factor of 10 on leakage and is consistent with NUREG–1061.

Since Watts Bar Units 1 and 2 are not in operation, there is no plant operating history showing that the pipe has not experienced stress corrosion cracking, water hammer, or other degradation. However, based on the operating histories of other Westinghouse plants, the staff concludes that stress corrosion, water hammer, and other degradation mechanisms are not issues for Watts Bar Units 1 and 2.

TVA provided material properties for the surge line based on the Certified Materials Test Report. In the LBB calculations, the minimum material properties at average pipe section temperature were used for the flaw stability evaluations; the average material properties were used for the leakage rate calculations.

TVA showed that the postulated leakage flaw is stable under normal plus SSE loads. The safety margin in terms of applied loads was shown to exceed 1.0 and it satisfies the guidance of NUREG–1061.

The staff conducted independent leak rate and flaw stability calculations and found that there was a discrepancy of 35 percent between the reported leakage flaw size (4.2 inches) and that calculated by the staff (5.65 inches) by using the PICEP computer code for Loading Case B/G. Case B, being a normal operating case at 653 °F, consists of the algebraic sum of pressure, deadweight and stratification of 36 °F; Case G, being a faulted heatup/cooldown case at temperatures between 135 °F and 455 °F, consists of the algebraic sum of seismic, pressure, deadweight, and stratification of 320 °F. In order to validate its leak rate calculation, TVA, in its letter of March 26, 1993, cited three sets of comparisons between predicted and measured values for various crack geometries from WCAP-11256 Supplement 1: (1) Figures 2.4–7 (Battelle–Henry Model) and 2.4–8 (Fauske–Henry–Griffith Model) for flow through cracks of 2.5 inches long and 0.20 mm to 1.12 mm wide; (2) two French data points in Table 2.4–2 for flow through a crack in a plate; (3) seven more data points in Figure 2.4–9 for flow through an annular clearance in a plate. Except for the second set of comparisons, the staff determines that the test data points cited by TVA are not applicable. Figure 2.4–7 is discounted because only two data points appeared in the flow rate range under 10 gpm (see Westinghouse document dated April 1987, "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Pressurizer Surge Lines of South Texas Projects Units 1 and 2") and the comparison made for these two was not favorable; Figure 2.4–8 is discounted because a comparison to the same test data (see EPRI NP–3596–SR, Rev. 1) indicates that predicted flow rates from the PICEP computer program are much closer to the measured values. The third comparison involves flows through an annular clearance and is not considered applicable because the geometry of annular clearances differs significantly from crack geometries.

So far, the only pertinent data to validate the Westinghouse calculation are the two French data points. This prompts the staff to conclude that Westinghouse's leak rate calculation is not acceptable unless more data can be submitted for evaluation and validation.

However, TVA's application of LBB to surge lines is still acceptable because TVA has demonstrated in its letter of March 26, 1993, that an alternative J–R analysis for the same stress as used in the Comanche Peak analysis, but using a leakage flaw size of 4.2 inches long, gives an applied J value of 4431 in.-lb/in.^{2} TVA also confirmed that J_{max} is 5500 in.-lb/in.^{2} and Watts Bar Units 1 and 2 surge line maximum stresses at the critical locations are approximately 30 percent lower than the comparable Comanche Peak Unit 2 stresses. Using twice the leakage crack size from PICEP, 11.3 inches, and adjusting for the 30-percent-lower stresses, the staff concludes that the actual applied J value for Watts Bar Units 1 and 2 would be even lower than 4431 ft-lb/in^2, and the margin between the leakage-size flaw and the critical-size flaw meets the minimum requirement of two for the worst load combination. The margin satisfies the guidance of NUREG–1061.

> The staff concludes that TVA's LBB analysis is consistent with the criteria in NUREG-1061, Volume 3, and therefore, complies with GDC 4. Thus, the probability of large pipe breaks occurring in the surge line is sufficiently low so that dynamic effects associated with postulated pipe breaks need not be a design basis. TVA may eliminate pressurizer surge line rupture from the design basis for Watts Bar Units 1 and 2.

The staff's efforts were tracked by TAC M83837 and M83838.

3.7 Seismic Design

3.7.3 Seismic Subsystem Analysis

In SSER 6, the staff reported that since the original design of the structures, systems, and components at Watts Bar, a number of problems were identified by various sources (inspection reports, TVA internal reports, employee concerns, etc.) in the areas of design, construction, and inspection/quality assurance of the plant features. To resolve these issues, the applicant was conducting corrective action programs to assure that plant features meet upgraded design criteria and licensing commitments. One phase of these programs consists of an engineering evaluation to validate the adequacy of the existing design. The staff then proceeded to describe the applicant's use of the "worst case," "critical case," and "bounding calculation" approaches.

The staff stated that "since all three approaches rely on either the actual configuration and attributes or the hypothetical combination of attributes, which is more severe, the staff considers the use of worst case, critical case, and bounding calculation approach acceptable." However, at that time, the staff had not yet reviewed the procedures used by the applicant to perform the walkthrough or the basis for grouping the configurations and identifying critical attributes. Therefore, Outstanding Issue 19(e) was opened to track this effort.

The staff subsequently performed an integrated design inspection (IDI) of the Watts Bar Unit 1 civil/structural areas during the period July 13 through August 7, 1992. The inspection findings were published in Inspection Report 50-390/92-201. The inspection team reviewed the applicant's procedures for evaluating the worst case, critical case, and bounding calculations for cable tray supports, conduit and conduit supports, and heating, ventilation, and air conditioning (HVAC) duct and duct supports. For the most part, the team found that the procedures for evaluating worst case, critical case, and bounding calculations provided an adequate basis to resolve the technical issues identified in the corrective action programs. However, the inspection report did identify some deficiencies with the implementation of the applicant's programs. The deficiencies relating to the evaluation of worst case, critical case, and bounding calculations were resolved in Inspection Report 50-390/93-201. The staff considers that the IDI and the resolution of deficiencies documented in these inspection reports constitute a sufficient review of the applicant's procedures for performing walkdowns and identifying critical attributes for Unit 1. Therefore, Outstanding Issue 19(e) is considered resolved for Unit 1.

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

In the SER, the staff stated that "The applicant will be required to comply with Section XI of the ASME Code endorsed by 10 CFR 50.55a 12 months before issuance of an operating license and/or the Technical Specifications," and proposed a license condition to ensure this is done. Subsequent to that, the staff issued Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing (IST) Programs." The applicant responded to Generic Letter 89-04 by letter dated August 21, 1989. In that letter, the applicant committed to submit the revised ASME Section XI Inservice Pump and Valve Test Program, in accordance with 10 CFR 50.55a(g)(4)(i) [now 10 CFR 50.55a(f)(4)(i)], six months before the projected date of operating license issuance. With a current target date of April 1994, the staff expects to receive the Watts Bar IST Program submittal in October 1993. On the basis of this commitment, the staff considers Proposed License Condition 2 no longer required. The staff's evaluation of the IST Program will be included in a future supplement to the SER, before an operating license is issued.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.2 Design Bases

4.4.2.3 Core Flow

The SER stated, quoting Section 4.4.1.3 of the FSAR, that a minimum of 92.5 percent of the reactor coolant flow will pass through the fuel rod region. Section 4.4.3 (below) documents the staff's evaluation of a change to this percentage.

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.2 Core Flow

In letters of June 15 and August 31, 1987, and March 31 and July 26, 1988, the applicant informed the staff of its intent to modify the internal components of the reactor to change the direction of the flow from downflow to upflow. In letter of March 8, 1993, the applicant submitted Westinghouse Topical Report WCAP-11627, "Upflow Conversion Safety Evaluation Report, Watts Bar Units 1 and 2," to support the changes made by Amendments 63 and 71 to Chapters 4 and 15 of the FSAR. The purpose of this conversion to upflow is to reduce the hydraulic pressure differential that exists across the baffle joints in the downflow configuration and to reduce the potential for fuel rod damage resulting from baffle joint cross-flow jetting.

The modification involved converting the reactor vessel downflow design to an upflow design. This upflow conversion consists of making changes to the reactor components, plugging the inlet flow holes in the core barrel, and drilling holes in the top former plate. These modifications change the flow from being downflow between the core barrel and baffle plate, to upflow, and have the effect of increasing the core bypass flow from 7.5 to 9.0 percent. Changing the flow path reduces the pressure differential across the baffle plates, thus eliminating the jetting of coolant between the joints and the baffle plates. The evaluation by Westinghouse included the impact of the modification on the internal components, the fuel assembly integrity, the core barrel plug, and the effect of the increase in bypass flow on appropriate LOCA, non-LOCA, and steam generator tube rupture postulated accidents.

The applicant evaluated the upflow modification and concluded that:

(1) The upflow modification has no direct impact on the reactor core system under FSAR-specified earthquake loading conditions, and has insignificant impact on LOCA forces.

(2) The upflow modification will reduce the crossflow from the baffle joint gap, while maintaining fuel rod structural integrity.

(3) A minimum of 91 percent of the thermal flow will pass through the fuel rod region of the core, and will be effective for fuel rod cooling. The remaining 9-percent core bypass flow (increasing from 7.5%) is not considered effective for heat removal. The 1.5-percent increase in core bypass flow results in a 1.62-percent reduction in core flow, which does not impact the current core limits (overtemperature delta-T/overpower delta-T) setpoints, and insignificantly affects the margins to non-LOCA analyses acceptance criteria.

(4) Coolability of the baffle/barrel region is enhanced since the flow through the baffle/barrel region increases as a result of the upflow modification.

(5) Both a large-break (a double-ended cold-leg guillotine break with a cross-sectional area equal to or greater than 1.0 square foot) with 0.6 break discharge coefficient, and a small-break (less than 1.0 square foot) LOCA have been analyzed using NRC-approved codes; the peak cladding temperatures calculated are 2126 °F (1163 °C) for a large-break and 2089 °F (reported as 1446.1 °F (786 °C) in WCAP-11627, but see Section 15.3.1 of this SSER for an updated analysis) for a small-break LOCA, respectively. These are below the acceptance criterion of 2200 °F (1204 °C) specified in 10 CFR 50.46.

(6) The core barrel plug has been designed and installed to meet the applicable requirements of ASME Boiler and Pressure Vessel Code, Section III, Division I, 1986 Edition, for all service loadings.

The staff has reviewed the applicant's submittals and has affirmed that the applicant's analyses were performed using approved computer codes, and the results of the accident analyses meet the acceptance criterion specified in 10 CFR 50.46. Furthermore, there is no significant impact on the margins to non-LOCA acceptance criteria and core limit setpoints. The staff, therefore, concludes that the proposed upflow modification is acceptable. This effort was tracked by TAC M85802 and M85803.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

In SSER 10, the staff stated that the issue of preservice inspection (PSI) was resolved for Unit 1. Subsequently, the staff discovered that a number of relief requests pertaining to PSI have not been properly addressed. The results of the review of these requests are given in Appendix Z to this SSER, as a supplement to Appendix Z, originally published in SSER 10. The results do not change the staff's conclusion stated in SSER 10 regarding the PSI program. The staff's efforts were tracked by TAC M86037 and M86038.

In the SER, the staff stated that "The initial inservice inspection program has not been submitted by the applicant. The staff will evaluate the program after the applicable ASME Code edition and addenda can be determined based on Section 50.55a(b) of 10 CFR 50, but before the first refueling outage when inservice inspection begins." The staff initiated proposed License Condition 4 to track this effort.

In a letter of September 10, 1993, the applicant committed to submit the Watts Bar Inservice Inspection (ISI) Program within six months after receiving the operating license. This will allow the applicant to incorporate the code requirements in effect 12 months before the operating license is issued. On the basis of this commitment, the staff considers proposed License Condition 4 no longer required. The staff's evaluation of the ISI Program will be published before the beginning of the first refueling outage.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

In Section 5.2.5 of the SER, the staff describes various ways of detecting reactor coolant intersystem leakage. In that description, the staff stated that if leakage is alarmed and confirmed in a flow path with no indicators, then the Technical Specifications should require that a water inventory material balance be initiated within 1 hour to determine the extent of the leakage. In a letter of March 15, 1993, the staff asked the applicant (TVA) to provide the appropriate technical specification. In a letter of June 18, 1993, the applicant responded, indicating that it could not find the leakage paths that are referred to in the SER. The staff is also not aware that any such leakage paths exist as described in the SER. Even if such a leakage path did exist, a requirement in the Technical Specifications for an inventory balance is inappropriate. Any leakage that is identified, including intersystem leakage, must be quantified in order to determine if the technical specification leak rate limit has been exceeded. However, the manner in which this determination is made is immaterial and is dependent on the source of the leakage. The staff, therefore, retracts the requirement identified in the SER and concludes that no requirements are necessary related to this issue beyond those identified in the Westinghouse Revised Standard Technical Specifications (NUREG-1431).

In the June 18, 1993 letter, the applicant also expressed a concern regarding the wording in the SER related to the detection of intersystem leakage through check valves. The SER implies that leakage across check valves is detected on an individual valve basis, when, in fact, both valves must leak in order for the leakage to be detected. This is standard industry design and is acceptable because as long as one valve is not leaking there is no intersystem leakage. However, the valves are leak tested on an individual basis. This is just a matter of clarification and does not change the staff's conclusions identified in the SER or any of its supplements. The reactor coolant pressure boundary leakage detection systems are, therefore, still acceptable. The staff's effort was tracked by TAC M76742.

5.4 Component and Subsystem Design

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

The reactor coolant system vents (RCSVs) are addressed in Item II.B.1 of the TMI Action Plan, NUREG-0737, and are required by 10 CFR 50.44(c)(3)(iii). Criteria related to the RCSV are provided in the TMI Action Plan, and in Section 5.4.12 of the Standard Review Plan (SRP).

The applicant described the RCSV in a letter dated August 12, 1982, and in the Watts Bar FSAR. In SSER 2, the staff noted that TVA had committed to install an acceptable RCSV system before fuel loading, and to use venting guidelines developed by the Westinghouse Owners Group (WOG). These commitments were found acceptable pending verification that the RCSV system was installed, although the staff had not then reviewed the RCSV system design. In SSER 5, the staff reported that the RCSV system was installed. On October 2, 1992, TVA submitted a revised response to the 1982 letter to reflect the removal of the upper head injection (UHI) system (evaluated in Section 6.3.1.1 of SSER 7).

The staff reviewed the design of the RCSV system as reported in the applicant's October 2, 1992, letter. This safety evaluation that follows was issued to TVA in a letter of April 28, 1993.

5.4.5.1 System Description

The RCSV consists of two subsystems: the reactor coolant vent system (RCVS), and the pressurizer power-operated relief valves (PORVs). The RCVS vents the reactor vessel and, effectively, the hot legs, while the PORVs vent the pressurizer. Both subsystems discharge to the pressurizer relief tank.

Beginning at the top of the reactor vessel (RV), the RCVS consists of the following:

- A nominal 1-inch-diameter Class 1 pipe is connected to one of the standpipes remaining after removal of the UHI system.
- A 3/8-inch-diameter orifice is provided in the pipe and forms the boundary between Class 1 and Class 2 piping/components.
- Nominal 1-inch-diameter Class 2 piping is provided between the orifice and two parallel "inboard" valves.
- These valves are solenoid-operated open/close Class 2 isolation valves. They are normally closed, fail-closed valves powered from different vital power supplies, and may be operated from the control room (CR). They are qualified to IEEE Standard 344-1975 and are included in the Westinghouse Valve Operability Program, which is an acceptable alternative to Regulatory Guide 1.48.
- Nominal 1-inch-diameter Class 2 piping connects from these valves to a common pipe, which then branches to two "outboard" valves, one in each branch.
- The outboard valves are throttle valves which otherwise are as described above for the inboard valves.
- Outboard valve discharge is via common non-nuclear piping that leads to the pressurizer relief tank (PRT).

The pressurizer PORVs are solenoid-operated valves and the venting path also serves as part of the cold overpressure protection system.

The applicant stated that vibration behavior associated with flow through the RCSV system is monitored during preoperational tests, and that supports are realigned to bring vibration into acceptable limits if unacceptable vibration is present. Staff acceptance is conditional upon TVA demonstrating acceptable vibration behavior. TVA should submit a brief statement when such demonstration can be made.

5.4.5.2 Evaluation

The evaluation with respect to each of the SRP Section 5.4.12 guidance items for the RCSV system is as follows:

(1) "Vent paths shall be provided on high points of the reactor coolant system (including the pressurizer on PWRs) to vent gases which may inhibit core cooling. For reactors with U-tube steam generators, procedures shall be developed to remove gases from the U-tubes since it is impractical to individually vent the thousands of U-tubes."

There are two RCSV paths: the RCVS and the pressurizer PORV. The RCVS vents the reactor vessel and, effectively, the hot legs, whereas the PORV vents the top of the pressurizer. The UHI standpipe extends 61 inches into the top of the reactor vessel, and the bottom of the standpipe is about 116 inches above the top of the hot-leg nozzle. Thus, the RV high point vent will not vent the top 61 inches of the RV head. This inability is of little practical concern since:

- Expansion of the trapped bubble can be controlled via the vent whenever the bubble is below the 61-inch elevation. The 116-inch distance to the hot leg effectively removes any potential concern that a slowly expanding bubble will enter the hot leg during natural circulation.
- Potential bubble volume is small in comparison to the unvented volume in the steam generators.
- Bubble volume during slow RCS depressurization can be held constant once the bubble reaches the bottom of the standpipe, and bubble expansion, therefore, will not significantly delay depressurization and cooldown.

The design satisfies the requirement for venting RCS high points. Procedures for Watts Bar have been developed from the WOG emergency procedures guidelines (EPGs). These procedures are addressed in Item 11 below.

(2) "A single failure of a vent valve, power supply, or control system shall not prevent isolation of the vent path."

For practical purposes, there are two parallel flow paths with redundant, independently powered isolation valves in each flow path. One single active failure can prevent venting the RCS through only one flow path. The redundant path then remains available for venting the RCS. The valves are fail-closed valves, and loss of power causes them to close. Failure of one valve to close will not prevent isolation of the vent path because of the series arrangement of valves in the piping.

(3) "Sufficient redundancy in the design shall be incorporated to minimize the probability of inadvertent actuation. Other methods to reduce the chances of inadvertent actuation, such as removing power or administrative controls, may be considered."

With two isolation valves in each flow path, and each flow path powered from different emergency buses, the failure of any one valve or power control will not inadvertently open a vent path. TVA has reviewed the design, control systems, and power supplies with respect to inadvertent valve actuation and has found no mechanism that will open more than one valve at a time other than deliberate actions from the control room. Each isolation valve is a fail-closed, normally deenergized valve. To avoid human error, because the valves do not have a power lockout, the valve control switches are distinctly labeled and separated. Because there are redundant valves in each flow path, one inadvertent actuation of a valve will not open a flow path to the RCS.

There is positive indication of the position of these valves in the control room. An inadvertent opening or a failure to close the vent will be detected by the vent valve position indications, and by the temperature indicator downstream of the first isolation valve, which will alarm in the control room.

(4) "Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for reactor pressure boundary must be met."

The RCVS is connected via an orifice to the unused standpipe that remains from the removed UHI system. This orifice forms the transition from Safety Class 1 of the standpipe and attached pipe to Safety Class 2. Between the orifice the outboard valve is ASME Code, Section III, Category I, Class 1 or Class 2 piping. Piping downstream of the outboard valve is American National Standards Institute (ANSI) B31.1, Category I(L).

(5) "The size of the vent line should be kept smaller than the size corresponding to the definition of a LOCA (10 CFR Part 50, Appendix A) to avoid unnecessary challenges to the ECCS."

The RCVS has a 3/8-inch-diameter orifice. The flow through such an orifice is within the capacity of one centrifugal charging pump. Therefore, the sizes of the vent lines are smaller than the size corresponding to the definition of a LOCA. There are no orifices in the PORV piping; however, the PORV is within the small-break LOCA range and block valves are available to isolate a flow of water.

(6) "Vent paths to the containment should discharge into areas that provide good mixing with containment air and are able to withstand steam, water, noncondensibles, and mixtures of the above."

The RCVS and PORVs are routed to the pressure relief tank. This discharge path has been previously analyzed (Watts Bar SER, Section 15.5.3) with respect to PORV openings and found acceptable. Flow rate from the RCVS will be significantly smaller than flow rate from the PORVs, and potential impacts of RCVS flow are expected to be no greater than those for flow from the PORVs.

(7) "The vent system shall be operable from the control room and provide positive valve position indication. Power shall be supplied from emergency buses."

The RCVS is operated from the control room. It can be tested for operability by cycling each valve through a full cycle from the control room. Valve movement can be verified by positive indications in the control room. The positive indication is from stem position switches on the inboard valves, and via linear variable differential transformers for the outboard valves. Power is provided by emergency buses, with the valves in each of the two flow paths powered by different emergency buses.

(8) "It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms."

All instruments and displays for the RCVS and PORV, which are located in the main control room, must be considered in the human-factor analysis required by NUREG-0737, Item I.D.1, "Control Room Design Review (CRDR)". Further, the staff has noted that task analyses were not conducted for some parts of the emergency operating procedures (EOPs) that may involve venting via the RCVS and the PORVs (see page 2 of Appendix L to SSER 6).

The task analysis difficulties, as applicable, must be resolved and a satisfactory staff audit must be completed to meet this requirement. The staff will address these issues in a site audit of the Watts Bar CRDR before an operating license is issued.

(9) "Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves."

TVA stated that the vent system is tested in accordance with IWV for Category B valves. This test must be satisfactorily completed.

(10) "The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344–1975 as supplemented by Regulatory Guides (RGs) 1.100 and 1.92, and SEP 3.92, 3.43, and 3.10. Equipment qualifications are in accordance with the May 23, 1980, Commission Order and Memorandum (CLI-80-21)."

TVA stated that (a) the RCVS is seismically designed and qualified to IEEE 344–1975, (b) RG 1.100 endorses IEEE 344–1975, (c) the valves conform to the intent of RG 1.92 and SRP Sections 3.9.2, 3.9.3, and 3.10, and (d) the RCVS is on the equipment qualification list of electrical equipment at Watts Bar that is qualified in accordance with 10 CFR 50.49. The staff will continue to review the Watts Bar equipment qualification program; this issue will be addressed as part of those efforts.

(11) "Procedures to effectively operate the vent system must consider when venting is needed and when it is not needed. Variety of initial conditions from which venting may be required shall be considered. Operator actions and the necessary instrumentation shall be identified."

Watts Bar has adopted and is using EOPs based on the WOG EPGs. The WOG EPGs provide instructions on how to determine when to use the RCVS, and cover a wide range of RCS pressure and temperature conditions. Function Restoration (FR) I.3, "Response to Voids in Reactor Vessel," addresses use of the RCSVs, and venting is identified in other parts of the EPGs. However, Section 5.5.6.2 of the Watts Bar FSAR states that "The system should not be used unless an inadequate water level is determined in the reactor vessel by the Reactor Vessel Level Instrumentation System...." This statement was verified by TVA's letter of March 19, 1993. This statement is incorrect since the Watts Bar functional restoration guidance places reliance upon incore thermocouples, with level instrumentation in a backup or guidance role, as described in a TVA letter dated January 24, 1992. The staff additionally confirmed that the WOG EPGs are consistent with reliance upon the incore thermocouples, not the level instrumentation. The FSAR discrepancy should be corrected.

As noted in Item 8, above, the staff knows that task analyses were not conducted for some parts of the EOPs that may involve venting via the RCVS and the PORVs. The task analysis difficulties, as applicable, must be resolved to satisfactorily meet the Item 11 requirement. The staff will perform Inspection Procedure 42001, "Emergency Operating Procedures" (see page 13-1 of SSER 9), before an operating license is issued, and this item will be addressed.

The FSAR addresses the characteristics of postulated missiles generated by failures of RCVS components and the potential effects from fluid spray from such failures. TVA stated that no significant impacts upon safety-related equipment occurs due to such postulated failures. The staff accepts TVA's assessment.

5.4.5.3 Conclusion

The staff concludes that TVA's high-point vent system is acceptable subject to satisfactory completion of these issues, already described in detail above:

(1) demonstration of acceptable vibration behavior during preoperational testing

(2) satisfactory completion of the task analysis pertinent to emergency operating procedures

(3) obtaining a satisfactory staff audit of the control room design

(4) satisfactory completion of vent system testing in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves

(5) satisfactory completion of the staff review of the environmentally qualified list of electrical equipment pertaining to the vent system

(6) satisfactory completion of staff Inspection Procedure 42001, "Emergency Operating Procedures"

(7) correction of Section 5.5.6.2 of the FSAR

All of these items involve ongoing or planned activities associated with completion of the Watts Bar licensing process. None require additional review with respect to this SER nor will they change the SER provided they are satisfactorily completed. TVA is asked to submit a letter, before issuing an operating license, stating briefly how and when these items were completed; the staff will track implementation of these items by TAC M84776 and M84777.

When these issues are satisfactorily implemented, the Watts Bar RCSV system will meet the criteria of 10 CFR 50.46, 10 CFR 50.55(a), and General Design Criteria 1, 14, and 30, and the design for the RCSV system will be acceptable based on compliance with guidance delineated in SRP Section 5.4.12.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In the SER, the staff identified the containment design pressure as 15.0 pounds per square inch gauge (psig). By Amendment 71, the applicant revised Section 6.2 of the Final Safety Analysis Report (FSAR) to indicate that the actual design pressure was 13.5 psig. As described in FSAR Section 3.8.2, 15.0 psig is considered the maximum internal pressure and the design internal pressure is 13.5 psig. This definition is in accordance with Paragraph NE-3312(b) of Section III of the ASME Code (1983) which states that the design internal pressure may differ from the maximum containment pressure but in no case shall the design internal pressure be less than 90 percent of the maximum containment internal pressure. This redefinition of the containment design pressure does not change the conclusions reached by the staff in the SER or any of its supplements because the maximum calculated containment pressure is still 11.21 psig, as identified in Section 6.2.1.1 of SSER 5, which is still less than the design pressure. This review was tracked by TAC M84234 and M84235.

6.2.1.1 Containment Structure

Maximum Pressure and Temperature Analysis

In Section 6.2.1 of the SER, the staff discussed the results of its review of the containment peak pressure and temperature analyses. The SER stated that a spectrum of main steamline breaks (MSLBs) had been analyzed for a "generic" ice condenser plant using the LOTIC-III program, and the peak conditions were found to be 4.5 psig and 327 °F (164 °C). The limiting event (for temperature) was determined to be a 0.6 ft^2 split break at 30-percent power with failure of auxiliary feedwater runout protection. The generic plant model used in the LOTIC-III analyses was based on parameters bounding both Watts Bar and Sequoyah, and the results thus encompass both plant designs. Based on the generic analysis, 327 °F was defined as the equipment qualification temperature basis for equipment in the lower compartment. In a letter of September 13, 1993, the applicant reaffirms this temperature.

In SSER 7, the staff noted an equipment qualification concern relating to the effect of superheated steam release during on MSLB event. It was noted that the previously calculated MSLB temperature profile which peaked at 327 °F may not be valid as a result of unaccounted heat transfer through uncovered steam generator tube bundles. As a result, Westinghouse revised the LOFTRAN and LOTIC-III codes (LOFTRAN for mass-energy release and LOTIC-III for containment response) to include, respectively, (1) superheat effects and (2) the cooling effect of ice condenser drainage to the lower compartment in the vicinity of the break. The results of reanalysis (applicable to the Catawba, McGuire, Sequoyah and Watts Bar facilities) indicated that the global compartment temperature for the lower compartment is conservative relative to the 327 °F equipment qualification temperature.

The staff, however, remained concerned about local hot spots in ice condenser plants, indicating that these effects would be addressed on a plant-specific basis. Westinghouse then used the three-dimensional code COBRA-NC (and the staff used another code, COMMIX, for confirmation) to evaluate the local effects. (The COBRA-NC study was done in response to a Catawba license condition and is applicable to Sequoyah and Watts Bar also.) Both analyses indicated that there are locally elevated temperatures in the vicinity of the steam discharge. A walkdown was done at Watts Bar to view the arrangement of equipment with respect to the location of the direct steam discharge path. In addition, Sequoyah piping arrangement drawings were reviewed. Upon consideration of (1) code conservatisms relating to treatment of entrained water, (2) the location and redundancy of equipment needed for safe shutdown, and (3) the arrangement of ice condenser drains, the staff concluded in SSER 7 that the qualification and performance of plant shutdown equipment will not be impaired by locally elevated temperature near the breaks.

In an earlier letter dated May 2, 1984, the applicant stated that it had developed and utilized improved analytical techniques to reanalyze the MSLB in support of its effort to delete a proposed Technical Specifications requirement for 20,000-ppm boron concentration in the boron injection tank (BIT). The letter stated that superheat was being considered as a separate issue. In Section 15.3.2 of SSER 3, the staff approved the applicant's proposed elimination of the BIT. The paragraphs that follow address the BIT modification's potential effect on the evaluation published in SSER 7.

The applicant incorporated BIT elimination into the FSAR by Amendment 69. The BIT elimination methodology involved the relaxation of an internal Westinghouse design criterion regarding return to criticality following an MSLB. The BIT removal analysis evaluated in SSER 3 discussed the effects on MSLB mass and energy releases to the containment. The limiting break continues to be a 0.6-ft^2 break at 30-percent power with auxiliary feed pump runout as noted above. From the figure provided in the report, the lower compartment global temperature, with elimination of BIT boron injection in addition to superheating, peaks at essentially the same temperature as that previously chosen for equipment qualification purposes prior to BIT elimination.

The staff reviewed the FSAR changes. As noted above, the staff had previously determined that, due to the physical arrangement of equipment and the location of ice condenser drains, the temperature selected for use in environmental qualification of equipment in the lower containment preceding BIT elimination need not be revised upward to reflect the local high-temperature effects in the vicinity of the limiting break steam jet. It is the judgment of the staff that because (1) the peak global lower compartment temperature has not significantly increased and (2) previous studies, as reported in SSER 7, indicate that safe-shutdown equipment is located so that it is unlikely to be affected by local high temperature, BIT boron elimination will not significantly affect the environmental response of the containment or safe shutdown equipment therein. This review was tracked by TAC M80152 and M80153.

6.2.4 Containment Isolation System

In the SER, the staff reported its evaluation of the plant's containment isolation capability. In a letter dated October 20, 1988, the applicant provided a proposed corrective action program (CAP) relating to the design of the Watts Bar containment isolation system. The CAP was proposed by the applicant in response to staff concerns raised during an inspection of the Sequoyah facilities (see Sequoyah Inspection Reports 50-327/86-20 and 50-328/86-09). Among the issues raised during the Sequoyah inspection was that the use of an inside-containment isolation valve, instead of an outside-containment isolation valve, as the second piping penetration barrier in containment piping penetrations referred to as "closed loops outside containment" (CLOC) does not conform to regulatory requirements. Standard Review Plan Section 6.2.4, Paragraph II.6.e gives acceptance criteria for the use of a CLOC in conjunction with an ESF or ESF-related system having a single isolation valve outside containment, but does not give such criteria for use of a CLOC with a single inside-containment isolation valve.

In its October 20, 1988, letter and in a subsequent letter of May 12, 1989, the applicant identified the specific Watts Bar piping penetrations having a CLOC with a single isolation valve located inside the containment arrangement. The systems and associated penetrations are:

- chemical and volume control system including the normal charging penetration X-16, seal injection penetrations X-43A/B/C/D, and boron injection tank discharge penetration X-22
- residual heat removal system including hot-leg injection penetration X-17, cold-leg injection penetration X-20A/B, reactor coolant system supply penetration X-107, and spray penetrations X-49A/B;
- safety injection system hot-leg injection penetrations X-21 and X-32, relief valve discharge to pressure relief tank penetration X-24, and cold-leg injection penetration X-33
- upper head injection penetrations X-108/109
- containment spray penetrations X-48A/B
- hydrogen analyzer system penetrations X-92A/B, X-99, and X-100

Except for the case of the seal injection penetrations, each penetration has other existing valves which could be redesignated and qualified as containment isolation valves (see Attachment C to applicant's October 20, 1988, letter), however, these valves were not intended to serve as isolation boundaries. The applicant committed to upgrade existing valves, and, for the seal penetration lines, to install additional isolation valves.

During the period from February 14 through May 3, 1989, the applicant and staff held a series of telephone conferences discussing piping penetration isolation requirements. The staff agreed that the existing piping penetration arrangements were acceptable and agreed to prepare a safety evaluation documenting the results of the discussions. In the aforementioned May 12, 1989, letter, the applicant withdrew the commitments related to the subject penetrations and reaffirmed its position that the use of CLOCs at Watts Bar with a single isolation valve inside the containment is consistent with the conclusions of the staff's 1982 SER, which accepted the isolation system design. The sections that follow delineate the current design of the affected penetrations.

6.2.4.1 Use of Inside (Versus Outside) Containment Isolation Valve

For each penetration, the staff examined the relative effects on (a) containment atmosphere isolation reliability and (b) accident mitigation reliability. The staff concludes that the single isolation valve inside the containment is acceptable. Although the Standard Review Plan (Section 6.2.4,II.6.e) discusses a single isolation valve outside the containment, it states that an isolation valve will be acceptable if (a) it can be shown that the system reliability is greater with only one isolation valve in the line, (b) a single active failure can be accommodated with only one isolation valve in the line, and (c) the system is closed outside the containment. Since a single isolation valve inside the containment satisfies these conditions, the staff finds the use of a single isolation valve inside the containment to be acceptable.

6.2.4.2 Evaluation of a Closed System

A closed system is defined (reference Regulatory Guide 1.141 and ANSI N271-1976) as:

> A piping system which penetrates and is a closed system either inside or outside the containment. Under normal operating conditions or loss-of-coolant accident conditions for closed systems inside containment, the fluid in the system does not communicate directly with either primary coolant or containment atmosphere.

CLOCs are typically portions of ECC systems. For such systems, the system accident mitigation functional reliability may be improved by the omission of one of the two required containment isolation valves. The improvement in system reliability is considered more beneficial than the potential decrease in containment integrity. Such a piping configuration does qualify for acceptance under General Design Criterion 56 on an "other defined basis."

Regulatory Guide 1.141 endorses ANSI N271-1976 "Containment Isolation Provisions for Fluid Systems," which specifies the following criteria that a CLOC must satisfy:

(1) not communicate with the outside atmosphere

(2) be designed to containment temperature and pressure conditions

(3) meet Safety Class 2 (ESF) design requirements

(4) withstand LOCA conditions

(5) meet seismic Category I design criteria

(6) be protected against overpressure from thermal expansion

(7) be protected against a high-energy line break when needed for containment isolation

A comparison of the Watts Bar design-basis criteria for CLOCs specified in FSAR Section 6.2.4.1 with the regulatory criteria indicates that the applicant's design criteria are consistent with the regulatory criteria, except for the lack of protection against overpressure from thermal expansion of fluid trapped between the closed loop. The staff, therefore, independently confirmed that thermal expansion overpressure protection is provided. The staff found that one of the five systems, the hydrogen analyzer system, has since been provided with an additional (outboard) isolation valve. Another system, the upper head injection system, has been removed from the plant design. The other systems are provided with overpressure protection from thermal expansion by means of relief valves which discharge to a header leading to the pressurizer relief tank.

In 1990, the staff performed systems walkdowns and confirmed that the CLOC systems are located in controlled leakage areas and that branch, drain, and instrument lines attached to the closed systems are isolated by at least two isolation valves.

On this basis, the staff finds that the systems in question qualify as "closed."

6.2.4.3 Conclusion

Having determined that the systems in question are "closed loops outside containment," and that the original modifications (i.e., October 20, 1988, letter) would not provide significant additional containment isolation dependability, the staff reaffirms the conclusion of acceptability stated in the SER and SSERs 1 through 11. This resolves Outstanding Issue 23.

6.6 Inservice Inspection of Class 2 and 3 Components

In SSER 10, the staff stated that the issue of preservice inspection (PSI) was resolved for Unit 1. Subsequently, the staff discovered that a number of relief requests pertaining to PSI have not been properly addressed. The results of the review of these requests are given in Appendix Z to this SSER, as a supplement to Appendix Z originally published in SSER 10. The results do not change the staff's conclusion stated in SSER 10 regarding the PSI program. The staff's efforts were tracked by TAC M86037 and M86038.

In the SER, the staff stated that "The initial inservice inspection program has not been submitted by the applicant. The staff will evaluate the program after the applicable ASME Code edition and addenda can be determined based on Section 50.55a(b) of 10 CFR 50, but before the first refueling outage when inservice inspection begins." The staff initiated proposed License Condition 4 to track this effort.

In a letter of September 10, 1993, the applicant committed to submit the Watts Bar Inservice Inspection (ISI) Program within six months after receiving the operating license. This will allow the applicant to incorporate the code requirements in effect 12 months before the operating license is issued. On the basis of this commitment, the staff considers proposed License Condition 4 no longer required. The staff's evaluation of the ISI Program will be published before the beginning of the first refueling outage.

9 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.6 Condensate Storage Facilities

In Section 9.2.6 of the SER, the staff indicated that the two condensate storage tanks reserved 200,000 gallons of condensate for each unit's auxiliary feedwater (AFW) system. In FSAR Amendment 72, the applicant revised this reserved amount to 210,000 gallons. The basis for the storage capacity is not affected and this correction is made for clarification purposes only. This does not change any of the staff's conclusions reached in the SER or supplements related to the condensate storage facilities or the AFW system. The staff's effort was tracked by TAC M85037 and M85038.

9.5 Other Auxiliary Systems

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System

In the SER, the staff described various ways of filling the diesel engine fuel oil storage tank. One of these ways involved routing a hose from the delivery vehicle to the diesel generator (DG) tank manway openings located in the hallway area of the DG building. The staff found that this method was "acceptable provided that fire watches are stationed in these areas during the tank filling period." The staff stated that this requirement will be included in the Technical Specifications.

Since the issuance of the SER, the staff issued Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," which provides guidance to relocate fire protection requirements. As a result, the staff, by letter dated March 15, 1993, asked the applicant to either provide appropriate technical specifications and bases, or to justify relocating the fire-protection requirements to the Technical Requirements Manual (TRM). In a letter of April 20, 1993, the applicant responded to this request, stating that a specific statement in the Technical Specifications or equivalent document calling for a specified fire watch when filling the diesel fuel storage tanks via the manways was unnecessary.

Regardless of the alternate method used to refill the storage tanks (the method described above is considered the least desirable), the fire-protection systems in the DG buildings are designed to protect against a fuel oil fire. Should the fire-protection features, both active and passive, be made inoperable, then appropriate compensatory actions would be taken in accordance with the TRM and/or Fire Protection Report requirements on the fire-detection, carbon dioxide, sprinkler, and fire-barrier systems. Furthermore, procedural controls require that a fire watch be established anytime a fire door is breached (which would occur when routing a hose to the refilling location in the DG buildings). Therefore, a separate and distinct fire watch would not be required during the refilling operation. The fire door watch provides the necessary additional protection.

On this basis, the staff concludes that the fire watch as described in Section 9.5.4.2 of the SER is not required and that the alternative refilling method is acceptable. This effort was tracked by TAC M76742.

10 STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

10.2.1 Turbine Generator Design

In Section 10.2.1 of the SER, the staff described the three independent overspeed turbine trip systems. By FSAR Amendment 72, the applicant revised the description of these three independent trip systems which differs somewhat from that in the SER. The description that follows supersedes the description in the SER, but does not alter the staff's conclusions reached in the SER; therefore, the turbine overspeed protection system is still acceptable.

During normal operation (speed-load control), the overspeed protection controller (OPC), which is set at 103 percent of rated speed, will rapidly close the governor and intercepter valves in case of an overspeed condition. Rotational speed is then maintained below this trip setpoint by oscillating the intercept valves between the closed and partially open position until the reheater steam is dissipated. If the OPC control system is in the automatic mode, the governor will take over speed control and will maintain normal load at rated speed. However, if the OPC is in the manual mode (normally only at low power levels during startup), the turbine generator will coast down to turning gear operation.

If, for some reason, the OPC control system does not function and the turbine speed increases to 110 percent of rated speed, the mechanical overspeed mechanism will trip closed all steam valves (throttle, governor, reheat, stop and intercept valves) and prevent the turbine speed from exceeding 120 percent of rated speed. The unit will then coast down to turning gear operation.

In addition to these speed control systems discussed above, an independent electrical overspeed trip that is redundant to the mechanical trip at 110 percent is provided in the analog electrohydraulic (AEH) control system. If the turbine speed increases to approximately 111 percent of rated speed, all steam valves will be tripped closed. This trip will be actuated by a contact output from the AEH controller which energizes a trip solenoid in the auto stop oil lines. Again, during the overspeed condition, turbine speed will remain below 120 percent of rated speed and the unit will coast down to turning gear operation.

The staff's efforts were tracked by TAC M85037 and M85038.

10.3 Main Steam Supply System

10.3.1 Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)

In Section 10.3.1 of the SER, the staff stated that the main steam isolation valves (MSIVs) would close on "high-high" containment pressure, or high steam flow coincident with low steam generator pressure or low reactor coolant average temperature. With the update to the Eagle-21 process protection system as described in FSAR Amendment 72, the closing signals for the MSIVs now consist of "high-high" containment pressure, low steamline pressure, and high steamline pressure rate (decreasing). The MSIVs will close within 5 seconds after receipt of any one of these three signals. These are the same three signals used for MSIV isolation at the Sequoyah nuclear plant and approved by the staff in the Sequoyah Technical Specifications. However, the basis for acceptance of the main steam system in Section 10.3.1 of the SER did not include the initiation signals. The staff, therefore, concludes that the evaluation and conclusions in Section 10.3.1 of the SER are still valid and the main steam system remains acceptable.

The staff's efforts were tracked by TAC M85037 and M85038.

14 INITIAL TEST PROGRAM

In the SER and in SSERs 3, 5, 7, 9, and 10, the staff reported on its review of Chapter 14, "Initial Test Program," of the Watts Bar FSAR. By FSAR Amendment 69, the applicant completely revised Chapter 14. Hence, the staff's evaluation reported in the SER, SSER 3 (except where it resolves Confirmatory Issue 41), SSER 5, and SSER 9 are superseded by the evaluation given here. The evaluation in Chapter 14 of SSER 7 (regarding deletion of proposed License Condition 31) and SSER 10 (regarding deletion of the natural circulation test) are not affected.

The staff raised a number of concerns while reviewing Amendment 69 in accordance with the Standard Review Plan (NUREG-0800), and these were transmitted to the applicant by letter of July 14, 1992, as a request for additional information (RAI). The applicant responded in letters of January 13 and April 2, 1993. Then the applicant submitted Amendment 74 by letter of May 21, 1993, to reflect resolution of most of the staff's concerns. What follows is the staff's evaluation of FSAR Chapter 14, as revised through Amendment 74.

14.2 Preoperational Tests

14.2.1 Concerns in the Staff's RAI of July 14, 1992

The staff requested that FSAR Section 14.2.1, "Summary of Test Program and Objectives," be modified to identify preoperational tests (or portions thereof) which are intended to be conducted (or have their results approved) after fuel loading, to describe the intended plant conditions for those tests, and to provide appropriate justification for delaying the tests. The TVA response stated that tests summarized in FSAR Table 14.2-1 will be completed and results will be approved before commencing fuel load. TVA also stated that any necessary exceptions, and associated justifications and test conditions, will be approved by TVA and submitted to the NRC staff before fuel load. The staff finds this response and the change incorporated into FSAR Chapter 14 by Amendment 74 acceptable.

The staff requested modifications to the FSAR regarding "Organization and Staffing" to identify the organizational reporting chain for system test engineers and power ascension engineers. The TVA response stated that the system engineers will report to their supervisors who, in turn, report to the startup manager. The power ascension engineers will report to a supervisor who, in turn, reports to the plant manager. This change to "Organization and Staffing" was incorporated into FSAR, Amendment 74. The staff finds this response and the incorporated change to Chapter 14 acceptable, as it clearly details the reporting chain for the engineers.

The staff asked the applicant to define what test procedure revisions would constitute "changes to the intent." The TVA response clarified that changes to test methods, objectives, or acceptance criteria will be considered intent changes. The staff finds this response and the incorporated change to Chapter 14 by Amendment 74 acceptable, as TVA has appropriately clarified what test changes would be subject to the Joint Test Group's review.

The staff requested clarification of the "Startup and Test Organization" description in FSAR Section 14.2.4.2. The TVA response changed the "Organization and Staffing" descriptions of Section 14.2.2, clarifying the "Startup and Test Organization." The staff finds this response and the incorporated change to Chapter 14 by Amendment 74 acceptable as the startup and test organizations are clearly identified.

The staff requested that FSAR Section 14.2.5, "Review, Evaluation, and Approval of Test Results" be modified to state that results of completed tests will be evaluated by the appropriate personnel or groups. The staff also asked that Section 14.2.5 reflect that appropriate remedial actions, including retesting, will be taken if acceptance criteria are not satisfied before proceeding to the next major test phase. TVA revised Section 14.2.5 to incorporate the requested alterations. The staff finds this response and the incorporated change to FSAR by Amendment 74 acceptable.

The staff requested that FSAR Section 14.2.6, "Test Records," be changed or an explanation be provided to demonstrate agreement with the guidance contained in Regulatory Guide 1.16, "Reporting of Operating Information Technical Specifications." The TVA response incorporated a change into Section 14.2.6 which shows agreement with the Regulatory Guide 1.16 guidance. The staff finds the incorporated change to Chapter 14 acceptable.

The staff requested that additional information be provided in FSAR Section 14.2.7, "Conformance of Test Programs With Regulatory Guides," to address compliance with the regulatory guides listed in Section 14.2.2 below. The applicant's response is evaluated there.

The staff requested that FSAR Section 14.2.11 be modified to make allowance, only when no reasonable alternative exists, for the potential need to proceed into low-power testing up to the 25-percent power testing plateau prior to completing testing of structures, systems, and components which are relied upon to prevent or mitigate the consequences of postulated accidents. The TVA response revised Section 14.2.11 in accordance with this request, except that the reactor coolant system (RCS) flow measurement test was proposed to be performed at the 50-percent power level consistent with Westinghouse guidance. In a subsequent conference call on March 9, 1993, TVA indicated that an RCS flow measurement test would be performed at the zero power level. TVA agreed to modify its response to describe this test. This response and the change incorporated by Amendment 74 are acceptable.

The staff 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. TVA responded that these test summaries would be revised later to provide the requested information. This item remains open pending TVA's submittal and the staff's review of the modified power ascension phase test summaries. This will continue to be tracked by TAC M82644 and M82645.

The staff requested that the "Preoperational Test Summaries" and "Power Ascension Test Program" be modified to specify the bases for determining acceptable system and component performance. The TVA response stated that the individual test summaries of FSAR Tables 14.2–1 and 14.2–2 will be revised to include appropriate references to acceptance criteria source documents. The preoperational test summaries have been modified in Amendment 74 to include more precise acceptance criteria. The staff reviewed the revised preoperational test summaries to assess the inclusion of references to acceptance criteria. This review found that the changes made to the individual preoperational test summaries do provide a consistent level of detail and traceability to the source documents for acceptance criteria. The staff finds the modified preoperational test summaries acceptable. However, the staff will review the power ascension test program individual test abstracts to assess the technical adequacy of acceptance criteria for demonstrating the system functional requirements that are important to safety, as described in the source documents, when these revised test summaries have been submitted. This item remains open with respect to the power ascension program. This will continue to be tracked by TAC M82644 and M82645.

The staff requested that additional details be submitted in the individual test abstracts to allow assessment of the adequacy of the Watts Bar FSAR commitments. The TVA response indicates that TVA considers that the individual test abstracts, as revised, do contain sufficient detail for this review. The staff reviewed the revised preoperational test summaries to assess the inclusion of sufficient detail for consistency with previous preoperational test commitments. The reviewed acceptance criteria were found to be acceptable and consistent with previous preoperational testing commitments. This response and the change incorporated in Amendment 74 to Chapter 14 are acceptable.

FSAR Section 14.2.7, as revised by Amendment 74, takes exception to Regulatory Guide 1.68, Revision 2, Appendix A, Item 1.j.(22). Item 1.j.(22) involves instrumentation that can be used to monitor plant conditions during the course of postulated accidents. The preoperational test status (FSAR Table 14.2–1) of each of the instrumentation systems discussed in Item 1.j.(22) was reviewed to determine the adequacy of testing. On the basis of this review, two instrument tests appear to be incomplete:

(1) Containment Wide-Range Pressure Indicators

This system is described in FSAR Sections 6.2.4, 7.3.1, 7.3.2, and Table 7.5–1. The preoperational test abstracts that reference this system are 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 verifies proper calibration of the instrumentation and annunciation of containment pressure. However, the referenced acceptance criteria (FSAR Section 5.2.7) do not describe containment pressure detection or indication. The test method described in Sheet 83, although verifying proper containment isolation between phase A and phase B, and referencing appropriate acceptance criteria (FSAR Section 6.2.4), does not address functionality of control room instrumentation. Proper control room indication and alarm function tests and acceptance criteria for the containment pressure instrumentation should be incorporated into the appropriate preoperational test abstract.

(2) Containment Water Level Monitors

This system is described in FSAR Sections 6.3.2, 6.3.4 and Table 7.5–1. The preoperational test abstracts that reference this system are Sheet 22, "Safety Injection System Test Summary," and Sheet 83, "Containment Isolation System Test Summary." The test method described in Sheet 22 and the associated acceptance criteria verify automatic operation of the RHR pump containment sump suction valves swapover and refers to FSAR Section 6.3. The test methods described in Sheet 83 and the associated acceptance criteria verify proper operation of all valves designed to isolate on receipt of a phase A or phase B containment isolation signal. It is noted that the Sheet 59 test method verifies proper calibration and annunciation of abnormal rise in the reactor building floor and equipment drain sump and the acceptance criteria refer to FSAR Section 5.2.7. However, none of these test abstracts specifically references the containment water level monitors. Proper control room indication and alarm function tests and acceptance criteria for the containment water level instrumentation should be incorporated into the appropriate preoperational test abstract.

In summary, FSAR Table 14.2–1 test abstracts adequately address testing of the reactor vessel water level monitors, containment high-range radiation detection devices, and the containment humidity monitors, but do not adequately describe testing of the containment wide-range pressure indicators and containment water level monitors. The staff will continue to track these issues by TAC M82644 and M82645.

14.2.2 Conformance With Regulatory Guides

RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," April 1982

The TVA response described changes to FSAR Sections 9.3.1.4 and 14.2.7.7 to address RG 1.68.3, and noted an exception to Regulatory Position (RP) C.8 for not testing a sudden loss of instrument air pressure. Correspondence of February 28, 1984, between R.C. Lewis and H.G. Parris, records the NRC acceptance of this exception. The staff notes that this response appears to be in conflict with TVA correspondence of July 12, 1990 in response to NRC Generic Letter (GL) 88–14. Enclosure 2 of the TVA correspondence indicates 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." This item will remain open pending clarification by the applicant of its commitment to perform rapid and gradual loss-of-air testing. This will continue to be tracked by TAC M82644 and M82645.

TVA is taking exception to RP C.8 not testing non-safety-related loads for response to a loss of system pressure as part of the auxiliary control air system preoperational test because this system does not supply non-safety-related loads. However, TVA has stated that individual non-safety-related components will be tested on a component basis to verify proper response to a loss-of-air condition. The staff finds this response and the incorporated change to FSAR Chapter 14 by Amendment 74 acceptable.

TVA is taking exception to RP C.11 by not demonstrating operability of compressed-air-system loads under increased pressure conditions as TVA considers that adequate safety features have been designed in the system, as described in FSAR Section 9.3.1.3, to prevent such occurrences. The staff finds this response does not offer sufficient information to adequately determine whether the proposed change to Chapter 14 is acceptable. This item, therefore, remains open pending submittal of additional information addressing the adequacy of safety features provided for the compressed-air system to ensure that excessive pressure conditions will not occur. This will continue to be tracked by TAC M82644 and M82645.

RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977

The TVA response stated that the potential for chlorine to pose a hazard to the main control room operators due to spills or transportation incidents is evaluated in FSAR Section 6.4.4.2. This analysis concludes that no credible chlorine hazard exists so that RG 1.95 is not applicable. The staff finds this response and the change incorporated by Amendment 74 acceptable.

RG 1.139, "Guidance for Residual Heat Removal," for comment, May 1978

The TVA response stated that this RG will not be used for development or conduct of the initial test program. In Section 5.4.3 and Chapter 14 of SSER 10 the staff agreed with the applicant that there is no need for the applicant to perform any natural circulation test, based on the applicability of Diablo Canyon test results. An exception to RG 1.139 incorporating the justification noted above was not in Amendment 74. This item remains open pending incorporation in a future FSAR amendment. This will continue to be tracked by TAC M82644 and M82645.

RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 1, October 1979

The TVA response stated that FSAR Section 14.2.7.10 will be added to indicate full compliance with RG 1.140 for testing of the non-safety-related ventilation exhaust filtration systems. The staff finds this response and the change incorporated by Amendment 74 acceptable.

RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants," Revision 2, August 1978

The staff requested that alternate approaches or exceptions to RG 1.68, Revision 2, be clearly described in the appropriate FSAR section (e.g., 14.2.7.4.a). The TVA response stated that Section 14.2.7 will be revised to clearly identify and differentiate all exceptions and alternate approaches to RG 1.68. Changes to Sections 14.2.7 and 14.2.7.4 were submitted showing the specific changes intended in the initial test program description. The staff finds this response and the change incorporated in Amendment 74 to FSAR Chapter 14 acceptable.

The staff requested that items from RG 1.68, Appendix A, that were not addressed by TVA be either (1) addressed as exceptions, (2) added to test summaries, or (3) explained where existing test summaries addressed them. The TVA response was adequate for most of the individual items in Appendix A to RG 1.68. However, the applicant indicated that it plans to abandon the failed fuel detector (FFD). Thus a test summary for the FFD, as recommended by RG 1.68 Appendix A 1.j.(12), was not provided. The staff finds this response and the change to Chapter 14 incomplete, and requires additional information regarding the system or components that will be used to satisfy the functions of the FFD. This item remains open. TVA should provide the technical basis for deleting the system and the alternate means that will be utilized to monitor fuel cladding integrity. This will continue to be tracked by TAC M82644 and M82645.

The staff requested that FSAR Sections 14.2.4, 14.2.5, 14.2.7, and the individual test summaries described in Tables 14.2-1 and 14.2-2 be modified to address the "Acceptance Test" process and applicable tests. The TVA response provided changes to the affected sections to address the "Acceptance Test" process, and new test abstracts were added for non-safety-related systems that will be tested under this process. The staff finds this response and the changes incorporated by Amendment 74 acceptable.

The staff requested that Section 14.2.7.4.a(3) be modified to commit to either (1) conduct in-place preoperational testing of reactor coolant system and main steam system power-operated relief valves, (2) provide vendor data that substantiate proper performance of these valves under design conditions, or (3) reference an analysis that determines the proper performance of these valves under design conditions (e.g., reference analysis similar to that provided in Q/R 413.07). The TVA response stated that the intent of the exception taken in Section 14.2.7.4.a(3) is in regard to in-place capacity testing and does not involve an exception to RG 1.68; therefore, this exception has been deleted. The valve performance will be demonstrated by other than in-place capacity testing. The staff finds this response and the change incorporated by Amendment 74 acceptable.

The staff requested that Section 14.2.7.4.a(4) be modified for the exception taken for RG 1.68, Appendix A, Item 1.g.2 to address the concerns of Branch Technical Position (BTP) PSB-1 regarding verification of adequate undervoltage protection and the validity of analytical models for the safety-related buses. The TVA response for FSAR Section 14.2.7.4.a(2) describes changes to address the concerns of BTP PSB-1. These changes state that emergency loads will not be tested with minimum and maximum design voltage available, but that emergency loads will be tested to demonstrate satisfactory starting and operating characteristics with the power supply voltage within the design operating range. Transformer taps will be adjusted to vary voltage levels from no-load to full-load conditions and tests will be performed to record vital bus voltage conditions, including the lowest analyzed voltage. This information will be compared to engineering voltage calculations to validate the analytical models used. The staff finds this response and the change incorporated by Amendment 74 acceptable.

The staff requested that Section 14.2.7.4.a(5) be modified to clearly indicate TVA's intention as to whether a containment design overpressure structural test would be conducted for Unit 2. The TVA response stated that the intent of the exception was to clarify that TVA did not plan to repeat the structural integrity test that was conducted on the Unit 1 containment, and that this position is not an exception to RG 1.68; thus Section 14.2.7.4.a(5) is not required and it will be deleted. The staff finds this response and the change incorporated by Amendment 74 to delete Section 14.2.7.4.a(5) as an acceptable exception. The FSAR now requires that the containment integrity structural test be performed only for Unit 2, as the Unit 1 test has been completed.

The staff requested that retesting be performed as necessary after modifications are performed that could impact the acceptability of completed test results. TVA responded that Section 14.2.7 was revised to delete exceptions for RG 1.68, Appendix A, Items 1.i.1, 1.m.1, 1.m.4, and 1.o.1, and will clearly indicate that any previously performed test will be evaluated to verify compliance with regulatory guidance and test acceptance criteria. TVA will evaluate any deviations from regulatory requirements, unacceptable test results, and post-test modifications to determine appropriate retest requirements. The staff finds this response and the change incorporated by Amendment 74 acceptable.

The staff requested that Section 14.2.7, Item 4.a(21) be modified to state that power reactivity coefficients will be determined to be in accordance with design at the recommended power levels. The TVA response refers to the staff's previous agreement to delete the 25-percent, 50-percent, and 75-percent power level testing as documented in SSER 3, and states that an exception is also taken for the 100-percent power level test based on an NRC-accepted exception granted at a recently licensed plant of similar design. The changes to Section 14.2.7 that describe the reasons for this exception are:

(1) The power coefficient is not directly measured but is inferred.

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

(3) The measurement also was previously deleted for other similar plants based on the results of performing the test at sister plants, thus demonstrating the ability to analytically predict this design characteristic.

The staff's review of SSER 3 revealed that the exception for 25-percent, 50-percent, and 75-percent power was granted for the reasons stated above. However, startup test programs for similar units have included performance of core reactivity balance testing at low power levels as part of justification for the exception to power coefficient testing. Therefore, TVA should modify the justification provided in 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 appropriate test abstract that performs core reactivity balance testing at low power levels should be provided or referenced. This item remains open, and will continue to be tracked by TAC M82644 and M82645.

The staff requested that FSAR Sections 14.2.7 and 14.2.11 be modified to state that approved preoperational and power ascension test procedures be available for staff review approximately 60 days preceding their intended use and not less than 60 days preceding the scheduled fuel loading date. TVA responded that FSAR Section 14.2.7.4.b will be revised to state that approved preoperational test procedures for satisfying FSAR testing commitments will be made available for NRC inspection approximately 30 days preceding their intended use, and that power ascension tests will be made available to the NRC 60 days preceding fuel load, in accordance with SSER 3. Section 14.2.11 will be revised similarly. The staff's review of SSER 3 confirms that this exception has been approved. The staff finds this response and the change incorporated by Amendment 74 acceptable.

RG 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," Revision 1, July 1978

The staff requested that the test abstract for demonstration of remote shutdown capability be modified to include information regarding RG 1.68.2 (RP C.2.c, C.3, and C.4.d) or for TVA to provide justification if not so modified. The TVA response stated that the test abstracts, "Integrated Hot Functional Testing" and "Shutdown From Outside the Control Room Coincident With Loss of Off Site Power," will be revised to reflect the guidance provided by RG 1.68.2. These test abstracts have been modified by Amendment 74 to reflect this guidance. The staff finds this response and the change incorporated by Amendment 74 to Chapter 14 acceptable.

RG 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," Revision 1, September 1975

The staff requested modifications to appropriate individual test summaries or to Section 14.2.7 to address the testing of ECCS flow from high-pressure and low-pressure pumps under hot operating conditions. The TVA response stated that the "Safety Injection System" test summary includes these tests and that the test abstract was modified to expand the information provided on these tests. The staff finds this response and the change incorporated by Amendment 74 acceptable.

RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977

The staff requested that the appropriate test abstracts be modified or that Section 14.2.7 be modified to justify the exception for RG 1.108 (RP C.2.(6), C.2.(9), and C.2.(4)). TVA responded that the diesel generator preoperational test summary will be modified to include information showing conformance with this regulatory guidance. The test abstracts have been modified in Amendment 74 to reflect this guidance. The staff finds this response and the change incorporated by Amendment 74 acceptable.

14.2.3 Conclusion

FSAR Chapter 14, as revised by Amendment 74, is generally adequate with the exception of the following items that TVA needs to address: (1) clarify commitments to perform rapid and gradual loss of air testing, (2) submit additional information addressing the adequacy of safety features provided which ensure that excessive pressure conditions will not occur in the compressed air system, (3) incorporate an exception to Regulatory Guide 1.139 for not performing the natural circulation test, (4) justify abandoning the failed fuel detector, (5) justify not performing a core reactivity balance, (6) submit power ascension test summaries to describe initial conditions and acceptance criteria, and (7) perform tests to demonstrate the functionality of the containment wide-range pressure indicators and containment water-level monitors. The staff will report on its evaluation of these open issues in a future SSER.

For areas not identified above, the staff finds the Watts Bar preoperational test program to be in accordance with the applicable review criteria. As mentioned in the preceding material, open issues will be tracked by TAC M82644 and M82645.

15 ACCIDENT ANALYSIS

15.3 Limiting Accidents

15.3.1 Loss–of Coolant Accident

By letter dated January 9, 1993, the applicant proposed to amend the Watts Bar FSAR to reflect a reanalysis of the small-break loss-of-coolant accident analysis (SBLOCA). The analysis of record peak cladding temperature (PCT), as reported in the FSAR by Amendment 63, was 1446.1 °F (786 °C). The SBLOCA reanalysis was necessary as a result of changes to the emergency core cooling system (ECCS) model that resulted in a change to the PCT of more than 50 °F (27.8 °C) for the limiting accident transient (in accordance with requirement of 10 CFR 50.46). Also factored into the reanalysis were reductions in the auxiliary feedwater (AFW) flow rates to represent the split flow path, and reductions in the safety injection (SI) flow rates to provide increased margin in the surveillance testing. TVA also incorporated the changes, already delineated in the FSAR and approved in Section 4.4.3.2 of this SER supplement (SSER 12), plant modifications from downflow barrel/baffel configuration to upflow barrel/baffle configuration. The evaluation that follows was transmitted to the applicant in a letter of May 17, 1993, finding the reanalysis and accompanying draft FSAR change pages acceptable. The applicant will update the FSAR in accordance with the accepted draft page changes.

Westinghouse Electric Corporation performed the SBLOCA reanalysis (4-inch and 3-inch breaks) for the applicant using its currently approved emergency core cooling system (ECCS) evaluation model, updated to included the changes and corrections outlined previously in the the applicant's letter dated July 22, 1991. The SBLOCA analysis was performed using the approved Westinghouse ECCS Small-Break Evaluation Model and NOTRUMP, an approved digital computer code developed to determine the RCS response to design-basis SBLOCAs and to address the NRC concerns expressed in NUREG–0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants." The changes affecting the SBLOCA analysis are (1) fuel rod model consistency with the fuel design codes that are used to establish initial conditions for LOCA analysis; (2) the SBLOCA rod internal pressure initial condition assumption; and (3) the NOTRUMP computer code solution convergence reliability for SBLOCA analysis.

In addition to these changes, the SBLOCA reanalysis also includes revised AFW flow rates (1050 gpm) to the steam generators, and more conservative (lower) SI flow rates (FSAR Figure 15.3–8n). The applicant indicated that the revised AFW flow rates more accurately represent the split flow path from the discharge of the two motor-driven AFW pumps and the one turbine-driven AFW pump. The applicant reduced the SI flow rates in the reanalysis to provide increased margin and flexibility for plant operation, maintenance, and surveillance activities.

The applicant determined that the 3-inch break is most limiting with a PCT of 2089 °F (1143 °C). This is an increase of 643 °F (357 °C) from the previous analysis of record PCT of 1446.1 °F (786 °C) reported by Amendment 63 to the FSAR. Of the increase, 130 °F (72 °C) is attributed to the ECCS model changes and the remaining 513 °F (285 °C) is attributed to the combined effects of the modified AFW and SI flows.

The applicant has reanalyzed its SBLOCA analysis for WBN Units 1 and 2 as required by 10 CFR 50.46. The staff finds the applicant's analysis to be acceptable in that (1) approved methods were used for the analysis, (2) the PCT of 2089 °F (1143 °C) (the new analysis of record) is below the acceptance criteria established by 10 CFR 50.46 of 2200 °F (1204 °C). The staff's efforts were tracked by TAC M85488 and M85489.

The large-break LOCA (LBLOCA) still remains the most limiting event, with an analysis of record PCT of 2126 °F (1163 °C) (as reported by Amendment 63 to the FSAR). As stated in a TVA letter of March 17, 1993, this is estimated to increase to 2129 °F (1165 °C). The SBLOCA reanalysis did not affect the PCT for the LBLOCA analysis. The current LBLOCA PCT of 2126 °F was analyzed with the upflow configuration and remains more limiting, bounding the new SBLOCA analysis.

15.3.6 Anticipated Transients Without Scram (ATWSs)

Status of Salem ATWS Event Issues

In SSER 11, the staff published a list of all documents delineating its efforts on Generic Letter (GL) 83–28 issues. The following paragraph pertains to one of them, Items 3.1.3 and 3.2.3, "Post-Maintenance Testing in Technical Specifications That Could Degrade Safety." The staff's efforts were tracked by TAC M77138, M77139 and M76742.

In letters of November 7, 1983 and January 17, 1986, the applicant submitted information on GL 83–28 items. The applicant stated that there would be no postmaintenance test requirements in the Technical Specifications (currently still under development) for either the reactor trip system or other safetyrelated components which could degrade safety. In a letter of April 20, 1993, the applicant stated that the Technical Specifications as they stood today were complete and correct with regard to reactor trip breaker testing, and affirmed that none would degrade, rather than enhance, safety. The staff has no more concerns on the Technical Specifications in this regard.

15.4 Radiological Consequences of Accidents

15.4.3 Steam Generator Tube Rupture

Proposed License Condition 41, regarding steam generator tube rupture (SGTR) was introduced in SSER 3. In SSER 5, the staff specified in detail the information needed from the applicant to fully resolve this proposed license condition. The five items that need to be addressed are summarized as follows:

(1) Demonstrate that the operator action times assumed in the analysis are realistic.

(2) Perform a site-specific SGTR radiation offsite consequence analysis.

(3) Evaluate the adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

(4) List systems, components, and instrumentation which are credited for accident mitigation in the plant-specific SGTR emergency operating procedure(s) (EOP(s)).

(5) Survey the designs of the primary and balance-of-plant systems to determine the compatibility with the bounding analysis in WCAP-10698.

In a letter of April 13, 1993, the applicant submitted results of the plant-specific margin-to-overfill SGTR analysis for Watts Bar using the Westinghouse LOFTTR2 computer code. The SGTR analysis assumptions used by the applicant were consistent with those of the reference plant analysis. The applicant has responded acceptably to Items 3, 4, and 5, described in detail in SSER 5. The staff reviewed the radiological assessment of the applicant's April 13, 1993, submittal, and concludes that the staff's original evaluation (in the SER) still bounds the radiological consequences done in accordance with WCAP-10698. This resolves Item 2.

The staff is continuing its review on the applicant's submittals regarding Item 1. Hence, proposed License Condition 41 remains unresolved.

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, F. G. Pagano to TVA, requesting additional information on containment high-range monitor.
February 3, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of January 27, 1993.
February 16, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the solid radioactive waste process control program.
February 18, 1993	Letter, P. S. Tam to M. O. Medford (TVA), informing of staff observation of Thermo-Lag tests.
February 22, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), granting authorization to use an alternative to the construction and installation requirements of Section III, Subsection NC/ND of the ASME Code.
March 2, 1993	Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming mechanical integrated design inspection.
March 2, 1993	Letter, P. S. Tam to M. O. Medford (TVA), agreeing that the submittal date for the individual plant evaluation on external events (IPEEE) can be changed to May 1995.
March 10, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of March 3, 1993.
March 11, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation on pressurized thermal shock.
March 12, 1993	Meeting summary by E. W. Merschoff, meeting of March 4, 1993, to discuss status of ongoing and planned activities.
March 15, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the draft Technical Specifications and Technical Requirements Manual.
March 18, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation on TVA's proposed fixes regarding station blackout.
March 25, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on pipe break spectra as a result of approval of use of leak-before-break technology.
April 2, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting copy of Proof and Review version of the Watts Bar Unit 1 Technical Specifications, and requesting comments.
April 7, 1993	Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming site review by E. Lee regarding use of Eagle–21.
April 13, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of April 7, 1993.
April 15, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on safety and relief valve testing.
April 16, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting copies of Watts Bar Safety Evaluation Report Supplement 11.

April 25, 1993	Meeting summary by E. W. Merschoff, enforcement conference of March 22, 1993, to address employee discrimination cases that have been filed with the U.S. Department of Labor.
April 28, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation on use of leak-before-break technology regarding the pressurizer surge line.
April 28, 1993	Meeting summary by E. W. Merschoff, meeting of April 21, 1993, to discuss current issues of concern and the status of ongoing and planned activities.
April 28, 1993	Meeting summary by A. F. Gibson, meeting to address preoperational test concerns.
May 3, 1993	Letter, P. S. Tam to N. J. Liparulo (Westinghouse), agreeing that Topical Report WCAP–13575 is proprietary.
May 5, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on Thermo-Lag test program.
May 17, 1993	Letter, P. S. Tam to M. O. Medford (TVA), accepting the new analysis of record for the small-break LOCA peak cladding temperature.
May 17, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of May 12, 1993.
May 19, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting a schedule to reanalyze the large-break LOCA per 10 CFR 50.46 requirement.
May 21, 1993	Meeting summary by E. W. Merschoff, meeting of May 13, 1993, to discuss ongoing and planned activities at Watts Bar.
June 2, 1993	Letter, S. A. Varga to M. O. Medford (TVA), transmitting Inspection Report 93–202 on the mechanical integrated design inspection.
June 9, 1993	Letter, F. J. Congel to C. S. Wingo (Federal Emergency Management Agency), transmitting the staff's evaluation of the State of Tennessee evacuation time estimate.
June 10, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the seismic response spectra used to analyze stability of the HVAC duct supports.
June 14, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the Eagle–21 system.
June 28, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of June 23, 1993.
June 29, 1993	Letter, S. A. Varga to M. O. Medford (TVA), closing open items in previous integrated design inspections, such as those documented in Inspection Reports 92–201 and 91–201.
June 29, 1993	Meeting summary by P. S. Tam, management meeting of June 23, 1993.
June 29, 1993	Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming site visit to address issues related to associated circuits in the fire protection program.
July 1, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting a copy of a videotape purportedly taken by a Fox TV reporter using a concealed camera.
July 1, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information of the Emergency Preparedness Plan emergency action levels (EALs).
July 2, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation on TVA's stability analysis for the underground barriers for the essential raw cooling water system.

July 13, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on Outstanding Issue 24, regarding main steamline break outside containment.
July 21, 1993	Meeting summary by P. S. Tam, routine licensing status meeting of July 20, 1993.

TVA Letters

August 12, 1982	Letter, L. M. Mills to E. Adensam (NRC), providing information on NUREG–0737, Item II.B.1, "Reactor Coolant System Vents."
January 17, 1983	Letter, L. M. Mills to NRC responding to informal questions regarding FSAR Table 3.2–2.
November 7, 1983	Letter, L. M. Mills to NRC, responding to Generic Letter 83–28.
January 17, 1986	Letter, J. A. Domer to NRC, submitting additional information on response to Generic Letter 83–28.
June 15, 1987	Letter, R. L. Gridley to NRC, advising of forthcoming submittal on removal of upper head injection system.
August 21, 1987	Letter, R. L. Gridley to NRC, submitting information on compliance with ASME Section III.
August 31, 1987	Letter, R. L. Gridley to NRC, submitting information regarding upflow modification.
December 24, 1987	Letter, R. L. Gridley to NRC, submitting information regarding compliance with ASME Section III.
March 31, 1988	Letter, R. L. Gridley to NRC, submitting draft change pages to the FSAR.
July 6, 1988	Letter, R. L. Gridley to NRC, submitting information regarding authorized nuclear inspectors.
July 28, 1988	Letter, R. L. Gridley to NRC, submitting draft FSAR pages to reflect upflow conversion and removal of upper head injection.
October 13, 1988	Letter, R. L. Gridley to NRC, submitting request to use an alternative to certain requirement of ASME Section XI.
October 20, 1988	Letter, S. A. White to NRC, submitting information on the proposed corrective action program on containment isolation.
March 21, 1989	Letter, R. L. Gridley to NRC, submitting information regarding essential raw cooling water system mortar-lined piping.
April 20, 1989	Letter C. H. Fox to NRC, submitting information regarding radiographic film reevaluation of the refueling water storage tank.
May 5, 1989	Letter, O. D. Kingsley to NRC, submitting revised Corporate Nuclear Performance Plan.
May 12, 1989	Letter, O. D. Kingsley to NRC, submitting additional information on containment isolation.
August 21, 1989	Letter, M. J. Ray to NRC, addressing certain aspects of Generic Letter 89–04 (regarding inservice testing).
September 11, 1989	Letter, M. J. Ray to NRC, submitting information regarding closed loops outside containment.
September 21, 1989	Letter, M. J. Ray to NRC, submitting request for approval to use an alternative to meet certain requirements of ASME Section III.

November 21, 1989	Letter, R. H. Shell to NRC, submitting request for approval to use an alternative to meet a requirement of ASME Section III.
November 21, 1989	Letter, R. H. Shell to NRC, submitting request for approval to use ASME Code Case N–460.
December 11, 1989	Letter, M. J. Ray to NRC, submitting request for approval to use an alternative to meet a requirement of ASME Section III.
April 5, 1990	Letter, E. G. Wallace to NRC, submitting request for approval to use an alternative to meet a requirement of ASME Section III.
April 13, 1990	Letter, E. G. Wallace to NRC, submitting request for approval to use an alternative to meet a requirement of ASME Section III.
July 12, 1990	Letter, E. G. Wallace to NRC, responding to Generic Letter 88–14 regarding instrument air supply system.
January 9, 1993	Letter, W. J. Museler to NRC, transmitting information on reanalysis of the small-break LOCA.
January 13, 1993	Letter, W. J. Museler to NRC, submitting response to NRC questions on FSAR Chapter 14.
February 4, 1993	Letter, W. J. Museler to NRC, transmitting record plans for the QA records corrective action program.
February 4, 1993	Letter, W. J. Museler to NRC, transmitting Revision 4 to the Vendor Information Corrective Action Program.
February 9, 1993	Letter, M. J. Burzynski, transmitting fourth annual report of the Employee Concerns Special Program.
February 10, 1993	Letter, W. J. Museler to NRC, transmitting response to request for additional information on fire barrier endurance testing program.
February 11, 1993	Letter, M. J. Burzynski to NRC, transmitting security personnel training and qualification plan.
February 11, 1993	Letter, W. J. Museler to NRC, submitting information on severe accident mitigation design alternatives.
February 12, 1993	Letter, M. J. Burzynski to NRC, transmitting additional information on the Radiological Emergency Plan.
February 17, 1993	Letter, W. J. Museler to NRC, transmitting additional information on the environmental qualification of electrical equipment, including discussion of mechanical equipment qualification.
February 23, 1993	Letter, W. J. Museler to NRC, submitting information on QA record plan on cables.
March 1, 1993	Letter, W. J. Museler to NRC, responding to request for additional information on fire protection program.
March 3, 1993	Letter, M. J. Burzynski to NRC, submitting information on TVA Nuclear QA Program.
March 8, 1993	Letter, W. J. Museler to NRC, transmitting Westinghouse Topical Report WCAP–11627 to support reactor internals upflow conversion.
March 16, 1993	Letter, W. J. Museler to NRC, submitting additional information on FSAR Chapter 15, as revised by Amendment 71.
March 17, 1993	Letter, W. J. Museler to NRC, notifying of recent changes in the ECCS evaluation model.

March 19, 1993	Letter, W. J. Museler to NRC, transmitting first Core Operating Limits Report (COLR).
March 19, 1993	Letter, W. J. Museler to NRC, transmitting proposed revisions to the Technical Requirements Manual.
March 19, 1993	Letter, W. J. Museler to NRC, transmitting additional information on reactor coolant system vents.
March 26, 1993	Letter, M. J. Burzynski to NRC, transmitting revised physical security plan.
March 26, 1993	Letter, W. J. Museler to NRC, submitting additional information regarding use of leak-before-break technology on the pressurizer surge line.
March 30, 1993	Letter, W. J. Museler to NRC, responding to the staff's questions on use of linear elastic analysis methods.
April 1, 1993	Letter, W. J. Museler to NRC, submitting updated response to Bulletin 79–28.
April 2, 1993	Letter, W. J. Museler to NRC, transmitting proposed changes to Chapter 14 of the FSAR.
April 2, 1993	Letter, W. J. Museler to NRC, informing of complete implementation of the Master Fuse Special Program.
April 8, 1993	Letter, M. J. Burzynski to NRC, submitting a revision to the TVA Quality Assurance Program.
April 8, 1993	Letter, W. J. Museler to NRC, submitting additional information on stability analysis for underground barriers for the essential raw cooling water piping.
April 8, 1993	Letter, W. J. Museler to NRC, submitting additional information on U-bolts.
April 13, 1993	Letter, W. J. Museler to NRC, submitting information on the postulated steam generator tube rupture accident.
April 15, 1993	Letter, W. J. Museler to NRC, submitting additional information on pipe whip.
April 16, 1993	Letter, W. J. Museler to NRC, transmitting clarification on Chapter 14 of the FSAR, as revised by Amendment 69.
April 16, 1993	Letter, W. J. Museler to NRC, transmitting response to Generic Letter 92–08 on Thermo-Lag fire-retardant materials.
April 16, 1993	Letter, W. J. Museler to NRC, transmitting information on indications of cracking in accumulator tank cladding.
April 20, 1993	Letter, W. J. Museler to NRC, responding to request for additional information on the draft Technical Specifications.
April 23, 1993	Letter, M. J. Burzynski to NRC, transmitting Tennessee Multi-Jurisdictional Radiological Emergency Response Plan.
April 28, 1993	Letter, W. J. Museler to NRC, responding to the staff's safety evaluation on station blackout coping measures.
April 30, 1993	Letter, W. J. Museler to NRC, commenting on the Proof and Review version of the Technical Specifications.
May 21, 1993	Letter, W. J. Museler to NRC, transmitting Amendment 73 of the FSAR.
May 21, 1993	Letter, W. J. Museler to NRC, transmitting Amendment 74 of the FSAR.

May 28, 1993	Letter, W. J. Museler to NRC, responding to request for additional information on the draft Technical Specifications.
June 1, 1993	Letter, W. J. Museler to NRC, transmitting additional information on material traceability.
June 2, 1993	Letter, W. J. Museler to NRC, submitting additional information on the Fire Protection Program.
June 4, 1993	Letter, W. J. Museler to NRC, commenting on the Proof and Review version of the Technical Specifications.
June 5, 1993	Letter, W. J. Museler to NRC, submitting information on severe accident mitigation design alternatives.
June 15, 1993	Letter, W. J. Museler to NRC, submitting additional information on control rod drive mechanism seismic qualification.
June 18, 1993	Letter, W. J. Museler to NRC, responding to the staff's request for additional information of March 15, 1993, regarding the draft Technical Specifications.
June 21, 1993	Letter, W. J. Museler to NRC, submitting additional information regarding stability of U-bolts.
June 30, 1993	Letter, W. J. Museler to NRC, submitting Amendment 75 of the FSAR.
June 30, 1993	Letter, W. J. Museler to NRC, submitting additional information on Chapter 15 of the FSAR as revised by Amendment 71.
June 30, 1993	Letter, W. J. Museler to NRC, submitting additional information on testing of Thermo-Lag fire barriers.
July 2, 1993	Letter, W. J. Museler to NRC, submitting additional information regarding hydrostatic testing of the reactor coolant system.
July 8, 1993	Letter, W. J. Museler to NRC, transmitting Revision 5 of the Q-List Corrective Action Program.
July 9, 1993	Letter, W. J. Museler to NRC, submitting final report on Thermo-Lag fire barrier testing.
July 9, 1993	Letter, W. J. Museler to NRC, submitting information on the radiological waste process control program.
July 12, 1993	Letter, B. S. Schofield to NRC, transmitting information on the October 1993 radiological emergency exercise plan.
September 10, 1993	Letter, W. J. Museler to NRC, committing to submit ISI program within six months after fuel loading.
September 13, 1993	Letter, W. J. Museler to NRC, submitting additional information on containment pressure due to postulated main steamline break event.

APPENDIX B

BIBLIOGRAPHY

Industry Codes and Standards

American Society for Mechanical Engineers, "Boiler and Pressure Vessel Code."

American National Standards Institute, ANSI B31.1, "Corrosion Control for ANSI B31.1 Power Piping SystemsB31 Guide."

___, ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems."

Electric Power Research Institute, EPRI NP-3596-SR, Revision 1, "PICEP: Pipe Crack Evaluation Program," December 1987.

Institute of Electrical and Electronics Engineers, IEEE Standard 344-1975, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

U.S. Nuclear Regulatory Commission

CLI-80-21, Commission Order and Memorandum, May 23, 1980.

NUREG-0737, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981.

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

NUREG-1431, "Standard Technical SpecificationsWestinghouse Plants," Vols. 1-3, September 1992.

Westinghouse Electric Corporation

WCAP-11256, Supplement 1, "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Pressurizer Surge Lines of South Texas Projects Units 1 and 2," April 1987.

WCAP-11627, "Upflow Conversion Safety Evaluation Report, Watts Bar Units 1 and 2," 1993.

APPENDIX E

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APPENDIX Z

SUPPLEMENTAL SAFETY EVALUATION: PRESERVICE INSPECTION RELIEF REQUESTS

Appendix Z was originally published in SSER 10. Since the publication of SSER 10, the staff discovered that a number of preservice inspection relief requests submitted by TVA have not been addressed. This supplement presents the staff's review of those requests.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION PROPOSED ALTERNATIVES TO ASME SECTION III REQUIREMENTS TENNESSEE VALLEY AUTHORITY WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 DOCKET NOS. 50-390 AND 50-391

1 INTRODUCTION

The staff's safety evaluation and authorization of reliefs were originally published in SSER 10. Since the publication of SSER 10, the staff discovered that a number of preservice inspection relief requests submitted by TVA have not been addressed. This supplement presents the staff's review of those requests.

10 CFR 50.55a, "Codes and Standards," permits the NRC staff to authorize alternatives to portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, on the basis of either "The proposed alternatives would provide an acceptable level of quality and safety" (10 CFR 50.55a(a)(3)(i)), or "Compliance with the specified requirements of this Code would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety" (10 CFR 50.55a(a)(3)(ii)).

During construction of Watts Bar Nuclear Plant, Units 1 and 2, an NRC team reviewed several proposed alternatives to ASME Code Section III requirements. The staff, at that time, found that the technical bases submitted by the applicant to support the proposed alternatives were acceptable. However, although these evaluations were documented in NRC Inspection Reports 50-390, 391/89-04 (August 9, 1989), 50-390, 391/90-02 (March 15, 1990), and 50-390, 391/90-04 (May 17, 1990), proper authorization of the proposed alternatives was never received. This supplement documents another technical review of the applicant's proposed alternatives to ASME Code Section III construction requirements, and gives valid authorizations for these, where appropriate.

2 APPLICANT'S PROPOSED ALTERNATIVES

A. Later Editions of ASME Code Section III

The construction code of record for Watts Bar, Units 1 and 2, is the ASME Code Section III, 1971 Edition through the Summer 1973 Addenda. In letters to the NRC of August 21, 1987, and July 6, 1988, TVA submitted proposed alternatives, and corresponding technical evaluations, to the code of record for specific welding-related issues. All of the proposed alternatives concern the use of later editions of ASME Code Section III, and involve 21 welding process specifications and 5 general construction specifications. The proposed alternatives are listed in detail in NRC Inspection Report 50-390, 391/89-04 (August 9, 1989). The staff reviewed the TVA evaluations and, pursuant to 10 CFR 50.55a(a)(3)(i), determined that the proposed alternatives produce an acceptable level of quality and safety. Therefore, the NRC authorizes the use of the applicant's proposed alternative.

B. Alternative Acceptance Criteria for Welds in Containment Sleeves

During independent evaluations of piping weld radiographs at Watts Bar, Units 1 and 2, several ASME-rejectable indications concerning weld and base metal imperfections, and many instances of poor film or technique quality were discovered. On the basis of these reviews, TVA initiated corrective actions to include repair of unacceptable indications and new radiography to correct technique or film quality deficiencies.

However, two inaccessible welds were identified as having radiographic indications that exceeded the acceptance criteria of ASME Code Section III. These welds are located in containment sleeves around the residual heat removal (RHR) pump suction lines and are embedded in reinforced concrete behind the stainless steel containment sump liner wall. These two butt welds attach a spool piece extension to a 24-inch-diameter stainless steel flued-head fitting and a 24-inch-diameter carbon steel pipe that forms the containment penetration for the RHR pump suction line. Radiographic interpretation revealed that:

(1) Weld 1-074B-D045-01A has an area of incomplete fusion less than 1/2 inch in length, and another area of aligned rounded indications and incomplete fusion with a combined length of less than 3/4 inch.

(2) Weld 1-074-D045-08A has one incomplete fusion indication approximately 3/16 inch in length.

In correspondence to the NRC dated August 19, 1988, and May 5, 1989, TVA proposed, as an alternative to ASME Code Section III acceptance standards, using the flaw evaluation rules in ASME Code Section XI for disposition of these welds. TVA used the evaluation criteria listed in ASME Code Section XI, Subarticle IWB-3600, to determine that the flaws located in these two welds will not propagate and cause failure during the design life of the plant. The staff concluded that TVA had adequately demonstrated, through conservative analysis and flaw size criteria, that the integrity of the two welds will not be affected by these existing fabrication imperfections and that an acceptable level of quality and safety is achieved by using the Section XI flaw acceptance criteria. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC authorizes use of the proposed alternative for the RHR containment sleeve penetration welds.

C. Repairs to the Refueling Water Storage Tanks

In accordance with a previous commitment to the NRC, TVA evaluated a vendor weld of the refueling water storage tanks (RWSTs). These tanks were fabricated on site by Pittsburgh-Des Moines Steel Corp. (PDM) in 1978. During the review, TVA identified PDM radiographs that did not meet ASME Code Section III requirements, either because of inadequate radiographic technique or imperfections in the weld zone. TVA determined that several welds had to be repaired.

However, PDM no longer possessed a valid ASME Code Section III Certificate of Authorization. Therefore, in letters of December 24, 1987, and April 20, 1989, TVA submitted a proposed alternative to Section III consisting of a plan to allow PDM to perform the repair work on an ASME Section III stamped component under TVA's Section XI Repair and Replacement Program. The plan included the use of the applicant's Quality Assurance (QA) Program, which had controls similar to those of ASME Code Section III, ensuring that fabrication and installation, examination, testing, authorized nuclear inspection (ANI), and authorized nuclear inservice inspection (ANII) requirements would be met.

The staff concludes that this approach to making repairs to RWSTs produces an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(i), use of ASME Code Section XI for repairs to the RWSTs is authorized.

D. Control of Work Activities per ASME Code Sections III and XI

At the request of the applicant, a meeting was held on June 26, 1987, to discuss the applicant's commitment to the ASME Code Section III for welding activities at Watts Bar Unit 1. The staff position regarding Section III welding was stated as follows:

(1) All work shall be performed to the code of record for construction, ASME Code Section III, 1971 Edition through Summer 1973 Addenda.

(2) The applicant shall review any repairs/modifications that were performed following closure of the N-5 packages. If cases were identified of work being performed in accordance with ASME Code Section XI, or by an organization that was not an ASME Code Section III certificate holder, the applicant shall identify these as exceptions to the code of record, and request approval from the NRC staff for proposed alternatives as prescribed by 10 CFR 50.55a (a)(3).

(3) Discrepant N-5 packages could be completed by supplementing the current N-5 packages. This method is consistent with Code Interpretation III-1-83-175, and was acceptable to the staff.

In letters of October 13, 1988, and May 10, 1991, the applicant submitted its review of welded repairs/modifications that were performed following closure of the N-5 data packages. The review included comparisons of ASME Code Sections III and XI programmatic requirements. The applicant identified those areas representing exceptions to the code of record and requested approval of alternatives proposed in lieu of the Section III requirements.

All of the repairs/modifications completed subsequent to the N-5 packages were performed under the applicant's ASME Code Section XI Repair and Replacement Program. This program included the use of the applicant's Quality Assurance (QA) Program and had controls similar to those of ASME Code Section III, ensuring that fabrication and installation, examination, testing, authorized nuclear inspection (ANI), and authorized nuclear inservice inspection (ANII) requirements would be met. However, in the area of pressure testing, ASME Code Section XI requirements are less strict than those of Section III. The applicant had committed to perform the pressure tests to the Section III requirements, and supplement the N-5 data reports in accordance with Section III.

The staff concludes that, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative provides an acceptable level of quality and safety. Therefore, the applicant is authorized to use the proposed alternative for welding-related activities subsequent to N-5 data reports.

E. Hydrotest of the Essential Raw Cooling Water System

During review and comparison of ASME Code Sections III and XI requirements, the applicant identified work issues representing exceptions to the code of record (see Section D above). A commitment was made to perform pressure testing in accordance with the Section III criteria for those components repaired/modified to the applicant's Section XI program. It was determined that portions of the essential raw cooling water (ERCW) system mortar-lined piping were inaccessible, as the piping outside of the building is buried, in part, under concrete missile shields.

Excavation of the buried piping to perform a Section III visual inspection during hydrotest would entail the removal, and subsequent replacement, of structural concrete and the outer protective coating on the pipe. In addition, seven valves would have to be removed and blind flanges installed to ensure a leakproof pressure boundary. In order to remove these valves, the entire ERCW train would have to be drained, which would affect many plant systems.

In a letter of March 21, 1989, the applicant requested that a Section III hydrostatic test be performed on the system as installed and, for the inaccessible areas, the pressure held for one hour to ascertain that the welds do not leak. All accessible welds (those inside the building) would be visually inspected when the required test pressure was reached.

The staff reviewed the applicant's request and determined that, pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with Section III hydrotest requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety. The applicant is, therefore, authorized to use a 1-hour hold time in lieu of a visual examination to determine leakage for inaccessible portions of the ERCW piping.

F. Quality Assurance Requirements for Materials' Suppliers

In a letter to the NRC of December 11, 1989, the applicant requested that the staff review an alternative to the requirements of ASME Code Section III, Paragraph NA-3451(a), entitled "Scope of Responsibility for Quality Assurance." The alternative was proposed to resolve a deficiency that had been identified during a review of the Watts Bar Heat Code Traceability Corrective Action Plan.

The corrective action for this deficiency involved upgrading of American Society for Testing and Materials (ASTM) certified pipe materials to meet the requirements for quality assurance (QA) outlined by the ASME Code Section III, Paragraph NA-3451(a). Paragraph NA-3451(a) states that the installer (the applicant in this case) shall be responsible for surveying and qualifying the quality systems programs of its material suppliers and manufacturers, including subcontracted services and nondestructive examinations.

During a heat code traceability review, TVA reported that the products of five heats of ASTM-A106 pipe material were supplied by a particular vendor to Watts Bar in 1974. However, the supplier of the pipe material was not audited and certified as an approved vendor by TVA until October 15, 1976. Therefore, the five heats of material were supplied by a vendor who was not certified at the time.

The applicant requested that, as an alternative to the requirements of ASME Code Section III, Paragraph NA-3451(a), the ASTM specification be evaluated to compare and assess QA attributes for equivalency to those cited in ASME Code Section III, Paragraphs NC/ND-2600. From the evaluation, the applicant determined that the ASTM specifications met the minimum QA requirements of ASME Code Section III, NC/ND-2600.

The staff reviewed the applicant's evaluation (attached to a TVA letter of September 11, 1989), and available QA records, and concluded that, pursuant to 10 CFR 50.55a (a)(3)(i), the QA attributes found in the ASTM specification represent an acceptable level of quality and safety for the pipe material as supplied. Therefore, the applicant is authorized to use the proposed alternative.

G. Acceptance Criteria for Containment Penetration Piping Welds

During construction of Watts Bar Unit 1, the applicant determined that the manufacturer of many of the containment penetration assemblies had opted to use system hydrostatic tests in lieu of separate component hydrostatic tests, as allowed by ASME Code Section III. The applicant did not discover that the manufacturer had elected to use this option until after the system hydrostatic tests had been completed. Consequently, although the required system hydrostatic tests were performed, and the penetration assemblies had been exposed to the required ASME Code Section III test pressures, no provisions were made to visually examine the manufacturer's shop welds during the test.

In letters to the NRC of November 21, 1989, and April 13, 1990, the applicant requested that the staff approve alternative acceptance criteria for the manufacturer's containment penetration assembly welds. ASME Code Section III requires that all safety class welds be visually examined during the hydrostatic test to detect evidence of leakage. In order to make the manufacturer's welds accessible for examination, several "windows" must be cut in the guard piping that surrounds the process pipes. The welds would be viewed by remote methods (mirrors or fiberoptic devices) only, then the "windows" in the guard piping would have to be repaired. The applicant stated that this type of inspection process is difficult to execute to ensure both completeness and accurate interpretation. Additionally, there is potential for damaging the process piping during the cutting or repair of the guard pipes.

The applicant stated that a "use-as-is" disposition of these containment penetrations is acceptable for the following reasons:

(1) The welds were fabricated and examined in accordance with ASME Code Section III requirements, with authorized nuclear inspection (ANI) involvement at the manufacturing plant.

(2) An acceptable hydrostatic or pneumatic test to ASME Code Section III requirements was performed on the field welds when the penetration assemblies to the plant piping systems were installed.

(3) All pressure boundary piping containing longitudinal weld seams that were used by the manufacturer to fabricate the assemblies was hydrostatically tested by the material supplier in accordance with the ASME material specification.

(4) Many of the vendor welds in question are in close proximity to the field welds visually examined during the system hydrostatic tests, and, therefore, it is reasonable to assume that any fabrication weld leakage would have been observed at that time.

(5) The manufacturer's welds were examined by radiography and found acceptable by the vendor in accordance with ASME Code Section III requirements.

On the basis of the discussion above, the staff concludes that, pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with the ASME Code Section III visual examination during hydrostatic pressure testing for these containment penetration assembly welds would present an unusual hardship without a compensating increase in the level of safety or quality of the welds. Therefore, the applicant is authorized to use the proposed alternative described above.

H. Pneumatic Test Pressure for the Control Air System

In correspondence of April 5, 1990, the applicant asked the staff to approve an alternative to the ASME Code Section III pneumatic test pressure criterion for Watts Bar Unit 1 control air system.

Portions of the control air system are ASME Code Class 3, and were fabricated and installed to the requirements of ASME Code Section III. These portions were pneumatically tested in accordance with Article ND-6000 of ASME Code Section III; however, the test pressure was insufficient due to an incorrect entry on the flow diagram for this system. The design pressure specified in the FSAR is 115 psig, but was listed in error as 105 psig on the flow diagram. The maximum operating pressure for the control air system is 105 psig.

Article ND-6000 of ASME Code Section III requires a test pressure of 125 percent of the system design pressure. The control air system should have been tested at 143.75 psig (125 percent of 115 psig). However, because the incorrect design pressure value was on the flow diagram, the system was tested to a pressure of 131.25 psig (125 percent of 105 psig). The result is that the control air system was tested to only 91.3 percent of the required test pressure.

The applicant stated that the pneumatic test, as performed, is adequate for the following reasons:

(1) The original pneumatic test was performed in accordance with the requirements of ND-6000, except for the required test pressure.

(2) The test pressure used was 125 percent of the normal maximum operating pressure.

(3) Testing at a higher pressure would not result in a significant increase in the stress levels of system components.

(4) The consequences of minor leakage would not be significant since there is sufficient capacity in this system to compensate for small leakage without affecting normal or safety functions.

(5) An increase in operating pressure in excess of 115 psig will be controlled by the safety/relief valves.

Performance of a second pneumatic test at the correct pressure would require installing scaffolding and removing instrumentation at many locations on the control air system. This would result in unusual difficulty and hardship without a compensating increase in the quality and safety of the control air system over that provided by the applicant's proposed alternative to test the system at 125 percent of normal system operating pressure. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), it has been determined that the proposed alternative provides an acceptable level of quality and safety and is, therefore, authorized for use in lieu of ASME Code Section III requirements.

I. Alternative Acceptance Criteria for RWST Vortex Drain Piping Welds

In a letter of September 21, 1989, the applicant submitted an alternative proposal to the NRC concerning ASME Code Section III, Class 2 weld acceptance requirements. The proposal requested approval to use ASME Code Section XI flaw evaluation techniques in lieu of the weld acceptance criteria listed in ASME Code Section III.

During review of vendor weld radiographs of the refueling water storage tanks (RWSTs) at Watts Bar Units 1 and 2, the applicant identified numerous disparities from ASME Code Section III examination requirements. The disparities included weld imperfections exceeding Section III acceptance criteria and improper radiographic techniques. The applicant has subsequently performed additional radiography and, where necessary, made repairs to the RWST welds to comply with the construction code of record (ASME Code Section III, 1974 Edition including Winter 1975 Addenda) for these Code Class 2 tanks.

However, two welds were identified that were not readily accessible for additional radiography or repairs; the vortex nozzle assembly welds located in the bottom of the RWSTs for both Units 1 and 2 contain weld imperfections and inadequate radiography per ASME Code Section III. Each vortex nozzle assembly consists of a cone subassembly formed from four segments of 5/16-inch-thick plate fabricated with vertical seam welds, and a pipe subassembly made of seam-welded 3/8-inch rolled plate. The cone and pipe subassemblies were joined with a full-penetration groove weld. The applicant has identified the following discrepancies in the two vortex nozzle assembly welds:

(1) The seam welds in the cone subassemblies for both units were not radiographed per ASME Code Section III, Subarticle NC–5280 requirements.

(2) Radiographic techniques for the Unit 1 circumferential weld (attaching the cone to the pipe) and the pipe seam weld do not comply with ASME Code Section III requirements for film quality and coverage.

(3) Weld defects that do not meet the acceptance criteria of ASME Code Section III have been identified in vortex nozzle assemblies for both units:

   (a) In Unit 1, a 3-inch-long lack of fusion exists in one of the cone subassembly's seam welds; this defect was identified in the radiograph of the circumferential weld attaching the cone subassembly to the pipe subassembly.

   (b) In Unit 2, a 3/8-inch unacceptable slag inclusion, and two linear indications each approximately 1/4-inch long, exist in the circumferential weld attaching the cone to the pipe. In addition, unacceptable slag, approximately 1/4 inch in length, was identified in one of the cone subassembly's seam welds, and six 1/8-inch-long indications exist adjacent to the circumferential weld.

The RWST vortex drain cone subassembly, and a portion of the pipe subassembly, including the attaching circumferential weld, are embedded in reinforced concrete. Except for the portion of pipe extending beyond the concrete surface into the pipe tunnel below the tank, it would be extremely difficult to expose the welds for additional radiography or repair. Because the subject welds were inaccessible, TVA performed fracture mechanics analyses in accordance with the method described in ASME Code Section XI, Subarticle IWB–3640 and Appendix C, and Code Case N–436.

The analyses indicated that the cone subassembly can withstand a longitudinal through-wall flaw of 48.9 inches or less, and still maintain structural integrity. It was shown that the circumferential cone-to-pipe assembly weld can withstand a through-wall flaw for as much as 70 percent of the circumference and still maintain structural integrity. The fracture mechanics analyses are documented in the September 21, 1989, letter mentioned above. In addition, various fabrication checklists and nondestructive examination reports (hydrostatic tests and surface examinations) indicate that the welds are of acceptable quality. On the basis of these examinations and the fracture mechanics analyses, the applicant concluded that the flaws will not lead to failure of the nozzle assemblies.

The staff has determined that TVA adequately demonstrated, through conservative analysis, that the integrity of the drain line vortex welds will not be compromised by existing weld flaws during the design life of the RWSTs. The staff has further concluded that, pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with the radiographic and weld acceptance requirements listed in ASME Code Section III would present an unusual difficulty without a compensating increase in the level of quality or safety. The applicant's proposed alternative of using the flaw evaluation criteria of ASME Code Section XI is authorized for the RWST drain line vortex welds at Watts Bar Units 1 and 2.

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11. ABSTRACT (200 words or less)

Supplement No. 12 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of (1) additional information submitted by the applicant since Supplement No. 11 was issued, and (2) matters that the staff had under review when Supplement No. 11 was issued.

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