NUREG–0847
Supplement No. 10

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 1992

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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), and Supplement No. 9 (June 1992) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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APPENDICES

OBE	operating basis earthquake
OCDM	Offsite Dose Calculation Manual
OGC	Office of the General Counsel
PAM	postaccident monitoring
PORV	pilot-operated relief valve
PSI	preservice inspection
PSP	physical security plan
PWR	pressurized-water reactor
QA	quality assurance
RADCON	radiological controls
RAI	request for additional information
RCS	reactor coolant system
RHR	residual heat removal
RO	reactor operator
RPV	reactor pressure vessel
RSB	Reactor Systems Branch
RTD	resistance temperature detector
RVLIS	reactor vessel level instrumentation system
RWST	refueling water storage tank
SAT	systems approach to training
SER	Safety Evaluation Report
SG	steam generator
SIS	safety injection system
SMM	subcooling margin monitor
SP	special program
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSE	safe shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SSI	soil-structure interaction
SwRI	Southwest Research Institute
TAC	technical assignment control
TI	temporary instruction
TLD	thermoluminescent dosimeters
TS	Technical Specifications
TVA	Tennessee Valley Authority
UT	ultrasonic
WBNPP	Watts Bar Nuclear Performance Plan
WGDS	waste gas disposal system
WISP	Workload Information and Scheduling System
WOG	Westinghouse Owners Group

ABBREVIATIONS

ADGB	associated diesel generator building
AFW	auxiliary feedwater
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BISI	bypassed and inoperable status indication
BIT	boron injection tank
BOL	beginning of life
BTP	branch technical position
CAP	corrective action program
CAQ	condition adverse to quality
CFR	Code of Federal Regulations
CNPP	Corporate Nuclear Performance Plan
CRDM	control rod drive mechanism
CS	containment spray
CSB	Containment Systems Branch
CVCS	chemical and volume control system
DAC	distance amplitude correction
DRD	direct reading dosimeter
ECCS	emergency core cooling system
EOP	emergency operating procedure
ERCW	essential raw cooling water
ERG	emergency response guideline
FSAR	Final Safety Analysis Report
GDC	general design criterion
GL	generic letter
HPSI	high-pressure safety injection
ICC	inadequate core cooling
ICCM	inadequate core cooling monitor
ICTC	incore thermocouple
IE	Office of Inspection and Enforcement
ISI	inservice inspection
LOCA	loss-of-coolant accident
NEMA	National Electrical Manufacturers Association
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), 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), and Supplement No. 9 (SSER 9, June 1992). As of this date, the staff has completed review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 68.

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 from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The 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. Results of this reevaluation will be published in a future SSER.

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. Appendices C, D, F, G, H through P, and R through Y are not changed by this SSER. Appendix Q, previously published in SSER 8, has been supplemented. Appendix Z was added in this SSER.

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

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1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. 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 is conveyed in the staff's summary of the monthly meeting regarding licensing status.

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
(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

[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 relevant documents are filed. Documents associated with each TAC number can be listed by the NRC document control system, NUDOCS/AD.

Issue	Status	Section
(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	Under review (SSER 6)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Opened (SSER 6)	3.7.2.12
(h) Category 1(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Resolved (SSER 9)	3.8
(k) Update FSAR per 12/18/90 letter	Resolved (SSER 8)	3.7
(20) Mechanical systems and components (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

Issue	Status	Section
(22) Removal of upper head injection system (TAC M77195)	Resolved (SSER 7)	6.3.1
(23) Containment isolation using closed systems (TAC M63597)	Awaiting submittal (SSER 7)	6.2.4
(24) Main steamline break outside containment (TAC M63632)	Awaiting submittal (SSER 7)	15.4.2
(25) Health Physics Program (TAC M63647)	Resolved (SSER 10)	12
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TAC M77550)	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. 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 is conveyed in the staff's summary of the monthly meeting regarding licensing status.

Issue	Status	Section
(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5) Upgrading ERCW system to seismic Category I (TAC M63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC M63618)	Resolved (SSER 5)	3.2.1

Issue	Status	Section
(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

Issue	Status	Section
(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
(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 information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

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)	Updated (SSER 5)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TAC M77132, M77133)	Resolved (SSER 10)	4.4.8
(4) Inservice Inspection Program (TAC M76881)	Updated (SSER 10)	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

Proposed Condition	Status	Section
(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 IE 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 (II.B.3) (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
(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)	Awaiting submittal (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

Proposed Condition	Status	Section
(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)	Updated (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 5)	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)

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 (dated July 1987), and NUREG-1232, Vol. 4 (dated 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 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)

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 November 1993.
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); to come.

(2) <u>Cable Tray and Tray Supports (TAC R00516)</u>

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by September 1993.
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); to come.

(3) <u>Design Baseline and Verification Program (TAC M63594)</u>

Program review status:	Complete: Inspection Report 50-390, 391/89-12 (November 20, 1989); NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by October 1993.
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); to come.

(4) <u>Electrical Conduit and Conduit Support (TAC R00508)</u>

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 September 1993.
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, 391/92-09 (June 29, 1992); 50-390, 391/92-201 (September 21, 1992); to come.

(5) Electrical Issues (TAC M74502)

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 October 1993.
NRC inspections:	Inspection Report 50-390, 391/90-30 (February 25, 1991); 50-390, 391/92-22 (September 18, 1992); to come.

(6) Equipment Seismic Qualification (TAC M71919)

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 September 1993.
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); to come.

(7) Fire Protection (TAC M63648)

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 August 1993.
NRC inspections:	To come.

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

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 October 1993.
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); to come.

(9) Heat Code Traceability (TAC M71920)

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)

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 August 1993.

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); to come.

(11) Instrument Lines (TAC M71918)

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 November 1993.

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 preoperational test program per Regulatory Guide 1.68, Revision 2. (See the staff's evaluation of FSAR Chapter 14 in a future SSER.)

(13) Quality Assurance Records (TAC M71923)

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).

Implementation status: Full implementation expected by September 1993.

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); to come.

(14) Q-List (TAC M63590)

Program review status: Complete: 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.

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

(15) Replacement Items Program (TAC M71922)

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 January 1993.

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); to come.

(16) Seismic Analysis (TAC R00514)

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)

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 (no program review).

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

NRC audits: Memorandum (publicly available), T. M. Cheng to P. S. Tam, January 23, 1992; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; letter, P. S. Tam (NRC) to M. O. Medford (TVA), May 26, 1992; to come.

(17) Vendor Information Program (TAC M71921)

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

Implementation status: Full implementation expected by March 1993.

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

(18) Welding (TAC M72106)

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: Full implementation expected by September 1992.

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); to come.

1.13.2 Special Programs

(1) Concrete Quality (TAC M63596)

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)

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 July 1993.

NRC inspections: To come.

(3) Detailed Control Room Design Review (TAC M63655)

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

Implementation status: Full implementation expected by August 1993.

NRC inspections: To come.

(4) Environmental Qualification Program (TAC M63591)

Program review status: NUREG-1232, Vol. 4; review in progress, results will be published in Section 3.11 of a future SSER.

Implementation status: Full implementation expected by July 1993.

NRC inspections: To come.

(5) Master Fuse List (TAC M76973)

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: Full implementation expected by September 1992.

NRC inspections:	Inspection Report 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); to come.

(6) <u>Mechanical Equipment Qualification (TAC M76974)</u>

Program review status:	NUREG-1232, Vol. 4; to come.
Implementation status:	Full implementation expected by July 1993.
NRC inspections:	To come.

(7) <u>Microbiologically Induced Corrosion (TAC M63650)</u>

Program review status:	Complete: NUREG-1232, Vol. 4; Appendix Q of SSER 8; Appendix Q of SSER 10.
Implementation status:	Full implementation expected by March 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); to come.

(8) <u>Moderate Energy Line Break Flooding (TAC M63595)</u>

Program review status:	NUREG-1232, Vol. 4; to come.
Implementation status:	Full implementation expected by October 1993.
NRC inspections:	To come.

(9) <u>Radiation Monitoring Program (TAC M76975)</u>

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 July 1993.
NRC inspections:	To come.

(10) <u>Soil Liquefaction (TAC M77548)</u>

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 to come.
NRC inspections:	Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990); audit report by L. B. Marsh (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, 1992; to come.

(11) Use-as-Is CAQs (TAC M77549)

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 to come.
NRC inspections:	Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); to come.

4 REACTOR

4.2 Fuel System Design

4.2.3 Mechanical Performance

In a letter of July 11, 1991, the staff issued a request for additional information (RAI) regarding two issues that arose from its review of the Watts Bar FSAR Amendment 65. The applicant responded to the staff's RAI in a letter of August 12, 1991. The staff reviewed the applicant's response and has reached the following conclusions:

(1) Use of SRSS, Top of Page 4.2-29

In this section of the FSAR, the applicant discusses the design loading conditions for the reactor vessel internals. The staff questioned the applicant's proposed use of a square root of the sum of the squares (SRSS) combination for safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) loads. In its response, the applicant stated that the proposed use of SRSS is supported by a Westinghouse generic analysis for a four-loop plant which is documented in topical report WNEP-7702, entitled, "Generic Stress Report 4-Loop Standard Reactor Core Support Structures Structural/Fatigue Analysis", June 1977, Westinghouse Proprietary Class 1.

The staff has no record of WNEP-7702 being submitted as a basis for using SRSS methodology for combining SSE and LOCA loads. However, in regard to the combination of service loads, in Standard Review Plan Section 3.9.3, the staff states, "The appropriate method of combination of these loads shall be in accordance with NUREG-0484, 'Methodology for Combining Dynamic Loads'." NUREG-0484 allows the use of SRSS methodology for SSE and LOCA load combinations. Therefore, the staff finds the change described in Amendment 65 acceptable provided that the applicant uses SRSS in accordance with the guidelines of NUREG-0484.

(2) CRDM Operability, Top of Page 4.2-34

In this section of the FSAR, the applicant discusses the seismic analysis of the control rod drive mechanisms (CRDMs). Amendment 65 deleted the last part of the first sentence on page 4.2-34 which stated that the seismic analysis should confirm the ability of the CRDMs to trip when subjected to seismic disturbances. In its RAI, the staff questioned how the applicant intended to verify the operability of the CRDMs under seismic disturbances. The applicant responded that the analysis referred to on FSAR page 4.2-34 is directed at the pressure boundary qualification of the components which make up the CRDMs, and was not intended to be used to confirm CRDM operability. During a subsequent discussion, the applicant indicated that the proposed change was the result of a Westinghouse review which determined that the original FSAR statement was in error.

In the August 12, 1991, response and in subsequent discussions, the applicant described the methods used to confirm CRDM operability, and proposed to revise the FSAR to include additional discussion on verification of CRDM operability.

The staff agrees that CRDM operability cannot be verified by the type of analysis described in this section of the FSAR and, therefore, the applicant's proposed change in Amendment 65 is acceptable. However, the staff has not yet reviewed all of the methods noted by the applicant for demonstrating its CRDM operability during and following seismic events. Therefore, the staff will review the applicant's methodology for seismic qualification of the CRDMs in a future SSER. This effort will be tracked by TAC M84249 and M84250.

4.4 Thermal Hydraulic Design

4.4.8 Instrumentation for Detecting Inadequate Core Cooling (TMI-2 Item II.F.2)

In the SER, the staff stated that its review of the applicant's inadequate core cooling (ICC) instrumentation was incomplete and unacceptable. The staff tracked its continued efforts on the Watts Bar ICC instrumentation review by proposed License Condition 3.

In a letter of January 24, 1992, the applicant proposed a revised response that superseded its previous responses concerning the issue, with the exception of the Westinghouse summary report of December 1980, "Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) for Monitoring Inadequate Core Cooling." That report was submitted as an attachment to the Item II.F.2 response in the applicant's letter of August 12, 1982. The Westinghouse summary report is still applicable to Watts Bar for the described equipment, with the exception of the processing and display electronics that are being replaced by Westinghouse's Inadequate Core Cooling Monitor-86 (ICCM-86).

The applicant has also modified the FSAR by Amendment 69 to reflect the ICC instrumentation design described in its January 24, 1992, letter.

4.4.8.1 Description of the ICCI

The Watts Bar ICC system consists of three instrumentation subsystems: (1) reactor vessel level instrumentation system (RVLIS), (2) incore thermocouple (ICTC) monitoring system; and subcooling margin monitor (SMM). The applicant submitted its evaluation with respect to conformance to NUREG-0737 in Table 1, Enclosure 1, of its January 24, 1992, submittal for these three subsystems.

Reactor Vessel Level Instrumentation System (RVLIS)

The RVLIS comprises three parts: a Westinghouse differential pressure measurement system, the processing electronics in the Westinghouse ICCM cabinet, and the Westinghouse output plasma display device supplied with the ICCM cabinet. The RVLIS differential pressure measuring system includes three differential pressure transmitters per division which are connected via sealed reference legs to (1) a spare penetration near the top of the reactor vessel upper head, (2) an existing bottom-mounted instrumentation guide tube at the seal table, and (3) a tap in two of the reactor coolant system (RCS) hot legs. The processing electronics is a Westinghouse ICCM microprocessor-based system. Each redundant division is housed in a separate cabinet in the auxiliary instrument room and is powered by a separate Class 1E battery-backed power source. The inputs from the differential pressure transmitters are compensated with inputs from the capillary line strap-on resistance temperature detector (RTD), RCS wide-range pressure, RCS wide-range hot-leg temperature, and reactor coolant pump status. The display system is a Westinghouse microprocessor-based system comprising three physical components: the plasma display, the electronics package, and the keypad located near the plasma display.

The RVLIS is not used as a primary indicator of an ICC condition, but is used to diagnose an approach to an ICC condition when subcooling margin has been lost and ICTC temperatures are still lower than temperatures at which ICC is assumed to exist.

The generic Westinghouse RVLIS has been reviewed and found acceptable by the staff and the evaluation was published in NUREG/CR-2628 entitled, "Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurements."

Incore Thermocouple (ICTC) Monitoring System

The ICTC system comprises sensors, cables, special connectors, processing electronics, and primary/backup (ICCM) and additional (computer) display devices. There are 65 ICTCs in conjunction with RCS wide-range temperatures, to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. The ICTC sensors are 65 Type K thermocouples that have been assigned to two redundant divisions [33 postaccident monitoring (PAM I) and 32 PAM II]. There are at least four thermocouples of each division per quadrant. Because of the particular thermocouple hardline cables that are routed through each Conoseal, the thermocouple cables do not meet minimum separation criterion until they exit the reactor cavity biological shield wall area. The Westinghouse microprocessor-based ICCM system is used to process the ICTC signals, using temperature compensation obtained from the RTDs located in the reference junction box. The ICCM provides the compensated ICTC temperatures to both the primary/backup display devices (the Westinghouse plasma displays) and, through isolation devices, to the additional display device (the P2500 plant computer printer or recorder).

The primary/backup display devices are the fully qualified Westinghouse ICCM microprocessor-based plasma displays which provide the operator the ICTC map page displaying a spatially oriented core map of real-time ICTC temperature, ICC quadrant summary page displaying the real-time minimum, average, and maximum temperatures for each quadrant in a core map outline, individual ICTC quadrant page providing a list of real-time ICTC temperatures by quadrant location and value, and auctioneered high core quadrant average temperature trend page.

The real-time values are updated approximately every 2 seconds. The trend displays are updated approximately every 20 seconds. The temperature indications range from 200 °F (93 °C) to 2,300 °F (1260 °C).

ICTC temperatures provide the primary indication of the existence of an ICC condition. This requires that the operators take immediate action.

The staff reviewed the applicant's ICTC system and found it acceptable since it meets the Item II.F.2, Attachment 1 guidance except for the non-separation between redundant divisions in the vicinity of the reactor vessel head, which has been previously approved by the staff for other Westinghouse plants.

Subcooling Margin Monitor (SMM)

The SMM derives the primary coolant's margin to saturation, an indicator of approach-to-boiling conditions, from various temperature and pressure inputs. These inputs include RCS wide-range pressure, RCS wide-range hot-leg temperature, and ICTC temperature. The algorithms reside in the microprocessor memory of the ICCM (primary display driver with fully qualified backup), a mass memory disc unit in the plant computer (additional display device driver). The margin to saturation is derived in the Westinghouse ICCM system for output to the plasma displays as well as qualified meters, and is also derived in the Westinghouse P2500 computer for output to a printer or recorder.

The subcooling margin displays available to the operator include subcooling margin trend page, heatup limit curve page, cooldown limit curve page, and subcooling diagnostics page. Subcooling margin is indicated continuously on separate qualified redundant digital meters on the control board. Subcooling margin is used by the operators as a primary indicator of the approach to, or the existence of, saturated conditions in the reactor coolant system.

4.4.8.2 Conclusion

The staff has reviewed the applicant's January 24, 1992, submittal, which supersedes previous responses (listed in the SER) concerning the guidance of NUREG-0737 on ICC instrumentation. The staff has also reviewed appropriate sections of the FSAR, as modified by Amendment 69 (this effort was tracked by TAC M82644 and M82645). The staff concludes the following:

(1) The applicant's commitments to install the Westinghouse ICCM-86 and associated hardware, to perform preliminary calibration and scaling of the ICC system, and to perform preoperational testing on the ICC system before fuel loading are acceptable.

(2) The applicant's commitment to complete final calibration and scaling of the ICC instrumentation before initial criticality is acceptable.

(3) The proposed ICC system meets the guidance of NUREG-0737, Item II.F.2, with deviations on the channel separation criteria in two areas (some sensor connections to the RCS are shared between two redundant divisions, and thermocouple cables do not maintain minimum separation between redundant divisions in the vicinity of the reactor vessel head). These exceptions have been judged acceptable in several other Westinghouse plants, and are thus acceptable for Watts Bar. Therefore, the proposed ICC system is acceptable.

The staff's final acceptance, however, is contingent on its review of the implementation letter report [see letter, P. S. Tam (NRC) to M. O. Medford (TVA) dated July 24, 1992]. That letter report should be submitted in time for the staff to complete its review before issuance of the full-power operating license. On the basis that the staff found the applicant's design, and the applicant's implementation schedule acceptable, proposed License Condition

3 is considered resolved. The staff will continue to track implementation activities by TAC M77132 and M77133.

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

This section, together with Section 6.6 and Appendix Z, fully resolves Outstanding Issue 5 for Unit 1. This section was prepared with the technical assistance of a contractor (see Appendix E).

5.2.4.1 Compliance With 10 CFR 50.55a(g) for Watts Bar, Unit 1

This evaluation supplements conclusions in the same section of the SER that addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The design of the ASME Code Class 1 and 2 components of the reactor coolant pressure boundary incorporates provisions for access for inservice examinations, as required by Paragraph IWA-1500 of Section XI of ASME Code. In 10 CFR 50.55a(g), the staff defines the detailed requirements for the preservice and inservice programs for light-water-cooled nuclear power facility components. On the basis of the construction permit date of January 23, 1973, this section of the regulations requires that a preservice inspection (PSI) program be developed and implemented using the editions and addenda of Section XI of the ASME Code in effect 6 months before the date of issuance of the construction permit. The components (including supports) may meet the requirements in subsequent editions and addenda of this code that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein. The basic PSI program complies with the requirements of the 1974 Edition of the code, including addenda through Summer 1975, with the following exceptions:

(1) Eddy current examination of heat exchanger tubing, for which the Summer 1975 Addenda do not provide, meets the requirements of the Summer 1976 Addenda.

(2) The following examinations are in accordance with the 1977 Edition, Summer 1978 Addenda of Section XI:

  (a) Class 2 pressure-retaining bolting examinations

  (b) Class 2 valve body weld examinations

  (c) component support integrally welded attachment examinations for piping, pumps, valves, and pressure vessels

  (d) component support examinations for piping, pumps, and valves

  (e) technique for ultrasonic examination of piping welds in accordance with Paragraphs IWA-2232(b) and IWA-2232(c) for examinations performed after October 20, 1981

(f) standards for evaluation of examination results for piping welds (Paragraph IWA-3000)

(g) examination of interior clad surfaces of reactor vessels and other vessels not required

(h) reactor vessel interior and core support structure examinations

(i) Class 1 pressure-retaining piping weld examinations (Examination Category B-J) performed after July 1, 1989

(j) Class 1 pressure-retaining dissimilar metal weld examinations (Examination Category B-F) performed after September 1, 1991

The staff reviewed the applicant's letters of June 9, July 1, and August 13, 1980; April 18, 1983; January 6, July 10, September 21, and November 7, 1984; January 30, February 19, May 14, and August 2, 1985; January 24, 1986; July 27, 1987; April 30, 1990; the results of a meeting with the applicant on November 16, 1981; a Southwest Research Institute (SwRI) supplemental report on the reactor pressure vessel (RPV) PSI ultrasonic examination limitations at Watts Bar, Unit 1 (see FSAR Question 121.23); the December 11, 1990, response to the NRC request for additional information; and the PSI program, through Revision 23, submitted November 4, 1991.

SwRI performed a preservice ultrasonic (UT) examination of the Unit 1 RPV during October and November of 1978. This examination was based on the requirements in Sections V and XI of the 1974 Edition, Summer 1975 Addenda, of the ASME Code. Except for the closure head and bolting, the RPV UT examinations were performed from the inside surface of the vessel using mechanized positioning equipment and an automatic data acquisition system. SwRI reported that the UT examinations revealed insignificant and geometric indications, as well as flaw indications, which were evaluated by SwRI as being acceptable under the code. In anticipation of meeting the reporting requirements of NRC Regulatory Guide 1.150 for future inservice inspection examinations of the RPV, SwRI performed a comprehensive review of the 1978 PSI data to further describe and quantify the examination limitations that were experienced.

Two types of limitations were encountered most frequently during the preservice inspection of the RPV welds and components: (1) interference from search unit wedge-to-component near-surface interface noise and (2) component interference with the scanning equipment and/or geometric shadowing of examination areas.

Interface noise inhibited resolution capabilities at the near surface for about 1/2 to 2 inches of sound (metal) path for shear wave examination and 1 to 2 inches of metal path for longitudinal wave examinations. However, SwRI noted that electronic gating did not result in any examination limitations, since the entire instrument screen presentation was monitored during the examinations, videotaped, and reviewed independently following the examinations. Refinements in equipment design and examination procedures have greatly reduced limitations resulting from geometric shadowing and/or component interference and will enable increased coverage of the ASME Code-required examination volumes for future inservice examinations.

Tables and figures in the RPV report also quantified the limitations to examination of specific RPV welds in terms of percent of code-required examination volume that was not effectively covered. Many areas not effectively examined during PSI received ASME Code Section III fabrication examinations, thereby demonstrating acceptable preoperational integrity. The staff considers these Unit 1 RPV examinations acceptable for PSI, as the fabrication and preservice examinations were consistent with the applicable code and the commercial practices at the time of the examination.

In the December 11, 1990, response to the NRC request for additional information regarding systems that may have been modified or reworked since original PSI examinations were performed, the applicant reported that all components that have been modified, repaired, or replaced will receive new PSI baseline examinations.

The staff reviewed the Watts Bar, Unit 1 PSI program through Revision 23, submitted November 4, 1991. Revision 23 contains a complete listing of the revised, withdrawn, and new requests for relief from the ASME Code Section XI requirements that the applicant has determined are not practical. The staff's evaluation of the relief requests is based on Revision 23, and not on any information from the previous submittal. All of the relief requests were supported by information pursuant to 10 CFR 50.55a(a)(3). The staff evaluated these requests for relief and concluded that the applicant has demonstrated that either (1) the proposed alternatives would offer an acceptable level of quality and safety or (2) compliance with the specific requirements of Section XI would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

After reviewing the applicant's submittal, and the authorization of relief from these preservice examination requirements, the staff concludes that the preservice inspection program for the reactor coolant pressure boundary at Watts Bar Nuclear Plant, Unit 1 is acceptable and in compliance with 10 CFR 50.55a(g)(2). The detailed evaluation supporting this conclusion appears in Appendix Z to this SSER.

The applicant has not submitted the initial inservice inspection (ISI) program. The staff requires that this program be submitted within six months of the date of issuance of the operating license. The ISI program plan will be evaluated on the basis of 10 CFR 50.55a(g)(4), which requires that the initial 120-month inspection interval comply with the requirements in the latest edition and addenda of Section XI of the code incorporated by reference in Paragraph 50.55a(b) on the date 12 months preceding the date of issuance of the operating license. This plan will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when inservice inspections commence. The staff's review of the ISI program will continue to be tracked by proposed License Condition 4.

5.2.4.2 Compliance With 10 CFR 50.55a(g) for Watts Bar Unit 2

The applicant submitted the Watts Bar, Unit 2 PSI program through Revision 11 on April 30, 1990. In Appendix D of the program, the applicant submitted a partial listing of requests for relief from the ASME Code Section XI requirements that the applicant has determined are impractical for Unit 2. The staff will continue its review on Unit 2 as submittals are provided by the applicant and will continue to track this effort by Outstanding Issue 5.

5.4 Component and Subsystem Design

5.4.3 Residual Heat Removal System

In the SER, the staff stated that it cannot reach a conclusion on the matter regarding natural circulation test [Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System"] until the Diablo Canyon findings have been reviewed and their applicability to Watts Bar evaluated and confirmed. The staff assigned Confirmatory Issue 15 to track this review.

By letter of July 11, 1991, the applicant submitted an analysis based on cooldown tests performed at Diablo Canyon to demonstrate compliance with BTP RSB 5-1. BTP RSB 5-1 requires that test programs for pressurized-water reactors (PWRs) include tests with supporting analysis to (1) confirm that adequate mixing of borated water added before or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing and (2) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operation procedures. In addition, the unit is to be designed so that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. The Diablo Canyon test results have been reviewed in the past and accepted by the staff as a basis for concluding that BTP RSB 5-1 had been met (safety evaluation of February 1987). The applicant has demonstrated, through comparison of systems and equipment, that the results of the Diablo Canyon natural circulation boron mixing cooldown tests are applicable to Watts Bar.

The applicant's analysis includes a two-part approach to demonstrate compliance with BTP RSB 5-1. First, the analysis compared the major systems related to natural circulation cooldown of Watts Bar to those of Diablo Canyon Unit 1 in sufficient detail to evaluate the system capabilities for natural circulation, boration, cooldown, and depressurization. The systems evaluated were the reactor coolant system (RCS), the auxiliary feedwater (AFW) system, the main steam system, the chemical and volume control system (CVCS), the residual heat removal (RHR) system, and reactor vessels and internal components. Second, the analysis includes a transient simulation that demonstrates the capability of the Watts Bar plant to attain cold shutdown conditions for a postulated worst-case scenario.

Each unit has four heat transfer loops with a steam generator and a reactor coolant pump and similar reactor power levels per loop (853 MWt for Watts Bar and 834 MWt for Diablo Canyon). Although the units have different reactor coolant pumps, the design features are similar enough not to affect natural circulation.

The Watts Bar steam generator (SG) has a shorter tube bundle elevation and incorporates a preheater in the lower tube bundle region which is not included at Diablo Canyon. An applicability study was performed to show that the SG tube bundle elevation differences have minimal effect on the natural circulation flow capabilities for Watts Bar. The longer tube bundle in the Diablo Canyon SG would result in approximately 8-percent higher driving head when compared to the Watts Bar SG and, consequently, a somewhat higher natural circulation flow rate.

Each unit incorporates two motor-driven pumps and one turbine-driven pump. The systems at both plants possess the same capabilities of supplying auxiliary feedwater to two steam generators from each motor-driven pump and to all steam generators from the turbine-driven pump.

The SGs at both units have pressure-relief valves which are used for plant cooldown. The Diablo Canyon SG pilot-operated relief valves (PORVs) are air-operated valves. There are four SG PORVs (one for each steam line), each with the capacity of 383,000 lb/hr at 775 psig. Watts Bar has four SG PORVs that are safety class seismic Category I and environmentally qualified with larger capacities.

The Diablo Canyon natural circulation cooldown test used the charging pumps to charge through the safety injection system (SIS) boron injection tank (at 20,000 ppm boron) into the RCS. Subsequent charging was aligned from the volume control tank in the CVCS. The boron concentration in the volume control tank was adjusted to 2,000 ppm to simulate charging from the refueling water storage tank (RWST).

At Watts Bar, the boric acid (nominal concentration of 21,000 ppm boron) is normally pumped from the boric acid storage tank by the boric acid transfer pumps to the suction of the centrifugal charging pumps. A backup source of boric acid is available from the RWST, at a minimum of 2,000 ppm boron.

The comparison of the RHR systems at both plants did not reveal any appreciable differences.

Another comparison of the natural circulation capabilities of Watts Bar and Diablo Canyon is the hydraulic resistance coefficients of the system piping. The hydraulic resistance coefficient is based on the vessel and internal components, thus accounting for the slight variation between the plants' coefficients (298.0×10^{-10} for Diablo Canyon and 262.9×10^{-10} for Watts Bar). The lower coefficient at Watts Bar would contribute to a slightly higher natural circulation flow rate at Watts Bar, all other factors being even.

The Diablo Canyon boron mixing test demonstrated adequate boron mixing under natural circulation conditions when highly borated water at low temperatures and low flowrates (relative to RCS temperature and flowrate) was injected into the RCS. The test also evaluated the time delay associated with boron mixing under these conditions. The acceptance criterion for this portion of the test was that the RCS hot legs indicated that boron concentration had increased by 250 ppm or more. This would require that the active portions of the RCS be borated.

The boron injection test conducted at Diablo Canyon consisted of flushing 20,000 ppm boron solution into the RCS from the boron injection tank (BIT). Within 12 minutes, natural circulation had provided adequate mixing to increase the boron concentration in the RCS by 340 ppm. The boron concentration in the BIT at Watts Bar is slightly higher than Diablo Canyon (21,000 ppm nominal).

Since natural circulation flow at Watts Bar is expected to be similar to the flow at Diablo Canyon, adequate mixing of boron would also be provided for Watts Bar.

The cooldown portion of the Diablo Canyon test demonstrated the capability to cool the RCS down to RHR system initiation conditions at approximately 25 °F/hour using all four steam generators on natural circulation. For Watts Bar, cooldown capability will be similar to Diablo Canyon because of similarities in the design of the RCS, AFW, main steam, and RHR systems. The upper head cooldown for Watts Bar is expected to occur at a rate comparable to or exceeding that of Diablo Canyon Unit 1. The upper head volume for Watts Bar is about twice the volume of Diablo Canyon; however, the Watts Bar reactor vessel spray nozzle between the downcomer and the upper head region has a flow area larger than that of Diablo Canyon. The Watts Bar upper head design results in better flow communication and mixing in the upper head during natural circulation cooldown. The enhanced flow-mixing capability of a T_{cold} upper head design plant, like Watts Bar, allows a maximum RCS natural circulation cooldown rate of 50 °F/hour (28 °C/hour) to be used under normal conditions. This rate is twice as fast as the recommended natural circulation cooldown rate of 25 °F (14 °C/hour) for T_{hot} upper head design plants such as Diablo Canyon.

The depressurization portion of the Diablo Canyon test demonstrated the capability to control pressure in the RCS under natural circulation conditions. Pressure control capability included the ability to maintain adequate RCS pressure and the ability to significantly reduce RCS pressure when needed to initiate RHR system operation. To meet these requirements, the Watts Bar pressure control and depressurization capability is expected to be similar to Diablo Canyon due to similarities in the design of the RCS and CVCS.

As part of the guidelines of BTP RSB 5-1, all Class 2 plants must demonstrate the ability to achieve cold shutdown conditions from full-power operation by using only safety-grade equipment and assuming a credible single failure. To meet these requirements, the applicant relied on a comparison to a previously performed test and on transient simulation of the Watts Bar natural circulation cold shutdown scenario which is discussed below.

The Westinghouse proprietary Transient Real-Time Engineering Analysis Tool (TREAT) computer code was used for performing the simulation. TREAT has been used in the development of Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) as well as plant-specific emergency operating procedures (EOPs). As part of the validation of TREAT for simulating a natural cooldown scenario, an analysis was performed for the Diablo Canyon natural circulation cooldown and boron mixing test of March 28-29, 1985, using a Diablo Canyon-specific TREAT model. The analysis demonstrated that TREAT adequately predicted the key elements that are important for the natural circulation cooldown scenario.

The worst-case scenario for the TREAT simulation of Watts Bar was defined on the basis of the limiting set of plant equipment available under the guidelines of BTP RSB 5-1. The most limiting single failure was determined to be an electrical train failure in which only one train of equipment would be available for the operator functions. The initiating event for the cold shutdown scenario was assumed to be a reactor trip with concurrent loss of offsite power and associated RCP trip. The reactor core is assumed to be operating at 102 percent of power with beginning-of-life (BOL) equilibrium xenon conditions preceding the initiating event.

The results of the TREAT thermal-hydraulic analysis of the Watts Bar natural circulation cold shutdown demonstrated that with the limited qualified equipment available for this worst-case scenario, the Watts Bar plant can attain RHR initiation conditions, primary pressure of 395 psia and T_{hot} at 350 °F, in approximately 13.8 hours including a 4-hour hot standby period. The one remaining train of RHR may then be initiated to bring the plant to cold shutdown conditions.

The applicant has compared equipment at Watts Bar to equipment at Diablo Canyon. The comparison proved the plants similar enough to justify the use of the Diablo Canyon natural circulation/boron mixing/cooldown test for Watts Bar. The applicant also included as part of the analysis a computer simulation of thermal-hydraulic behavior during the Watts Bar natural circulation cooldown scenario. This was done using the Westinghouse TREAT computer code. The applicant successfully demonstrated through this simulation that the Watts Bar plant can attain RHR initiation conditions in approximately 13.8 hours.

The staff finds the methods used and conclusion drawn by the applicant acceptable.

This resolves Confirmatory Issue 15. See also Section 13.2.1 and Chapter 14 for more information.

6 ENGINEERED SAFETY FEATURES

6.6 Inservice Inspection of Class 2 and 3 Components

This section, together with Section 5.2 and Appendix Z, fully resolves Outstanding Issue 5 for Unit 1. This section was prepared with the technical assistance of a contractor (see Appendix E).

6.6.3 Compliance With 10 CFR 50.55a(g) for Watts Bar, Unit 1

This evaluation supplements the same section of the SER that addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The design of the ASME Code Class 2 and 3 components incorporates provisions for access for inservice examination, as required by Paragraph IWA-1500 of Section XI of the ASME Code. In 10 CFR 50.55a(g), the staff defines the detailed requirements for the preservice and inservice programs for light-water-cooled nuclear power facility components. On the basis of the construction permit date of January 23, 1973, this section of the regulations requires that a preservice inspection program be developed and implemented using the editions and addenda of Section XI of the ASME Code in effect 6 months before the date of issuance of the construction permit. The components (including supports) may meet the requirements in subsequent editions and addenda of the code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The basic PSI program complies with the requirements of the 1974 Edition of the code, including addenda through Summer 1975, with the following exceptions, which are in compliance with the 1977 Edition, Summer 1978 Addenda of Section XI:

(1) Class 2 pressure-retaining bolting examinations

(2) Class 2 valve body weld examinations

(3) component support integrally welded attachment examinations for piping, pumps, valves, and pressure vessels

(4) component support examinations for piping, pumps, and valves

(5) technique for ultrasonic examination of piping welds in accordance with Paragraphs IWA-2232(b) and IWA-2232(c) for examinations performed after October 20, 1981

(6) standards for evaluation of examination results for piping welds (Paragraph IWA-3000)

(7) examination of interior clad surfaces of reactor vessels and other vessels not required

The staff reviewed the applicant's letters of June 9, July 1, and August 13, 1980; April 18, 1983; January 6, July 10, September 21, and November 7, 1984; January 30, February 19, May 14, and August 2, 1985; January 24, 1986;

July 27, 1987; April 30, 1990; the results of a meeting with the applicant on November 16, 1981; the December 11, 1990, response to the NRC request for additional information; and the PSI program, through Revision 23, submitted on November 4, 1991.

In the December 11, 1990, response to the NRC request for additional information regarding systems that may have been modified or reworked since original PSI examinations were performed, the applicant reported that all components that have been modified, repaired, or replaced will receive new PSI baseline examinations.

The November 4, 1991, submittal contained Enclosure 2 describing how the applicant is upgrading the preservice examinations of Class 2 pressure-retaining piping welds to be compatible with future ISI examination requirements. Preservice examination requirements for the emergency core cooling system (ECCS), high-pressure safety injection (HPSI) system, residual heat removal (RHR) system, and containment spray (CS) system Class 2 piping welds will be updated to use portions of ASME Code Section XI 1983 Edition, Winter 1983 Addenda. Paragraph IWC-1220 (Components Exempt from Examination) and Examination Category C-F-1 of Table IWC-2500-1 will be utilized for exemption requirements, weld selection, extent of examination, and examination method. The applicant states that selected welds that have not already been examined will be examined to establish baseline data for future inservice inspections. The staff has reviewed the submittal and finds the proposed examination plan, as outlined in Enclosure 2, acceptable. Enclosure 3 of the November 4, 1991, submittal contains a TVA commitment to incorporate this upgrade in the next revision of the Preservice Inspection Program, TI-50A.

The staff reviewed the Watts Bar, Unit 1, PSI program through Revision 23, submitted November 4, 1991. Revision 23 contains a complete listing of the revised, withdrawn, and new requests for relief from the ASME Code Section XI requirements that the applicant has determined are not practical. The staff's evaluation of the relief requests is based on Revision 23, and not on any information from the previous submittals. All of the relief requests were supported by information pursuant to 10 CFR 50.55(a)(3). The staff evaluated these requests for relief and concluded that the applicant has demonstrated that either (1) the proposed alternatives would offer an acceptable level of quality and safety or (2) compliance with the specific requirements of Section XI would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

After reviewing the applicant's submittals, and granting authorization of relief from certain preservice examination requirements, the staff concludes that the preservice inspection program for ASME Code Class 2 and 3 systems and components at Watts Bar Nuclear Plant, Unit 1 is acceptable and in compliance with 10 CFR 50.55a(g)(2). The detailed evaluation supporting this conclusion appears in Appendix Z to this report.

The applicant has not submitted the initial inservice inspection (ISI) program. The staff requires that this program be submitted within six months of the date of issuance of the operating license. The ISI program plan will be evaluated on the basis of 10 CFR 50.55a(g)(4), which requires that the initial 120-month inspection interval comply with the requirements in the latest edition and addenda of Section XI of the code incorporated by reference in Paragraph 50.55a(b) on the date 12 months preceding the date of issuance of the operating license. This plan will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when inservice inspection commences. The staff's review of the ISI program will continue to be tracked by proposed License Condition 4.

6.6.4 Compliance With 10 CFR 50.55a(g) for Watts Bar, Unit 2

The applicant submitted the Watts Bar, Unit 2 PSI program, through Revision 11, on April 30, 1990. In Appendix D of the program, the applicant submitted a partial listing of requests for relief from the ASME Code Section XI requirements that the applicant has determined are impractical for Unit 2. The staff will continue its review on Unit 2 as submittals are provided by the applicant and will continue to track this action as Outstanding Issue 5.

8 ELECTRICAL POWER SYSTEMS

8.3 Onsite Power Systems

In a letter of February 16, 1985, the applicant informed the staff that it has designed and constructed an additional diesel generator unit that can be used to replace any of the four existing diesel generator units if one should be out of service. That letter was supplemented by other letters: July 23, October 21, November 19, and December 16, 1985. In a letter of July 28, 1986, the staff requested additional information. The applicant responded to that request in a letter of September 13, 1991.

The staff asked the applicant to discuss how the diesel engine starting and control circuit logic system would preclude the engine from failing to start due to a depletion of the starting air supply from repeated activation of the starting relay. This type of failure was discussed in IE Information Notice 83-17, which stated that a built-in time-delay relay in the engine starting and control circuit logic assured that the engine would come to a complete stop before attempting a restart. The applicant's response stated that IE Information Notice 83-17 referenced a type of system logic design that would constitute a requirement to have a built-in time-delay relay necessary to preclude engine start failures. The time-delay relay would block fuel to the emergency diesel generator, but would not inhibit starting air. Therefore, it is possible to deplete the starting air reserve supply during futile attempts to restart the emergency diesel generator. However, the applicant stated that its engine starting and control circuit logic did not have a time-delay mechanism that would function similarly to the type described in IE Information Notice 83-17. Therefore, the condition does not apply to the Watts Bar diesel generators. The staff has reviewed the applicant's response and accepts the applicant's assessment.

This review was tracked by TAC M63606, M82032, and M82033. The staff's review of other issues related to the additional diesel generator may also be found in various sections of SSER 9.

9 AUXILIARY SYSTEMS

In a letter of February 16, 1985, the applicant informed the staff that it has designed and constructed an additional diesel generator unit that can be used to replace any of the four existing diesel generator units if one should be out of service. That letter was supplemented by other letters: July 23, October 21, November 19, and December 16, 1985. In a letter of July 28, 1986, the staff requested additional information. The applicant responded to that request in a letter of September 13, 1991.

In Section 3.5.1.4 of SSER 9, the staff addressed the issue of Watts Bar's conformance to Tornado Missile Spectrum D for the building housing the additional diesel generator. The staff concluded that the design of the additional diesel generator system and associated diesel generator building (ADGB) can withstand Tornado Missile Spectrum D and is in accordance with requirements of General Design Criteria (GDC) 2 and 4 of Appendix A to 10 CFR Part 50 as they relate to protection against natural phenomena and externally generated missiles and is, therefore, acceptable. In Sections 9.4.5 and 9.5.4.1 of SSER 9, and in Section 8.3 of this SSER, the staff documented review results of other issues related to the additional diesel generator.

The staff has completed its review of the applicant's submittals listed above, in particular, the September 13, 1991, response, and documented its review findings in the following sections. The review was tracked by TAC M63606, M82032, and M82033.

9.2 <u>Water Systems</u>

9.2.1 Essential Raw Cooling Water and Raw Cooling Water Systems

The applicant was asked to clarify discrepancies between Figures 9.2-6 and 9.2-11 in the FSAR pertaining to the raw cooling water system and its associated valving. The staff has reviewed the FSAR figures and the applicant's September 13, 1991, response and considers the discrepancies resolved.

9.4 <u>Heating, Ventilation, and Air Conditioning System</u>

9.4.5 Engineered Safety Features Ventilation System

The applicant was asked to define the term "periodic" with regard to testing of the ADGB ventilation and heating systems. The applicant stated that the systems are tested initially as part of the preoperational test program (Test TVA-74C). The systems will only be tested again after maintenance or modification activities have been completed to verify proper operation of the system or component. The staff was concerned about the operability and availability of the safety-related ventilation system because of the effect the ventilation system could have on the diesel engine operability if the ventilation system were not functional. The heating system is a non-safety-related system and does not require periodic testing after preoperational tests have been completed.

The staff discussed with the applicant on April 27, 1992, to address the concern regarding periodic testing of the ventilation system for the diesel generator building. The applicant stated that the ventilation systems will be functionally tested to verify that the system or component is operable during Technical Specifications-required diesel engine surveillance tests, and after maintenance or modification activities were completed. The staff considers such requirements concerning the ventilation system functional testing appropriate to ensure operability. The applicant has, by Amendment 70 to the FSAR, revised Section 9.4.5.2.2.4, except for editorial changes, in accordance with the commitment. Thus, the staff has no more concerns in this area.

The applicant was asked to describe the effects of a muffler room exhaust fan failure or exhaust blockage on diesel generator operation. In addition, the applicant was asked to submit information on the missile protection for the muffler fan exhaust structure. The applicant stated that the muffler room exhaust fans are neither safety related nor essential for diesel generator operation. The applicant also stated that the failure of the muffler room exhaust fans would only degrade the habitability of the room, not the operation of the engine. Likewise, blockage of the exhaust fans would not impair operation of the diesel engine, but would only elevate the temperature inside the muffler room. The applicant's response resolves the staff's concern.

The applicant was asked to discuss and justify the provisions made to prevent blockage of the air intakes to the ADGB's 480-V auxiliary board room and electrical board rooms of the other diesel generators, and the tornado missile protection provided for the air intake structures. The applicant stated that the design of the air intakes for the diesel generator buildings makes blockage of the air intakes by snow, ice, or debris from tornados highly unlikely. Air enters under each missile shield from all four sides of the shield and it is very unlikely that the entire available airflow area could be blocked. Combined with the design, the intakes are sufficiently large to ensure that the required airflow can be maintained even with a significant percentage of the airflow area blocked. Manual operator action initiated by the applicant can remove any blockage from the intake structures through access to the rooftops of the buildings. The staff has reviewed the response and considers the justification and design provisions of the air intake structures to be appropriate. However, in response to the staff's question, the applicant has committed to investigate the potential for external blockage of the air intake structure by a missile impact and will report its evaluation and any resulting design change in a future submittal before fuel is loaded. The staff will review the evaluation of missile impacts when submitted by the applicant, and will continue to track this issue by TAC M82032 and M82033.

9.5 Other Auxiliary Systems

The applicant was asked to verify and update 55 previously answered questions concerning the four original emergency diesel generators for applicability to the additional fifth diesel generator and its associated building. The applicant stated that of the 55 previous questions, 29 responses are identical for both the four original diesel generators and the additional diesel generator. The applicant updated 25 responses specifically for the additional diesel generator and associated building. The applicant stated that one question did not apply.

The staff has reviewed the 25 updated responses and has no more concerns. The staff did make one observation: the response to Question 040.90 stated that the maximum flood level was 743 ft. 5 in., while in FSAR Section 2.4.14.1.1 the applicant indicated it was 740.1 ft. The response to Question 040.90 was submitted before the SER was issued in 1982, and the applicant serialized all pre-SER responses and incorporated them in the FSAR. The staff recognizes the historical nature of the response, accepts 740.1 ft. as the currently documented maximum flood level, and has no more question about the discrepancy.

9.5.1 Fire Protection

The applicant was asked to submit the fire-hazards analysis for the ADGB. The applicant stated that the fire-hazards analysis for the ADGB will be included in the Watts Bar Fire Protection Report which will be incorporated into FSAR Section 9.5.1 in a future amendment. In a letter of February 5, 1992, the applicant submitted the Watts Bar Fire Protection Report; the staff is reviewing this report within the framework of Outstanding Issue 12. Thus, the applicant has fulfilled its commitment.

The applicant was asked to submit a comparison of the fire-protection features of the ADGB to Branch Technical Position (BTP) 9.5-1, Appendix A guidelines and applicable sections of 10 CFR Part 50, Appendix R. The applicant stated in its response that the ADGB design would be limited to a comparison to BTP 9.5-1, Appendix A, Sections F.9 and F.10 guidelines. The staff has reviewed the ADGB design for compliance with Sections F.9 and F.10 of Appendix A and considers the design in conformance with these sections. The requested comparison was submitted on February 5, 1992, in the Watts Bar Fire Protection Report; the staff is reviewing this report within the framework of Outstanding Issue 12.

The applicant was asked to verify that the fire-fighting systems installed in the ADGB meet GDC 3 of Appendix A to 10 CFR Part 50 to ensure that fire-fighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components important to safety. In response, the applicant stated that the ADGB automatic fire-suppression system uses a pre-action valve to control flow into the sprinkler piping and closed-head spray nozzles. The sprinkler piping will remain dry and a rupture would not result in discharge of water unless a fire has been detected and a control signal has been initiated. Inadvertent operation is unlikely because the closed-head spray nozzles only allow water to flow if a fusible link melts and opens the valve in the presence of an actual fire. The response resolves the staff's concern.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

The applicant was asked to discuss how the fuel oil storage tank fill lines design meets tornado missile protection and seismic Category I requirements. The staff reviewed the applicant's response with regard to the design of the ADGB fuel oil storage tank fill lines. The design of the ADGB fuel oil storage tank fill lines is similar to that of the previously accepted (in the SER) design for the four diesel generators in the diesel generator building. The justification of the protection provided for seismic events and tornado missiles is appropriate, and the response is acceptable.

The applicant was asked to discuss why the pump's high-level interlock feature is not employed when using the ADGB fuel oil transfer pump, or while transferring fuel to the ADGB fuel oil storage tanks from the yard storage tank, as is incorporated in the design of the diesel generator building storage tank pumps. In response to the question, the applicant submitted a proposed clarification of the design statement to be incorporated into FSAR Section 9.5.4.2. The staff reviewed the proposed clarification and it is acceptable. The applicant submitted the clarification of the design statement in Amendment 69.

In SSER 9, the staff erroneously inferred that there is one fuel oil storage tank for each of the five emergency diesel engines. In reality, as is clearly described in FSAR Section 8.3.1.1 and Figures 8.3-1 and 8.3-1A, each diesel engine has four interconnected tanks embedded in the building foundation floor which hold more than a seven-day supply of fuel oil. This correction does not change the staff's conclusion in the SER.

9.5.6 Emergency Diesel Engine Starting System

The applicant was asked to discuss measures that have been incorporated in the diesel engine electrical starting system to protect the components from water spray. The applicant stated that the emergency diesel engines at Watts Bar would use a compressed-air starting system and would have only two electrical components (solenoid valve and pressure switch) that could be affected by water spray. The solenoid valve is a type also used for outdoor applications. The pressure switch is enclosed in a National Electrical Manufacturers Association (NEMA) Type 13 enclosure which is designed for water spray. The response is appropriate and the design of the electrical components is acceptable for protecting against water spray.

The applicant was asked to discuss any diesel engine control functions supplied by the air starting system, and to state whether such controls could interfere with the diesel engine's ability to perform its safety function once it has started. The applicant stated that there are no pneumatic engine controls supplied by the air starting system and no air-operated trips on any of the emergency diesel engines at Watts Bar. This response is acceptable.

9.5.7 Emergency Diesel Engine Lubricating Oil System

The applicant was asked to submit information on the provisions to replenish the lube oil system without interrupting operation of the diesel generator. The applicant stated that oil can be added to the sump at any time from 55-gallon drums through the scavenger strainers on the diesel generators. Administrative procedures to add oil are established and available for trained personnel to use when necessary. The provision to replenish lube oil resolves the staff's concern.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

The applicant was asked to address concerns regarding the products of combustion from a fire in the air intake/muffler room, or from the diesel generator exhaust gases being introduced into the ADGB 480-V auxiliary board room or the electrical board rooms of the other four diesel generators. The effect of degraded air taken into the diesel generators could cause a failure of more than one diesel generator from combustion particles and combustion gases affecting the electrical components in these rooms. The applicant stated that the design of the building structural features and ventilation systems for the additional diesel engine and ADGB should preclude the introduction of degraded air from a fire or diesel generator exhaust gases. In addition, the applicant explained that for a fire in an air intake/muffler room, smoke and other products of combustion would not adversely affect the electrical components in the diesel generator building electrical board rooms or the ADGB 480-V auxiliary board room. The staff has reviewed the applicant's response and considers the design of the ventilation system adequate to mitigate the stated problem.

The applicant was asked to discuss procedures for inspections, surveillance requirements, and testing that will be performed on the diesel generator exhaust system to prevent failure of the system that could result in one or more of the diesel generators being inoperable. The applicant stated that the intake and exhaust piping components for each diesel generator unit are located in separate rooms. The postulated exhaust system failure could propagate to components in the affected diesel generator intake system with the net effect on safety a loss of a single diesel generator. The staff has reviewed the response and finds the design of the diesel generator intake and exhaust system acceptable in accordance with acceptance criteria in the SRP.

The applicant was asked to verify that the pressure losses through the diesel generator air intake and exhaust system do not exceed manufacturer's recommendations. The applicant stated that pressure switches are provided in the systems alarm to identify conditions that exceed manufacturer's limitations. In addition, based on calculations performed during preoperational tests, the design of the system has been determined to be within the manufacturer's recommendations. Staff review of the applicant's response determined that the design of the diesel generator air intake and exhaust systems is acceptable.

11 RADIOACTIVE WASTE MANAGEMENT

11.7 NUREG-0737 Items

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

In SSER 5, the staff concluded that the applicant's leakage-reduction program is acceptable, and stated that the proposed License Condition 24 will be resolved if the applicant accepts the inclusion of the waste gas disposal system (WGDS) in the leakage-reduction program. In SSER 6, the staff reported that, as a result of the applicant's compliance with Generic Letter 89-01, dated January 31, 1989, the specification for the WGDS would be relocated to the Offsite Dose Calculation Manual (ODCM). The staff's concern was that WGDS requirements be formally registered.

In two letters, each dated August 27, 1992, the applicant submitted the ODCM and the draft Technical Specifications (TS), respectively. The WGDS is located in Section 5.7.2 of the draft TS, not in the ODCM. The draft TS is being reviewed by the staff, to be completed before issuance of a low-power license.

The applicant's inclusion of the WGDS specification in the draft TS resolved the staff's original concern that the specification be formally registered. Hence, the staff concludes that proposed License Condition 24 is no longer needed.

12 RADIATION PROTECTION

The staff has reviewed FSAR Amendments 65 through 71 and a letter from the applicant of January 3, 1991, against the guidelines of Standard Review Plan (SRP) Chapter 12. The staff's review efforts were tracked by TAC M63647, M80143, M80144, M82644, M82645, M84234, and M84235. The staff performed part of its review on site on August 26, 1992. The following sections document the staff's findings on Amendments 65 through 71. The applicant is amending the FSAR to reflect the recently revised 10 CFR Part 20; the staff will report its review of that amendment in a future SSER.

12.4 Design Features

The applicant has revised the operational test frequency of area radiation monitors from monthly to quarterly. Radiation monitor stability is ensured by a program that includes check source response checks at least once a month, and analog channel operational tests quarterly, as well as a two-point channel calibration, at least once every 18 months with a source traceable to national standard sources. This program is consistent with provisions of American National Standards Institute (ANSI) Standard 6.8.1-1981 and meets the acceptance criteria in the SRP. Therefore, it is acceptable.

The applicant has designed the facility in five radiation zones based on radiation source intensity and occupancy requirements. Zones IV and V correspond to areas of the plant that are high-radiation areas as defined in 10 CFR Part 20. The applicant has proposed to control access to these high-radiation areas by technical specifications. The staff is developing the Watts Bar Technical Specifications and will ensure that appropriate requirements are imposed consistent with current guidelines. The staff finds the applicant's program meets the provisions of 10 CFR 20.1601(c) and the acceptance criteria in SRP Section 12.3 and is, therefore, acceptable.

12.6 Control Program

The applicant's radiological controls (RADCON) program, formerly the Health Physics Program, is administered by the Radiological Controls Manager. The RADCON organization includes health physics professionals and technicians. Personnel qualifying for positions in the RADCON program (including the RADCON Manager) after January 1, 1990, will meet the training and qualification criteria of Regulatory Guide 1.8, Revision 2 (April 1987). Personnel qualified for these positions before that date meet the training and qualifications of Regulatory Guide 1.8, Revision 1-R (May 1977).

The applicant has revised the radiation dosimetry processing. Personnel radiation monitoring is conducted quarterly using thermoluminescent dosimeters (TLDs) supplemented by real-time dose tracking with direct reading dosimeters (DRDs). The onsite TLD processing by Watts Bar personnel is accredited through the TVA accreditation, as a TLD processing laboratory in all eight categories described in ANSI Standard N13.11-1983, by the National Voluntary Laboratory Accreditation Program. The program for monitoring personnel as described in the FSAR is adequate to ensure that Watts Bar personnel will not exceed the dose limit in 10 CFR Part 20, and meet the requirement of 10 CFR 20.1501(c).

On the basis of its review, the staff finds that the applicant's organizational structure can provide adequate support for the Watts Bar RADCON program, and can ensure independence from the operations staff by having the RADCON Manager and the Operations Manager report to the same management level. The organizational changes in FSAR amendments stated above meet the staff's acceptance criteria in SRP Chapter 12, and are, therefore, acceptable. On the basis of this conclusion, Outstanding Issue 25 is considered resolved.

13 CONDUCT OF OPERATIONS

13.2. Training

13.2.1 Licensed Operator Training Program

In Chapter 14 of the SER, the staff stated that the applicant made a number of changes to the initial test program because of staff comments. One of these changes involved adding a test to address the requirements of TMI-2 Task Action Plan Item I.G.1, "Training During Low Power Testing." In a letter of June 16, 1981, the staff stated that the tests should fulfill a number of objectives, including:

> Training
>
> Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance and recovery from natural circulation mode. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.

In a letter of July 11, 1991, the applicant provided an assessment of the applicability of the Diablo Canyon Nuclear Plant natural circulation test to Watts Bar Nuclear Plant, and proposed to delete the natural circulation test from plant startup tests. In Section 5.4.3 of this SSER, the staff accepted the applicant's assessment.

In SSER 9, the staff reported that the applicant certified that the licensed operator training programs have been developed using a systems approach to training (SAT) and utilized a certified simulation facility. The SAT-based training program should ensure that the activities listed in 10 CFR 55.59(c)(3)(i), as appropriate to the facility, are included in the licensed operator training program, specifically, "loss of core coolant flow/natural circulation" as listed in 10 CFR 55.59(c)(3)(i)(J). Hence, the staff concludes that the training requirement of Item I.G.1 is satisfied. This effort was tracked by TAC M79317 and M79318.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

In the SER, the staff proposed two license conditions, 28 and 29, regarding TMI Items I.C.7, "Nuclear Steam Supply System Vendor Review of Procedures," and I.C.8, "Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating Licenses," respectively. Additional information related to the actions was provided by the applicant in a letter of July 27, 1992. The staff has reviewed the applicant's submittals.

Proposed License Condition 28 addresses the applicant's commitment for nuclear steam supply system (NSSS) vendor (Westinghouse Electric Corporation) review of both the Watts Bar power-ascension test procedures and the emergency operating procedures (EOPs) as stated in the applicant's September 14, 1981, response to the TMI Task Action Plan. In a letter of July 27, 1984, TVA stated that vendor review of the Watts Bar power ascension tests and emergency procedures had been completed and that comments and appropriate revisions would be incorporated before Unit 1 startup. The applicant has produced draft emergency operating instructions (EOIs) that conform to the Westinghouse Owners Group Emergency Response Guidelines, Revision 1-A, except for justified plant-specific deviations. These draft EOIs were developed in accordance with TVA's standardized EOI development program, TVA Corporate Standard STD-12.16, "Emergency Operating Instruction Control." Further, TVA is making contractual arrangements for vendor participation in the verification and validation phase of the EOI development process. As a result, the staff believes that the EOP review by the NSSS vendor as specified in proposed License Condition 28 is no longer necessary.

Proposed License Condition 29 regards confirmation of the Watts Bar EOIs to ensure consistency with the Sequoyah Nuclear Plant EOIs with plant-specific differences identified. The staff believes this is no longer necessary based on the availability of the Watts Bar plant-specific simulator for use in verification and validation of the EOIs for Watts Bar (see Section 13.2.1 of SSER 9). Therefore, proposed License Condition 29 is no longer necessary.

For information regarding the staff's planned activities on the Watts Bar EOIs, see Section 13.5.2.1 of SSER 9.

13.6 Physical Security Plan

13.6.4 Access Requirements

In SSER 1, the staff proposed to impose a condition in the Watts Bar operating license because Section 9.1 of the applicant's physical security plan (PSP) allowed designation of the containment as a nonvital area when the fuel is out of the core during major refueling and major maintenance. The staff stated that this was in violation of the regulation and was, therefore, unacceptable.

In a letter of June 17, 1992, the applicant submitted a revised proposed PSP that superseded all previous security commitments and previous PSPs, including one submitted on July 27, 1990. The staff reviewed the June 17, 1992, submittal in view of proposed License Condition 38, and concluded that the provisions for the protection of the containment during major refueling and major maintenance meet the intent of the regulation. Therefore, the staff concludes that proposed License Condition 38 is no longer needed.

14 INITIAL TEST PROGRAM

In the SER, the staff stated that the applicant made a number of changes to the initial test program because of staff comments. One of these changes involved adding a test to address the requirements of TMI-2 Task Action Plan Item I.G.1, "Training During Low Power Testing." In a letter of June 16, 1981, the staff stated that the tests should fulfill a number of objectives, including:

> <u>Testing</u>
>
> The tests should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation and without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.
>
> If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the above training objectives.

In a letter of July 11, 1991, the applicant provided an assessment of the applicability of the Diablo Canyon Nuclear Plant natural circulation test to Watts Bar Nuclear Plant, and proposed to delete the natural circulation test from plant startup tests. In Section 5.4.3 of this SSER, the staff accepted the applicant's assessment. On the basis of the staff's acceptance, and in accordance with the guidance of Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System," the staff agrees that there is no need for the applicant to perform any natural circulation test. This effort was tracked by TAC M79317 and M79318.

15 ACCIDENT ANALYSIS

15.3 Limiting Accidents

15.3.6 Anticipated Transients Without Scram (ATWS)

Status of Salem ATWS Event Issues

On July 8, 1983, the NRC issued Generic Letter (GL) 83-28 in response to the ATWS events at Salem Nuclear Generating Station. This letter addressed actions to be taken by licensees and applicants to ensure that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in pressurized-water reactors.

The staff completed its review of all of the applicant's submittals in response to GL 83-28 and found them acceptable. The following documents were issued to communicate the staff's acceptance of various issues:

- Item 1.1, Post-Trip Review (Program and Procedure) - letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated August 13, 1990 (TAC M77285, M77286)
- Item 1.2, Post-Trip Review (Data and Information Capability) - Inspection Report 50-390, 391/86-04, dated May 28, 1986
- Item 2.1, Equipment Classification and Vendor Interface (Reactor Trip System Components) - letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 18, 1990 (TAC M63610)
- Item 2.2, Part 1, Equipment Classification Program - letter from S. C. Black, NRC, to O. D. Kingsley, TVA, dated June 1, 1989; Part 2 - letter from F. J. Hebdon, NRC, to O. D. Kingsley, TVA, dated September 7, 1990 (TAC M76312, M76313)
- Items 3.1.1 and 3.1.2, Post-Maintenance Testing of Trip System Components - Inspection Report 50-390, 391/86-04, dated May 28, 1986 (TAC M64345, M64346)
- Items 3.1.3 and 3.2.3, Post-Maintenance Testing in Technical Specifications That Could Degrade Safety - letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated July 2, 1990 (TAC M77138, M77139)
- Items 3.2.1 and 3.2.2, Post-Maintenance Testing of All Other Components - Inspection Report 50-390, 391/86-04, dated May 28, 1986
- Item 4.1, Trip System Reliability (Vendor-Related Modifications) - Inspection Reports 50-390/84-53 and 50-391/84-42, dated August 1, 1984 (TAC M77017, M77018)

- Items 4.2.1 and 4.2.2, Preventive Maintenance for Trip Breakers - letter from P. S. Tam, NRC, to M. O. Medford, TVA, June 18, 1992 (TAC M77019, M77020)

- Items 4.2.3 and 4.2.4, Trip Breaker Life Testing and Periodic Replacement - review terminated by the staff on March 23, 1992 (TAC M77086, M77087)

- Item 4.5.1, Reactor Trip System Reliability-Functional Testing - memorandum (available in the Public Document Room) from P. S. Tam to F. J. Hebdon, dated October 9, 1990 (TAC M64349)

- Items 4.5.2 and 4.5.3, Reactor Trip System On-Line Testing - letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 28, 1990 (TAC M64350, R00186)

17 QUALITY ASSURANCE

In SSER 5 the staff provided its evaluation of the applicant's Nuclear Quality Assurance (QA) Program, submitted by letters dated February 15 and June 5, 1990. The staff stated that the applicant's program is in compliance with applicable NRC regulations and is acceptable for the operations phase of Watts Bar.

Subsequent to that evaluation, the applicant submitted additional revisions to the Nuclear QA Program and the staff has issued evaluations by letters. The staff's evaluations are hereby incorporated by reference as follows:

- Letter, E. G. Wallace to NRC, dated January 18, 1991--NRC review and acceptance in letter, B. A. Wilson to D. A. Nauman, dated April 16, 1991.

- Letters, E. G. Wallace to NRC, dated April 1, September 23, December 4, 1991, and January 15, 1992--review and acceptance in letter, A. F. Gibson to D. A. Nauman, dated March 3, 1992.

The applicant's revisions did not change the staff's conclusion reached in SSER 5. All cited documents are filed under TAC M76972.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT,

UNITS 1 AND 2, OPERATING LICENSE REVIEW

NRC Letters and Summaries

March 5, 1992	Letter, P. S. Tam to O. D. Kingsley (TVA), acknowledging that TVA's submittal of February 5, 1992, supersedes previous submittals on fire protection.
March 11, 1992	Letter, P. S. Tam to TVA Senior Vice President, stating that changes to the emergency core cooling system (ECCS) evaluation model are acceptable per 10 CFR 50.46.
March 19, 1992	Letter, P. S. Tam to TVA Senior Vice President, accepting Amendments 54 through 63 for FSAR Section 2.5.
March 27, 1992	Letter, P. S. Tam to TVA Senior Vice President, stating that compliance with Regulatory Guide 1.97 is acceptable.
March 27, 1992	Summary by P. S. Tam of routine licensing status meeting held on March 24, 1992.
March 30, 1992	Letter, P. S. Tam to TVA Senior Vice President, transmitting supplemental safety evaluation on the Master Fuse List Special Program.
April 1, 1992	Summary by P. S. Tam of management meeting held on March 24, 1992.
April 9, 1992	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the mechanical equipment qualification program.
April 10, 1992	Letter, P. S. Tam to M. O. Medford (TVA), transmitting staff position on the quality assurance records corrective action program.
April 10, 1992	Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming site review to determine TVA's readiness for the integrated design inspection.
April 17, 1992	Letter, S. A. Varga to M. O. Medford (TVA), describing status of actions discussed in March 24, 1992, management meeting.

April 24, 1992	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting safety evaluation on non-destructive examination procedures.
April 24, 1992	Letter, P. S. Tam to M. O. Medford (TVA), transmitting supplemental safety evaluation on cable issues corrective action program.
May 1, 1992	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the environmental qualification program.
May 4, 1992	Letter, P. S. Tam to M. O. Medford (TVA), concurring with proposed use of coping approach to address station blackout.
May 6, 1992	Summary by P. S. Tam of routine licensing status meeting held on April 29, 1992.
May 26, 1992	Letter, P. S. Tam to M. O. Medford (TVA), transmitting supplemental audit report on civil calculations program.
May 26, 1992	Letter, P. S. Tam to M. O. Medford (TVA), informing of dates for the upcoming integrated design inspection.
May 28, 1992	Summary by P. S. Tam of routine licensing status meeting held on May 20, 1992.
June 9, 1992	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation accepting Revision 5 of the corrective action program on quality assurance records.
June 11, 1992	Letter, P. S. Tam to M. O. Medford (TVA), accepting TVA's responses to NRC Bulletin 88-05.
June 11, 1992	Letter, S. D. Ebneter to O. D. Kingsley (TVA), agreeing that TVA may resume full Watts Bar construction activities.
June 16, 1992	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting copies of Supplement 9 of the Watts Bar Safety Evaluation Report.
June 23, 1992	Letter, P. S. Tam to M. O. Medford (TVA), accepting TVA's proposed response date to Generic Letter 88-20, Supplement 4, on individual plant evaluation on external events.
July 2, 1992	Summary by P. S. Tam of routine licensing status meeting held on June 29, 1992.
July 14, 1992	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on FSAR Chapter 14.

July 14, 1992	Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming site review of FSAR Chapter 12.
July 14, 1992	Letter, P. S. Tam to M. O. Medford (TVA), accepting TVA's response to the issue on natural circulation.
July 21, 1992	Letter, P. S. Tam to M. O. Medford (TVA), acknowledging receipt of submittal on TVA's program for assurance of completion and for assurance of quality.
July 21, 1992	Summary by P. S. Tam of management meeting held on July 17, 1992.
July 21, 1992	Letter, P. S. Tam to N. J. Liparulo (Westinghouse), accepting proprietary classification of topical report WCAP-12774.
July 24, 1992	Letter, P. S. Tam to M. O. Medford (TVA), requesting future certification of installation completion of instrumentation to detect inadequate core cooling.
July 27, 1992	Letter, P. S. Tam to M. O. Medford (TVA), informing of the staff's completion of review of Revision 3 of the corrective action program on replacement parts.
August 12, 1992	Summary by P. S. Tam of routine licensing status meeting held on July 30, 1992.

TVA Letters

March 2, 1992	Letter, M. J. Burzynski to NRC, informing of planned changes in the Employee Concerns Special Program closure process.
March 30, 1992	Letter, J. H. Garrity to NRC, responding to Bulletin 88-11 on pressurizer surge line thermal stratification.
April 1, 1992	Letter, J. H. Garrity to NRC, submitting additional information on FSAR Section 2.5.
April 1, 1992	Letter, J. H. Garrity to NRC, submitting additional information in response to the staff's audit report of January 31, 1992, on the civil calculation program.
April 6, 1992	Letter, J. H. Garrity to NRC, responding to the staff's position on structural steel evaluation criteria.
April 13, 1992	Letter, J. H. Garrity to NRC, transmitting Norris Laboratory Report WR28-2-85-131 on containment sump screen performance.
April 17, 1992	Letter, M. J. Burzynski to NRC, transmitting topical report TVA-NPOD89-A on TVA nuclear group organization.

April 21, 1992	Letter, J. H. Garrity to NRC, submitting additional information on response to Bulletin 88-05.
April 22, 1992	Letter, J. H. Garrity to NRC, submitting TVA's proposed strategy to comply with 10 CFR 50.63 on station blackout.
April 23, 1992	Letter, M. O. Medford to NRC, advising of implementation of access authorization program that meets requirements of 10 CFR 73.56 and Regulatory Guide 5.66.
April 30, 1992	Letter, J. H. Garrity to NRC, submitting additional information on shallow undercut anchors.
April 30, 1992	Letter, J. H. Garrity to NRC, submitting information on program for assurance of completion and assurance of quality.
May 1, 1992	Letter, J. H. Garrity to NRC, submitting information on microbiologically induced corrosion in safety-related fire-protection pipes.
May 6, 1992	Letter, W. J. Museler to NRC, transmitting FSAR Amendment 70.
May 7, 1992	Letter, W. J. Museler to NRC, responding to Bulletin 88-11 on pressurizer surge line thermal stratification.
May 15, 1992	Letter, W. J. Museler to NRC, transmitting Revision 5 of the corrective action program on quality assurance records.
June 15, 1992	Letter, W. J. Museler to NRC, submitting associated circuit calculations for the fire-protection program.
June 16, 1992	Letter, M. J. Burzynski to NRC, transmitting Revision 0-J of the Physical Security Plan.
June 18, 1992	Letter, W. J. Museler to NRC, submitting draft revision pages to FSAR Chapter 8.
June 22, 1992	Letter, W. J. Museler to NRC, requesting approval for use of leak-before-break technology to pressurizer surge line.
June 30, 1992	Letter, W. J. Museler to NRC, transmitting draft FSAR pages to address the issue of moderate-energy line break flooding.
July 1, 1992	Letter, W. J. Museler to NRC, clarifying certain statements in Revision 4 of the corrective action program for replacement parts.

July 7, 1992	Letter, R. H. Shell to NRC, responding to Generic Letter 92-01 regarding reactor vessel structural integrity.
July 9, 1992	Letter, W. J. Museler to NRC, submitting additional information on the corrective action program on design baseline and verification.
July 17, 1992	Letter, W. J. Museler to NRC, reporting changes to the ECCS evaluation model per 10 CFR 50.46.
July 17, 1992	Letter, W. J. Museler to NRC, transmitting draft FSAR pages to address the issue of pressurized thermal shock.
July 24, 1992	Letter, W. J. Museler to NRC, advising of completion of implementation of the special program on use-as-is conditions adverse to quality.
July 27, 1992	Letter, W. J. Museler to NRC, submitting information to address proposed License Conditions 28 and 29 (NUREG-0737 Items I.C.7 and I.C.8).
July 27, 1992	Letter, W. J. Museler to NRC, advising of completion of implementation of the special program on soil liquefaction.
July 27, 1992	Letter, W. J. Museler to NRC, addressing open issues in the last audit report on the civil calculation program.
August 27, 1992	Letter, W. J. Museler to NRC, transmitting the Offsite Dose Calculation Manual.
August 27, 1992	Letter, W. J. Museler to NRC, transmitting the draft Technical Specifications.

APPENDIX B

BIBLIOGRAPHY

Pacific Gas and Electric Co.

Updated Safety Analysis Report for Diablo Canyon Power Plant Units 1 and 2.

Tennessee Valley Authority

Final Safety Analysis Report for Watts Bar Nuclear Plant Units 1 and 2.

U.S. Nuclear Regulatory Commission

Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," July 1981.

Safety Evaluation Report, Diablo Canyon, Unit 1, "Natural Circulation, Boron Mixing, and Cooldown Test," February 1987.

APPENDIX E

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APPENDIX Q

SUPPLEMENTAL SAFETY EVALUATION: MICROBIOLOGICALLY

INDUCED CORROSION SPECIAL PROGRAM

(This report supplements the safety evaluation issued as Appendix Q in SSER 8.)

SUPPLEMENTAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO THE MICROBIOLOGICALLY INDUCED CORROSION SPECIAL PROGRAM

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2
DOCKET NO. 50-390 AND 50-391

1.0 INTRODUCTION

In SSER 8 and in a letter to the applicant dated September 13, 1991, the staff published a safety evaluation on the applicant's Microbiologically Induced Corrosion (MIC) Special Program. Subsequently, in a letter of May 1, 1992, the applicant submitted information to clarify the MIC Special Program. The staff reviewed this information and concluded that various parts of the safety evaluation need to be updated.

2.0 EVALUATION

A. Inspection

At the end of the first paragraph, add a sentence to read: "The safety-related portion (auxiliary feedwater) of the fire-protection system is also included in the MIC monitoring program TI-31.13."

C. Leak Position

Replace the entire section with the following:

> The staff's position on continued operation after detection of a leaking pipe is that structural integrity is not maintained and a repair/ replacement in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI is required. If the applicant desires relief from ASME Code Section XI repair/ replacement requirements, the provisions of Generic Letter 90-05 should be followed. The use of Generic Letter 90-05 requires that the NRC grant relief from the ASME Code Section XI requirements. Generic Letter 90-05 can only be used after a full-power license has been granted and the reactor has started to operate. Relief from the ASME Code Section XI requirements will not be considered for a unit that is not operating.

E. Treatment

Replace the first sentence of the second paragraph with: "The applicant has installed a bromine/chlorine biocide injection system for treatment of the new water system, including the ERCW and safety-related portions (auxiliary feedwater) of the fire-protection system."

3.0 CONCLUSION

Replace the entire conclusion with:

> The staff concludes that if the Watts Bar MIC program for detection, assessment, and control of MIC in the ERCW system and in the safety-related portion (auxiliary feedwater) of the fire-protection system, is properly implemented, and the commitments in Enclosure 2 of the applicant's February 2, 1991, letter are met, they will provide reasonable assurance that the systems will not lose their capability to perform safety functions due to MIC damage. However, if leakage should occur after an operating license has been granted, the requirements of Generic Letter 90-05 shall apply, and a written request for relief is required for the interim period until a code repair is made during the next scheduled outage exceeding 30 days, but no later than the next refueling outage. Although not a safety concern, the use of biocides and the proposed dispersant/corrosion inhibitor treatment program without prior cleaning of the system may not be as effective as would be expected for a system that had been previously mechanically or chemically cleaned. The cleaning would remove slime, scale, and other material and would improve the effectiveness of biocide and dispersant/corrosion inhibitor treatment.

Principal Contributor: James Davis, Materials and Chemical Engineering Branch, NRR

Dated: August 1992

APPENDIX Z

SAFETY EVALUATION REPORT: PRESERVICE INSPECTION

RELIEF REQUESTS

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PRESERVICE INSPECTION RELIEF REQUESTS

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1

DOCKET NO. 50-390

1 INTRODUCTION

For nuclear power facilities whose construction permits were issued on or after January 1, 1971, but before July 1, 1974, 10 CFR 50.55a(g)(2) specifies that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Code Class 1 and 2 must meet the preservice examination requirements in editions and addenda of Section XI of the ASME Code in effect six months before the date of issuance of the construction permit. 10 CFR 50.55a(g)(2) also states that components (including supports) may meet the requirements in subsequent editions and addenda of this code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

The staff reviewed the applicant's submittals of June 9, July 1, and August 13, 1980; April 18, 1983; January 6, July 10, September 21, and November 7, 1984; January 30, February 19, May 14, and August 2, 1985; January 24, 1986; July 27, 1987; April 30 and December 11, 1990; and November 4, 1991; the agreements in a public meeting with the applicant on November 16, 1981; and the applicant's preservice inspection (PSI) program, through Revision 23, submitted on November 4, 1991. Revision 23 of the PSI program contains a complete listing of the revised, withdrawn, and new requests for relief from the ASME Code Section XI requirements that the applicant has determined are not practical. The staff's evaluation of the relief requests contained in this report is based on Revision 23, and not on information from the previous submittals. The relief requests were supported by information pursuant to 10 CFR 50.55(a)(3). Therefore, the staff reviewed the applicant's submittal according to the requirements of the applicable code and determined if the applicant demonstrated that the proposed alternatives would offer an acceptable level of quality and safety, or if compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

2 TECHNICAL REVIEW CONSIDERATIONS

The construction permit for Watts Bar, Unit 1 was issued on January 23, 1973. ASME first published rules for inservice inspection in the 1971 Edition of Section XI. No preservice or inservice inspection requirements existed before that date. Since the Watts Bar, Unit 1 plant system designs and purchase of long lead-time components were well under way by the time the Section XI rules became effective, full compliance with the access and inspectability requirements was not always possible. The applicant has updated the PSI program to meet requirements in subsequent editions of the code and addenda that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The PSI program is based on the 1974 Edition, Summer 1975 Addenda of Section XI of the ASME Code, with the exceptions discussed in Sections 5.2.4 and 6.6 of SSER 10.

Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain examination and testing requirements that, by themselves, offer the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specifications. As a part of these examinations, all of the primary pressure boundary full-penetration welds were volumetrically examined (radiographed), and the system was subjected to hydrostatic pressure tests.

The intent of a preservice examination is to establish a reference or baseline before the initial operation of the facility. The results of subsequent inservice examinations can be compared with the original condition to determine if changes have occurred in response to inservice degradation. If review of the inservice examination findings shows no change from the original condition, no action is required. Where baseline data are not available, all flaws must be treated as new flaws generated during service and evaluated in accordance with Section XI of the code.

Another benefit of the preservice examination is providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during component fabrication. Successful performance of preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using another volumetric test method.

In the case of Watts Bar, Unit 1, a large portion of the preservice examination required by the ASME Code was performed. Failure to perform a 100% preservice examination of the welds identified below will not significantly affect the assurance of the initial structural integrity; those parts not performed are addressed by the requests for relief within this report.

In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as a part of the inservice inspection program. Requiring supplemental examinations to be performed before plant startup would result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The performance of supplemental examinations, such as surface examinations, in areas in which volumetric inspection is difficult, will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar ASME Code Section III fabrication examinations.

In cases where portions of the required examination cannot be performed because of a combination of component design and current examination technique limitations, the development of new or improved examination techniques will continue to be evaluated. As improvements in these areas are achieved, the staff will require that these new techniques be incorporated into the inservice examination program for the components or welds that received a limited preservice examination.

Several of the preservice inspection relief requests involve limitations to the examination of the required volume of a specified weld. The inservice inspection (ISI) program is based on the examination of a representative sample of welds to detect generic degradation. If the welds identified in the PSI relief requests must be examined again, the possibility of augmented ISI will be evaluated during review of the applicant's initial 10-year ISI program. An augmented program may include increasing the extent or frequency, or both, of inspection of accessible welds.

3 RELIEF REQUESTS

The applicant originally submitted requests for relief from the ASME Code Section XI requirements that it considered not practical (letters of June 9, July 1, and August 13, 1980). Additional information on these requests for relief was obtained from a public meeting with the applicant on November 16, 1981. In Revision 8 of the Watts Bar, Unit 1 PSI program, submitted on February 17, 1983, these relief requests were revised and resubmitted as Appendix E. The Watts Bar, Unit 1 PSI program, including the requests for relief, was revised in its entirety in Revision 12, submitted on April 13, 1984, and again in Revision 22, submitted on April 30, 1990. Revision 23 of the PSI program, submitted on November 4, 1991, contains a complete listing of the revised, withdrawn, and new requests for relief from the code requirements that the applicant has determined are not practical. Therefore, evaluation of the relief requests contained herein is based on Revision 23, and not on any information from the previous submittals. On the basis of this information and review of the design, geometry, and materials of construction of the components, certain preservice requirements of ASME Code Section XI have been determined to be impractical. Imposing these requirements would result in hardships or unusual difficulties without a compensating increase in the levels of quality and safety. Therefore, pursuant to 10 CFR 50.55(a)(3), conclusions that these preservice requirements are impractical are justified. Unless otherwise stated, references to the code refer to the ASME Code Section XI, 1974 Edition, Summer 1975 Addenda.

3.1 Relief Request ISI-1, Examination Categories B-F, B-J, C-F, and C-G, Class 1 and 2 Piping Welds, Notches for Calibration, 50% Distance Amplitude Correction (DAC) Recording

Code Requirement: Section XI, Paragraph IWA-2232 references Paragraph T-530 of ASME Code Section V, Article 5 for the ultrasonic examination of certain piping and vessel welds. Calibration shall be performed using side-drilled hole reflectors. All indications in excess of 20% of the reference level shall be evaluated.

Paragraph IWA-2232 also requires that the ultrasonic examination of Class 1 and 2 ferritic vessels, 2 inches and over in wall thickness, be performed based on Appendix III of Section XI.

Applicant's Code Relief Request: Relief is requested to use 5% of wall thickness (t) notches or, at the applicant's option, 10%t notches for calibration in lieu of side-drilled holes. The use of calibration notches is requested for both piping welds and unclad vessel welds in material less than 2 inches in thickness. In addition, relief is requested to record indications greater than 50% DAC.

Applicant's Proposed Alternative: The applicant currently uses 5%t notches in lieu of side-drilled holes and states that, at its discretion, notches at a nominal depth up to 10%t may be used. The applicant considers these notches technically acceptable on the basis of the calibration requirements of Paragraph III-3430 and Supplement 7 of Appendix III of the 1977 Edition, Summer 1978 Addenda, of ASME Code Section XI.

Applicant's Basis for Requesting Relief: The applicant considers the use of side-drilled holes with standard ultrasonic examination techniques impractical in materials less than 0.375-inch thick; a repeatable distance-amplitude-correction (DAC) curve cannot be established because of saturation of the part by ultrasound.

Paragraph T-537 of ASME Code Section V, 1974 Edition, Summer 1975 Addenda, requires evaluation of all indications in excess of 20% of the reference level (DAC curve). The inherent difficulties in fulfilling these requirements have been recognized and changes have been incorporated into the Summer 1978 Addenda of ASME Code Section XI, Appendix III, Paragraph III-4500, and Paragraph IWA-2232, which requires recording all indications greater than 50% DAC with an evaluation of the indications greater than 100% DAC. The applicant considers the use of strip chart recorders an acceptable recording method for the latter requirements.

Staff Evaluation: The applicant has proposed 5%t or, at the applicant's discretion, up to 10%t notches in lieu of side-drilled holes as reference reflectors for basic calibration for pipe welds and vessels not covered by Appendix I of Section XI. In 10 CFR 50.55a(g), the staff permits updating to meet later-approved code editions. The use of notches for the examination of piping welds is specified in later applicable editions of Section XI, Appendix III, 1977 Edition, Winter 1978 Addenda. For piping welds, this code edition also requires recording of all indications greater than 50% DAC and evaluation of those indications greater than 100% DAC (Paragraph IWA-2232 and Paragraph III-4500 of Appendix III). The staff concludes that the use of the above-specified calibration reflectors and the recording and reporting levels for indications for the preservice examination of piping meet a subsequent referenced edition of Section XI, as permitted by 10 CFR 50.55a(g)(2). Therefore, relief is not required for piping welds.

For the examination of unclad vessel welds in ferritic material less than 1-1/2 inches in thickness, Section XI of the ASME Code specified that side-drilled holes must be used for calibration. The staff identified the components in Examination Categories B-B and C-A pressure-retaining welds in ASME Code Class 1 and 2 vessels in the applicant's PSI program. The relief request essentially involves ASME Code Class 2 vessels because the Class 1 vessels must be examined according to Appendix I of Section XI. In addition, most of the Examination Category C-A welds subject to preservice examination are approximately 1 inch or less in wall thickness; many of the vessel welds are less than 0.5 inch in wall thickness.

It is commercial practice to use calibration reflectors similar to pipe standards to perform the ultrasonic examination of relatively thin-walled vessel shell welds. On the basis of the conclusions in the first paragraph of this evaluation on piping, and the discussion above on vessels, the staff concludes that the use of the notched calibration reflectors, and the recording and reporting level for indications for the preservice examination of relatively thin-walled vessel welds are acceptable alternatives to the code requirements. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), use of the applicant's proposed alternatives is authorized.

The applicant considers recording indications greater than 50% DAC on a strip chart recorder acceptable for PSI. Because the code does not distinguish between manual recording of data and automatic systems, the use of strip chart recorders is permitted by the code. Therefore, strip chart recorders are acceptable to the staff and relief is not required for this specific item. Although the staff concludes that recording indications greater than 50% DAC are acceptable for the baseline data and should detect significant flaws, if present, the applicant should incorporate the following when performing inservice examination for both ferritic and austenitic piping welds:

(1) Any crack-like indication, regardless of ultrasonic amplitude, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determine the shape, identity, and location of the reflector.

(2) Any indication investigated and found to be other than geometrical or metallurgical in nature should be evaluated and corrected.

3.2 <u>Relief Request ISI-2, Examination Category B-L-2, Reactor Coolant Pumps</u>

Relief Request ISI-2 has been withdrawn by the applicant.

3.3 <u>Relief Request ISI-3, Examination Category B-M-2, Class 1 Valves Exceeding 4-in. Nominal Pipe Size</u>

Relief Request ISI-3 has been withdrawn by the applicant.

3.4 <u>Relief Request ISI-4, Examination Categories B-F, B-J, C-F, and C-G Piping Welds (66 Welds Total)</u>

<u>Code Requirement</u>: Section XI, Table IWB-2600, Examination Category B-F welds are required to receive 100% surface and volumetric examinations as defined by Table IWB-2500. Table IWB-2600, Examination Category B-J, and Table IWC-2600, Examination Categories C-F and C-G welds are required to receive 100% volumetric examinations as defined by Tables IWB-2500 and IWC-2500.

<u>Applicant's Code Relief Request</u>: Relief is requested from examining 100% of the code-required volume of 66 specific pressure-retaining welds. The November 4, 1991, submittal of Request for Relief ISI-4 lists the welds, including drawing numbers, physical configuration, and remarks on the scan limitations.

Applicant's Proposed Alternative: A "best effort" ultrasonic examination has been performed in addition to the visual examination performed during system leakage and hydrostatic pressure tests. Also, surface examinations have been performed on all accessible areas of the welds.

Applicant's Basis for Requesting Relief: The applicant states that, in some cases, it was impractical to inspect all welds in accordance with Paragraph T-532 of Article 5, Section V, of the ASME Code, or Appendix III, Subarticle III-4400 of Section XI of the ASME Code (1977 Edition, Summer 1978 Addenda) and achieve meaningful results because of hanger interference, component geometry, or valve and pump casings adjoining the welds. These weld interferences, as well as an approximate percentage of code volume examined for each of the welds, are listed in the November 4, 1991, submittal of Relief Request ISI-4.

Staff Evaluation: This relief request is acceptable for PSI based on the following considerations:

(1) Other similar welds in the same piping runs received full code examinations. Thus, the integrity of the pressure boundary was verified by sampling.

(2) The subject piping welds received a system hydrostatic test in accordance with ASME Code Section III and may also receive a system hydrostatic test at each inspection interval in accordance with ASME Code Section XI.

(3) The accessible portions of the welds listed above received a preservice volumetric and surface examination in accordance with ASME Code Section XI.

(4) The subject welds have received volumetric examination by radiography in accordance with ASME Code Section III during fabrication.

An NRC Region II inspector performed a routine inspection at the Watts Bar, Unit 1 site on September 11 through 15, 1989 (see Inspection Report 50-390, 391/89-15 of September 27, 1989). This NRC inspection included, in part, a review of the Unit 1 PSI plan, reviews of the active requests for relief from required PSI examinations, and random in-field visual verifications to confirm that the relief requests were justified. For Relief Request ISI-4, 11 welds were selected at random and visually confirmed as either being totally uninspectable by ultrasonic techniques or as having limited ultrasonic examination coverage. A review of the historical examination records for the 11 welds selected indicated that the examination commitments stated in the request for relief had been fulfilled.

The staff has determined that the limited Section XI examinations, along with the supplemental surface examinations and the fabrication examinations performed during construction, provide reasonable assurance of the preservice structural integrity of the subject welds. Compliance with the specific requirements of Section XI for the subject welds would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), relief is authorized as requested.

3.5 Relief Request ISI-5, Examination Category C-A, Class 2 Steam Generator Welds (Four Welds Total - One on Each Steam Generator)

Code Requirement: The preservice examination shall include a volumetric examination of the circumferential butt welds that are gross structural discontinuities in the steam generators. This includes weld metal and base metal for one plate thickness beyond the edge of the weld joint. The examination shall cover at least 20% of each circumferential weld, uniformly distributed among three areas around the vessel circumference. In the case of multiple vessels of similar design, size, and service, the required examinations may be limited to one vessel or may be distributed among the vessels.

Applicant's Code Relief Request: Relief is requested from performing the required volumetric examination of one of the subject circumferential butt welds on the steam generators.

Applicant's Proposed Alternative: None.

Applicant's Basis for Requesting Relief: The applicant states that one circumferential shell weld on each steam generator (welds SG-4B-5-1, SG-4B-5-2, SG-4B-5-3, and SG-4B-5-4) is partially inaccessible for examination because of the upper steam generator support brackets. Weld SG-4B-5-1 was examined on a best-effort basis for the preservice examination, and the applicant determined that at least 55% of the code-required volume was examined.

Staff Evaluation: An NRC Region II inspector performed a routine inspection at the Watts Bar, Unit 1 site on September 11 through 15, 1989 (see Inspection Report 50-390, 391/89-15, of September 27, 1989). This NRC inspection included, in part, a review of the Unit 1 PSI plan, reviews of the active requests for relief from required PSI examinations, and random in-field visual verifications to confirm that the relief requests were justified. For Relief Request ISI-5, the inspector confirmed that, as the applicant has stated, these welds are inaccessible for volumetric examination to the extent required by the code, and that the applicant's commitments as stated in the request for relief had been fulfilled.

The staff has determined that the subject welds cannot be examined to the extent specified by the code because of interference by the upper support brackets. These welds have been examined volumetrically by radiography in accordance with ASME Code Section III during fabrication. Therefore, the staff has concluded that the limited best-effort preservice volumetric examination, the fabrication examinations performed during construction, and the hydrostatic pressure test offer reasonable assurance of the preservice structural integrity of the welds. This particular joint, in the existing steam generators throughout industry, has not had any inservice failures. The staff has also determined that compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), relief is authorized as requested.

3.6 Relief Request ISI-6, Examination Category B-B, Reactor Vessel Weld W01-02

Code Requirement: The code requires a 100% volumetric examination of the lower head circumferential dollar weld. This includes weld metal and base metal for one plate thickness beyond the edge of the weld.

Applicant's Code Relief Request: Relief is requested to perform a limited preservice examination on weld W01-02 under conditions and with equipment and techniques equivalent to those that are expected to be used for subsequent inservice examinations.

Applicant's Proposed Alternative: None. A 100% manual ultrasonic preservice examination of the weld will be conducted from the vessel outside diameter. A remote ultrasonic examination will be conducted from the vessel inside diameter on all accessible areas of the weld.

Applicant's Basis for Requesting Relief: The applicant used automated, remote inspection devices to examine most of the reactor vessel welds. These examinations were conducted from the vessel inside diameter. However, the lower head dollar weld (weld W01-02) on the reactor pressure vessel is partially inaccessible for examination from the vessel inside diameter because of instrumentation tubes that penetrate the lower head. Portions of the weld can be examined from one side (as permitted by Paragraph I-5121 of Section XI) using automated equipment and will be examined in accordance with Paragraph IWB-3511.1 of Section XI.

Staff Evaluation: The applicant stated that preservice examinations have been performed using automated, remote inspection devices to examine most of the reactor vessel welds from the vessel interior. The applicant intends to conduct future inservice examinations from the vessel interior with remote inspection devices to minimize radiation exposure to personnel. Certain portions of weld W01-02 are partially inaccessible for ISI examination because of instrumentation tubes that penetrate the lower head. In addition, as part of the preservice examination, the applicant has examined 100% of the code-required volume of circumferential weld W01-02 from the vessel exterior using supplemental manual ultrasonic examination techniques.

The staff has determined that the partial examination using remote inspection devices, supplemented by 100% examination using manual ultrasonic examination techniques, constitutes an examination equivalent to the ASME Code, Section XI, PSI requirement and provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), use of the applicant's proposed alternatives is authorized.

3.7 Relief Request ISI-7, Examination Category B-D, Steam Generator Nozzles

Relief Request ISI-7 has been withdrawn by the applicant.

3.8 Relief Request ISI-8, Examination Category C-D, Class 2 Pressure Retaining Bolting

Relief Request ISI-8 has been withdrawn by the applicant.

3.9 Relief Request ISI-9, Examination Category B-L-1, Reactor Coolant Pump Casing Weld (Four Pumps)

Code Requirement: Section XI, Table IWB-2600, Examination Category B-L-1, Item B5.6 requires a 100% volumetric examination of pump casing welds as defined by Table IWB-2500.

Applicant's Code Relief Request: Relief is requested from performing the code-required preservice volumetric examination of the reactor coolant pump casing welds.

Applicant's Proposed Alternative: All casing welds on the four reactor coolant pumps will receive surface examination for the preservice examinations.

Applicant's Basis for Requesting Relief: Each reactor coolant pump casing consists of a two-piece, welded, Type 304 stainless steel casting. The applicant states that the present capability of ultrasonic testing is not sufficient to examine cast material of this thickness and achieve meaningful results.

Staff Evaluation: The staff has determined that the preservice surface examination, in conjunction with the volumetric examinations performed during fabrication on the reactor coolant pump casing welds, provides an acceptable alternative to the code-required volumetric preservice examination. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), relief is authorized as requested.

With regard to limitations determined by the metallurgical properties of cast stainless steel, the staff will continue to monitor the development of new or improved examination techniques. As improvements are made in these areas, the staff may require that they be made part of the inservice examination requirements for the components or welds that received a limited preservice examination.

3.10 Relief Request ISI-10, Examination Category C-A. Regenerative Heat Exchanger Circumferential Butt Welds (5 Welds)

Code Requirement: Section XI, Tables IWC-2520 and IWC-2600, Examination Category C-A, requires volumetric examination of at least 20% of each circumferential butt weld (head-to-shell, tubesheet-to-shell). This examination shall be uniformly distributed among three areas around the vessel circumference.

Applicant's Code Relief Request: Relief is requested from performing the required examination on three areas uniformly distributed around the regenerative heat exchanger circumference.

Applicant's Proposed Alternative: The five subject circumferential welds have been volumetrically examined in all accessible areas. The total volume examined for each of these welds exceeds the 20% examination requirement.

Applicant's Basis for Requesting Relief: The regenerative heat exchanger has 12 circumferential welds requiring examination. Seven of these welds can be examined to the extent required by the code. Twenty percent of each of the five remaining circumferential tubesheet-to-shell welds (RHX-2, RHX-6, RHX-7, RHX-10, and RHX-11) can be examined; however, because of permanent and nonremovable supports, the examination area cannot be uniformly distributed.

Staff Evaluation: An NRC Region II inspector performed a routine inspection at the Watts Bar, Unit 1 site on September 11 through 15, 1989 (see Inspection Report 50-390, 391/89-15 of September 27, 1989). This NRC inspection included, in part, a review of the Unit 1 PSI plan, reviews of the active requests for relief from required PSI examinations, and random in-field visual verifications to confirm that the relief requests were justified. For Relief Request ISI-10, the NRC inspector performed an in-field visual observation of these welds and confirmed the problem of examination as stated above. A review of the historical examination records for these five welds indicated that the examination commitments stated in the request for relief had been fulfilled.

The code requirement for this examination category is based on sampling; for example, only a portion of the weld is inspected to determine whether a generic degraded condition exists. The applicant has volumetrically examined the subject welds in all accessible areas. This exceeds the 20% requirement; therefore, the staff has determined that the volumetric examinations performed on all accessible regions are an acceptable alternative to the required uniformly distributed preservice examination, and provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(i), use of the applicant's proposed alternative is authorized.

3.11 Relief Request ISI-11, Examination Category C-A, Letdown Heat Exchanger and Excess Letdown Heat Exchanger

Code Requirement: Section XI, Tables IWC-2520 and IWC-2600, Examination Category C-A, requires volumetric examination of at least 20% of each circumferential butt weld (head-to-shell, tubesheet-to-shell). This examination shall be uniformly distributed among three areas around the vessel circumference.

Applicant's Code Relief Request: Relief is requested from performing the required examination on three areas, uniformly distributed around the vessel circumference.

Applicant's Proposed Alternative: The subject circumferential welds have been volumetrically examined in all accessible areas. The total volume exceeds the 20% examination requirement.

Applicant's Basis for Requesting Relief: At least 20% of the circumferential welds have been examined; however, because of geometrical interference, the applicant could not distribute the examination area uniformly.

Staff Evaluation: The code requirement for this examination category is based on sampling; for example, only a portion of the weld is inspected to determine whether a generic degraded condition exists. The staff has determined that the volumetric examinations performed on all accessible regions exceed the required extent of examination and, therefore, represent an acceptable alternative to the required uniformly distributed preservice examination. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), use of the applicant's proposed alternative is authorized.

3.12 Relief Request ISI-12, Examination Category B-J, Reactor Coolant System Main Loop Piping Welds

Relief Request ISI-12 has been withdrawn by the applicant.

3.13 Relief Request ISI-13, Ultrasonic Examination Technique for Piping Welds

Code Requirement: In accordance with 10 CFR 50.55a(g)(2), components (including supports) may meet the requirements in subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b).

Applicant's Code Relief Request: Relief is requested to update the preservice examinations for piping to only portions of the related requirements of the respective editions and addenda.

Applicant's Proposed Alternative: As specified in Relief Requests ISI-1 and ISI-4.

Applicant's Basis for Requesting Relief: The Watts Bar PSI program is based on the 1975 Edition, Summer 1975 Addenda, except for selected categories. The ultrasonic examination technique [IWA-2232(b), IWA-2232(c), and Appendix III] and evaluation (IWA-3000) of piping welds were updated to the 1977 Edition, Summer 1978 Addenda of ASME Code Section XI, except for Appendix III, Paragraph III-3410 (material), III-3430 (calibration notches), and III-4450 (inaccessible welds).

Staff Evaluation: An ambiguity exists in the statement of the regulation associated with the updating of the PSI and ISI programs to meet the provision of later-referenced code editions. The requirement of the regulation applicable to Watts Bar is discussed above. However, other preservice and inservice inspections based on 10 CFR 50.55a(g)(3)(v) and 50.55a(g)(4)(iv), respectively, are required to meet similar provisions, except the cited paragraphs indicate that "portions of editions or addenda may be used."

The applicant has submitted this request for relief to obtain approval to use only "portions of" the 1977 Edition, Summer 1978 Addenda. The staff reviewed the parallel paragraphs associated with the updating of PSI and ISI programs and concluded that 10 CFR 50.55a(g)(2) permits the use of only portions of later-referenced code editions and addenda. The staff reviewed and found acceptable the use of portions of the 1977 Edition, Summer 1978 Addenda, during the evaluation of the PSI program. The major items are discussed in Sections 5.2.4 and 6.6 of SSER 10. Specific requirements that were determined to be impractical are discussed in the evaluation of Relief Requests ISI-1 and ISI-4. On the basis of this conclusion, the staff has determined that relief is not required for Relief Request ISI-13.

3.14 Relief Request ISI-14, Examination Category C-A, Residual Heat Removal Heat Exchanger (1 Weld)

Code Requirement: Section XI, Tables IWC-2520 and IWC-2600, Examination Category C-A requires volumetric examination of at least 20% of each circumferential butt weld at structural discontinuities. This volumetric examination is to be on three areas uniformly distributed around the vessel circumference.

Applicant's Code Relief Request: Relief is requested from performing the volumetric examination on three areas uniformly distributed around the circumference of weld RHRHX-2-1A.

Applicant's Proposed Alternative: The subject weld was volumetrically examined in all accessible areas.

Applicant's Basis for Requesting Relief: Only approximately 18% of weld RHRHX-2-1A can be examined because of limitations from the residual heat removal (RHR) heat exchanger inlet and outlet nozzles and the RHR heat exchanger support pad attachment plates.

Staff Evaluation: An NRC Region II inspector performed a routine inspection at the Watts Bar, Unit 1 site on September 11 through 15, 1989 (see Inspection Report 50-390, 391/89-15 of September 27, 1989). This NRC inspection included, in part, a review of the Unit 1 PSI plan, reviews of the active requests for relief from required PSI examinations, and random in-field visual verifications to confirm that the relief requests were justified. For Relief Request ISI-14, the inspector confirmed that, as the applicant has stated, these welds are inaccessible for volumetric examination to the extent required by the code, and that historical examination records for this weld indicated that the examination commitments stated in the request for relief had been fulfilled.

The code requirement for this examination category is based on sampling; for example, only a portion of the weld is inspected to determine whether a generic degraded condition exists. The applicant has volumetrically examined the subject weld in all accessible areas and has stated that this weld had a radiographic examination during fabrication in accordance with ASME Code Section III. Therefore, the staff has determined that the volumetric examination performed on all accessible areas and the fabrication examination are acceptable alternatives to the required uniformly distributed preservice examination. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), use of the applicant's proposed alternatives is authorized.

3.15 Relief Request ISI-15, Examination Category C-B, Residual Heat Removal Heat Exchanger Nozzle-to-Vessel Attachment Welds (2 Welds)

Code Requirement: Section XI, Tables IWC-2520 and IWC-2600, Examination Category C-B requires a 100% volumetric examination of Class 2 nozzle-to-vessel attachment welds.

Applicant's Code Relief Request: Relief is requested from examining 100% of the code-required volume of RHR heat exchanger nozzle-to-vessel welds RHRHX-4-1A and RHRHX-3-1B.

Applicant's Proposed Alternative: None. The subject nozzle-to-vessel attachment welds were volumetrically examined in all accessible areas.

Applicant's Basis for Requesting Relief: These welds received a limited ultrasonic examination because of limitations from the RHR heat exchanger nozzle geometry and RHR heat exchanger support pad attachment plates.

Staff Evaluation: The staff has determined that the subject welds cannot be examined to the extent specified by the code because of the geometry of the nozzles and support pad attachment plates. These welds have received a radiographic examination during fabrication in accordance with ASME Code Section III. Therefore, the staff has concluded that the limited best-effort preservice volumetric examination, the fabrication examinations performed during construction, and the hydrostatic pressure test provide reasonable assurance of the preservice structural integrity of these welds. In order for the applicant to comply with the specific code requirement, the RHR heat exchanger would have to be redesigned and refabricated. This burden would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), relief is authorized as requested.

4 CONCLUSIONS

The staff has not determined any practical method by which the applicant can meet all the specific PSI requirements of Section XI of the ASME Code for the existing Watts Bar Nuclear Plant, Unit 1. Compliance with all the exact Section XI-required inspections would delay the startup of the plant in order for the applicant to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Even after the redesign efforts, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing facility has already been established by the construction code fabrication examinations.

On the basis of review and evaluation, the staff concludes that the public interest is not served by imposing certain provisions of Section XI of the ASME Code when the proposed alternative would provide an acceptable level of quality and safety, or when compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3), relief is authorized from these requirements for the reasons discussed herein.

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11. ABSTRACT (200 words or less)

Supplement No. 10 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. 9 was issued, and (2) matters that the staff had under review when Supplement No. 9 was issued.

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