NUREG–0847
Supplement No. 6

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

April 1991

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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), and Supplement No. 5 (November 1990) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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*Titled "Control Room Design Review" in SER.
**Titled "General" in SER.
†Titled "Conclusions" in SER.

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APPENDICES

ABBREVIATIONS

ABGTS	auxiliary building gas treatment system
ACRS	Advisory Committee on Reactor Safeguards
AI	administrative instruction
AIRD	action-information requirements detail
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BISI	bypassed and inoperable status indication
BTP	branch technical position
CAP	corrective action program
CFR	Code of Federal Regulations
CIS	containment isolation system
CNPP	Corporate Nuclear Performance Plan (NUREG-1232, Vol. 1)
CSB	Control Systems Branch
CSOC	closed system outside containment
DCRDR	detailed control room design review
DOE	Department of Energy
ECA	emergency contingency action
ECCS	emergency core cooling system
EGTS	emergency gas treatment system
EMD-GM	Electromotive Division of General Motors
EOI	emergency operating instruction
EQ	environmental qualification
ERCW	emergency raw cooling water
ERCWS	emergency raw cooling water system
ERG	emergency response guideline
ESF	engineered safety feature
FEMA	Federal Emergency Management Administration
FSAR	final safety analysis report
GDC	general design criterion
GL	generic letter
HED	human engineering deficiency
HAUP	hanger and analysis update program
HVAC	heating, ventilation, and air conditioning

IDVP	independent design verification program
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IMI	instrument maintenance instruction
ISI	inservice inspection
IST	inservice testing
LANL	Los Alamos National Laboratory
LBB	leak before break
LOCA	loss-of-coolant accident
LPMS	loose-parts monitoring system
MERITS	Methodically Engineered, Restructured, and Improved Technical Specifications
MR	maintenance request
MSLB	main steamline break
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSRS	Nuclear Safety Review Staff
NSSS	nuclear steam supply system
PDR	Public Document Room
PHMS	permanent hydrogen mitigation system
PM	preventive maintenance
PMF	probable maximum flood
PORV	power-operated relief valve
PSI	preservice inspection
PWR	pressurized-water reactor
QA	quality assurance
RBPVS	reactor building purge ventilation system
RCCA	rod cluster control assembly
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RTD	resistance temperature detector
SAIC	Science Applications International Corporation
SALP	systematic assessment of licensee performance
SER	safety evaluation report
SGTR	steam generator tube rupture
SI	safety injection
SNM	special nuclear material
SP	special program
SPDS	safety parameter display system
SSER	supplement to SER
SSI	soil-structure interaction
STS	Standard Technical Specifications

TAC	technical assignment control
TIPTOP	turbine integrity program with turbine overspeed protection
TMI-2	Three Mile Island Unit 2
TVA	Tennessee Valley Authority
UHI	upper head injection
USI	unresolved safety issue
WBNPP	Watts Bar Nuclear Performance Plan (NUREG-1232, Vols. 1 and 4)
WOG	Westinghouse Owners Group

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), and Supplement No. 5 (SSER 5, November 1990).

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

In addition to the guidance of the SRP, the staff would from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The implementation status of the applicable ones is summarized in Section 1.14.

Since SSER 4 was issued, Watts Bar licensing activities have been put on hold because of problems identified at TVA plants (see Section 1.13 for details). Thus, no supplements were issued in the ensuing five years. SSER 5 was issued in November 1990, signifying the staff's resumption of licensing activities. This supplement (SSER 6) provides more recent information regarding the resolution or status of some of the outstanding and confirmatory issues, and proposed license conditions identified in the SER and its supplements. Some of the issues addressed in previous SSERs may be subject to further review as a result of the applicant's corrective actions which are under way (see Section 1.13).

Each of the following sections or appendices of this supplement (SSER 6) is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.* Appendix E is a list of principal contributors to this supplement. Appendix G continues to note errata. This supplement made no changes in Appendices C, D, F, H, I, and J. In Appendices K, L, and N, the staff's safety evaluations of October 26, 1990; April 28, 1990; and February 11, 1991, are reproduced. In this SSER, the staff has also added Appendix M, which updates the resolution of several generic and specific seismic concerns, and Appendix O, which notes several other concerns that the applicant has satisfactorily addressed.

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

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Washington, D.C. 20555

1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. This SSER updates the status of some of those items and adds new issues. The current status of each of the issues is tabulated below and the relevant SER or SSER section is indicated. Those issues that are, to date, unresolved will be addressed in future SSERs. Outstanding Issues 18 through 22 are added in this SSER.

Issue*	Status	Section
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Inservice pump and valve test program (TAC 74801)	Updated (SSER 5)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC 71919)	Updated (SSER 6)	3.10
(b) Environmental (TAC 63591)	Under review	3.11
(5) Preservice inspection program (TAC 63627)	Under review	5.2.4, 6.6
(6) Pressure-temperature limits for Unit 2	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) BTP CSB 6-4	Resolved (SSER 3); see License Condition 8	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2

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

Issue*	Status	Section
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC 63649)	Under review (SSER 2, SSER 3)	8.2
(12) Fire protection program (TAC 63648)	Under review	9.5.1
(13) Quality classification of diesel generator auxiliary system piping and components (TAC 63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC 63638)	Resolved (SSER 3, SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC 63657)	Under review	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC 76972)	Updated (SSER 5)	17
(18) Seismic classification of cable trays and conduit (TAC R00508 and R00516)	Opened (SSER 6)	3.2.1 and 3.10
(19) Seismic design concerns (TAC 79717):		
(a) Number of OBE events	Opened (SSER 6)	3.7.3
(b) 1.2 Multi-mode factor	Opened (SSER 6)	3.7.3
(c) Code usage	Opened (SSER 6)	3.7.3
(d) Conduit damping values	Opened (SSER 6)	3.7.3
(e) Worst case, critical case, bounding calculations	Opened (SSER 6)	3.7.3
(f) Mass eccentricities	Opened (SSER 6)	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	Opened (SSER 6)	3.9.3
(i) Pressure relief devices	Opened (SSER 6)	3.9.3.3
(j) Structural issues	Opened (SSER 6)	3.8
(k) Update FSAR per 12/18/90 letter	Opened (SSER 6)	3.7, 3.8, 3.9
(20) Mechanical systems and components (TAC 79718)		
(a) Feedwater check valve slam	Opened (SSER 6)	3.9.1
(b) New support stiffness and deflection limits	Opened (SSER 6)	3.9.3.4
(21) Removal of RTD bypass system (TAC 63599)	Opened (SSER 6)	5.1
(22) Removal of upper head injection system (TAC 77195)	Opened (SSER 6)	6.3.1

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

In addition to these 22 issues, the staff has, in the 6 years since SSER 4 was published, identified a number of new issues that require resolution. However, these issues have not yet been reviewed to the degree that the staff can classify them as outstanding issues, confirmatory issues, or proposed license conditions. The status of the staff's reviews will be published in future SSERs; for the time being, these issues are tracked by the NRC WISP (Workload Information and Scheduling Program) with the following titles and TAC numbers assigned:

TAC 63597	Containment Isolation Using Closed Systems
TAC 63607	T_{SAT} Indication in the Auxiliary Control Room
TAC 63621	Main Steam Line Break Inside Containment
TAC 63632	Main Steam Line Break Outside Containment
TAC 63644	Hydrogen/Oxygen Monitoring System
TAC 63647	Health Physics Program
TAC 77550	Conformance With Regulatory Guide 1.97, Instruments To Follow Course of Accident
TAC 77553	Offsite Dose Calculation Manual
TAC 77661	Offsite Radiological Monitoring Program
TAC 77845	Containment Sump Screen Design Anomalies
TAC 77861	Operating, Maintenance and Emergency Procedures

1.8 Confirmatory Issues

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. This supplement updates the status of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The current status of each of the original issues is tabulated below, with the relevant SER or SSER section indicated. Resolution of issues that are outstanding, to date, will be addressed in future SSERs. Confirmatory Issue 43 is added in this SSER.

Issue	Status	Section
(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5) Upgrading ERCW system to seismic Category I (TAC 63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC 63618)	Resolved (SSER 5)	3.2.1

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 63625)	Updated (SSER 6)	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 63603)	Awaiting information	5.4.3
(16) Atmospheric dump valve testing	Resolved (SSER 2)	5.4.3
(17) Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC 63623)	Resolved (SSER 5)	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1
(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 63649)	Under review	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC 63649)	Under review	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2 (TAC 63649)	Under review	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC 63649)	Under review	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC 63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC 63648)	Under review	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14
(42) Submergence of electrical equipment as result of a LOCA (TAC 63649)	Under review	8.3.3.1.1
(43) Safety parameter display system	Opened (SSER 6)	18.2, App. P

1.9 Proposed License Conditions

In Section 1.9 of the SER and SSERs, 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 proposed license conditions are tabulated below, with the corresponding NUREG-0737 item number given in parentheses (as appropriate) and the relevant SER or SSER section indicated.

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 74801)	Updated (SSER 5)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TAC 77132 and 77133)	Under review	4.4.8
(4) Inservice Inspection Program (TAC 76881)	Unchanged (SSER 3)	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Resolved (SSER 5)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) noble gas monitor (TAC 63645)	Resolved (SSER 5)	11.7.1
(b) iodine particulate sampling (TAC 63645)	Resolved (SSER 6)	11.7.1
(c) high range in-containment radiation monitor (TAC 63645)	Resolved (SSER 5)	12.7.2
(d) containment pressure	Resolved (SSER 5)	6.2.1
(e) containment water level	Resolved (SSER 5)	6.2.1
(f) containment hydrogen	Resolved (SSER 5)	6.2.5
(7) Modification to chemical feedlines (TAC 63622)	Resolved (SSER 5)	6.2.4
(8) Containment isolation dependability (II.E.4.2) (TAC 63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC 77208)	Under review (SER)	6.2.5, App. C
(10) Status monitoring system/BISI (TAC 77136, 77137)	Under review (SER)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Resolved (SSER 5)	7.8.1

Proposed Condition	Status	Section
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6
(13) DC monitoring and annunciation (TAC 63649)	Under review (SSER 3)	8.3.2.2
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC 63649)	Under review (SER)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (II.B.3) (TAC 77543)	Updated (SSER 3, SSER 5)	9.3.2
(20) Fire protection program (TAC 63648)	Unchanged (SER)	9.5.1
(21) Performance testing for communications systems (TAC 63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC 63640)	Resolved (SSER 5)	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Resolved (SSER 5)	10.3.4
(24) Primary coolant outside containment (III.D.1.1) (TAC 63646)	Updated (SSER 6)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC 63592)	Under review	13.4
(26) Use of experienced personnel during startup (TAC 63592)	Under review	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC 63656)	Under review	13.3

Proposed Condition	Status	Section
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC 77861)	Under review	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC 77861)	Under review	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3
(31) Initial test program (TAC 79872)	Opened (SER)	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 63631)	Resolved (SSER 5)	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC 77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC 63655)	Updated (SSER 6)	18.1
(38) Physical Security Plan (TAC 63657)	Under review	13.6
(39) Control of heavy loads (NUREG-0612) (TAC 77560)	Updated (SSER 3)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC 64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC 77569)	Updated (SSER 3, SSER 5)	15.4.3
(42) Loose-parts monitoring system (TAC 77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC 73723 and 73724)	Opened (SSER 5)	18.2

1.11 Nuclear Waste Policy Act of 1982

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

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

1.12 Approved Technical Issues for Incorporation in the License as Exemptions

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

(1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4)
(2) Criticality monitor (Section 9.1, SSER 5) (TAC 63615)

1.13 Implementation of Corrective Action Programs and Special Programs

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

NUREG-1232, Vol. 4, documented the staff's general review of most of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847. In accordance with that commitment, this new section 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 71917)

Program review status: NUREG-1232, Vol. 4; to come.

Implementation status: Full implementation expected by September 1992.

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); to come.

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

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

Implementation status: Full implementation expected by October 1991.

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); to come.

(3) <u>Design Baseline and Verification Program (TAC 63594)</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 June 1992.

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); 50-390/90-201 (March 22, 1991); to come.

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

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

Implementation status: Full implementation expected by December 1991.

NRC inspections: Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); to come.

(5) <u>Electrical Issues (TAC 74502)</u>

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

Implementation status: Full implementation expected by March 1992.

NRC inspections: Inspection Report 50-390, 391/90-30 (February 25, 1991); to come.

(6) <u>Equipment Seismic Qualification (TAC 71919)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.2.1; review in progress.
Implementation status:	Full implementation expected by March 1992.
NRC inspections:	Inspection Report 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); to come.

(7) <u>Fire Protection (TAC 63648)</u>

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 July 1991.
NRC inspections:	To come.

(8) <u>Hanger and Analysis Update Program (TAC R00512)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3; review in progress.
Implementation status:	Full implementation expected by March 1992.
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); to come.

(9) <u>Heat Code Traceability (TAC 71920)</u>

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 to come later.
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: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3; review in progress.

Implementation status: Full implementation expected by September 1991.

NRC inspections: Inspection Report 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); to come.

(11) Instrument Lines (TAC 71918)

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 (the safety evaluation is reproduced as Appendix K in SSER 6).

Implementation status: Full implementation expected by August 1991.

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).

(12) Prestart Test Program (TAC 71924)

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: TVA expects to complete and approve test results by June 1992.

NRC inspections: Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-12 (June 19, 1990); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-17 (August 14, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-30 (February 25, 1991); to come.

(13) Quality Assurance Records (TAC 71923)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 17.3 of a future SSER.

Implementation status: Full implementation expected by January 1992.

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

(14) Q-List (TAC 63590)

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 May 1991.
NRC inspections:	Inspection Report 50-390, 391/90-08 (September 13, 1990); to come.

(15) Replacement Items Program (TAC 71922)

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 (the safety evaluation is reproduced as Appendix N in SSER 6).
Implementation status:	Full implementation expected by October 1991.
NRC inspections:	To come.

(16) Seismic Analysis (TAC R00514)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7; review in progress.
Implementation status:	Full implementation expected by April 1991.
NRC inspections:	Inspection Report 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); to come.

(17) Vendor Information Program (TAC 71921)

Program review status:	Complete: Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (the safety evaluation was reproduced as Appendix I in SSER 5).
Implementation status:	Full implementation expected by December 1991.
NRC inspections:	To come.

(18) Welding (TAC 72106)

Program review status:	Complete: Inspection Report 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 January 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); to come.

1.13.2 Special Programs

(1) Concrete Quality (TAC 63596)

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

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

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

(2) Containment Cooling (TAC 77284)

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

Implementation status: Full implementation expected by June 1991.

NRC inspections: To come.

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

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

Implementation status: Full implementation expected by August 1991.

NRC inspections: To come.

(4) Environmental Qualification Program (TAC 63591)

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 1991.

NRC inspections: To come.

(5) Master Fuse List (TAC 76973)

Program review status: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; review in progress.

Implementation status: Full implementation expected by November 1991.

NRC inspections: To come.

(6) Mechanical Equipment Qualification (TAC 76974)

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 October 1991.
NRC inspections:	To come.

(7) Microbiologically Induced Corrosion (TAC 63650)

Program review status:	NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by June 1991.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); to come.

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

Program review status:	NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by January 1992.
NRC inspections:	To come.

(9) Radiation Monitoring Program (TAC 76975)

Program review status:	Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Section 12 of the SER and SSERs.
Implementation status:	Full implementation expected by June 1992.
NRC inspections:	To come.

(10) Soil Liquefaction (TAC 77548)

Program review status:	NUREG-1232, Vol. 4; review in progress, results will be published in Section 2.5 of a future SSER.
Implementation status:	Full implementation expected by February 1991.
NRC inspections:	Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990); to come.

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

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by July 1992.
NRC inspections:	Inspection Report 50-390, 391/90-19 (October 15, 1990); to come.

1.14 Implementation of Applicable Bulletin and Generic Letter Requirements

In SSER 5, Section 1.1, the staff stated that from time to time generic requirements or recommendations are issued in the form of bulletins and generic letters. The staff committed to prepare a summary of the implementation status of the applicable ones here in SSER 6. The interim result of such effort is shown in Sections 1.14.1 and 1.14.2. Because such a long time has elapsed since these were addressed, the staff will reevaluate all bulletins and generic letters to determine if additional actions need to be taken. The staff will especially evaluate the appropriateness of implementation schedules. The evaluations will be completed before issuance of an operating license.

1.14.1 Bulletins

Bulletin 79-02, Pipe Support Base Plate Bolt Design (TAC 63625)

TVA response: Letters, J. A. Domer to NRC, August 22, 1985; E. G. Wallace to NRC, January 31, 1991.

NRC action: TVA response under review as Confirmatory Issue 9.

Implementation status: Full implementation expected before fuel load.

Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems

TVA response: Letters, L. M. Mills to NRC, November 2, 1983; J. W. Hufham to NRC, November 19, 1984.

NRC action: No plant-specific document issued.

Implementation status: Full implementation expected before fuel load.

Bulletin 79-28, Possible Malfunction of NAMCO Model EA180 Switches at Elevated Temperatures

TVA response: Letter, L. M. Mills to NRC, March 28, 1980; more later.

NRC action: Inspection Report 50-390, 391/80-15.

Implementation status: Full implementation expected before fuel load.

Bulletin 80-06, ESF Reset Control

TVA response: Letter, E. G. Wallace to NRC, December 20, 1990.

NRC action: Watts Bar SER (NUREG-0847), Section 7.3.5; Inspection Report 50-390/80-12.

Implementation status: To come.

Bulletin 80-11, Masonry Wall Design

TVA response: Letters, L. M. Mills to NRC, February 12, August 20, September 14, 1981; January 22, 1982.

NRC action: Inspection Report 50-390, 391/90-26 (January 8, 1991).

Implementation Status: To come.

Bulletin 83-07, Apparent Fraudulent Materials Sold by Ray Miller, Inc.

TVA response: Letter, L. M. Mills to NRC, March 22, 1984.

NRC action: To come.

Implementation status: Not applicable.

Bulletin 84-03, Refueling Cavity Water Seal

TVA response: Letter, J. W. Hufham to NRC, December 6, 1984.

NRC action: To come.

Implementation status: Full implementation expected before first refueling.

Bulletin 85-01, Steam Binding of Auxiliary Feedwater Pumps (also Generic Letter 88-03)

TVA response: Letter, R. L. Gridley to NRC, January 27, 1986.

NRC action: Inspection Report 50-390, 391/90-20 (September 25, 1990).

Implementation status: Complete.

Bulletin 85-02, Undervoltage Trip Attachments of DB-50 Breakers

TVA response: Letter, J. W. Hufham to NRC, December 3, 1985.

NRC action: To come.

Implementation status: To come.

Bulletin 85-03, Motor-Operated Valve Common Mode Failures

TVA response: Letter, R. H. Shell to NRC, May 12, 1986.

NRC action: Letter, G. G. Zech to S. A. White (TVA), February 2, 1988.

Implementation status: Full implementation expected by June 1994, in accordance with requirement of Generic Letter 89-10.

Bulletin 86-02, Static O-Ring Differential Pressure Switches

TVA response: Letter, R. L. Gridley to NRC, November 20, 1986.

NRC action: Inspection Report 50-390, 391/90-24 (December 17, 1990).

Implementation status: No implementation required.

Bulletin 87-01, Thinning of Pipe Walls (TAC 73561 and 73562)

TVA response:	See also response to Generic Letter 89-08; letter, R. L. Gridley to NRC, September 18, 1987.
NRC action:	Memoranda, N. Markisohn to S. C. Black, August 21, 1989 (available in PDR); P. S. Tam to Document Control Desk, June 26, 1990 (available in PDR).
Implementation status:	To come.

Bulletin 87-02, Fastener Testing To Determine Conformance With Applicable Specifications (TAC 77015 and 77016)

TVA response:	Letters, R. Gridley to NRC, December 8, 1987; March 16, 1988; April 15, July 6, September 12, 1988; January 27, 1989.
NRC action:	Letter, S. C. Black to O. D. Kingsley (TVA), August 18, 1989.
Implementation status:	To come.

Bulletin 88-01, Defects in Westinghouse Breakers (TAC 77714 and 77715)

TVA response:	Letter, R. Gridley to NRC, April 13, 1988.
Staff action:	Letter, P. S. Tam to O. D. Kingsley (TVA), October 10, 1990.
Implementation status:	Full implementation expected 6 months before fuel load.

Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes (TAC 67329 and 67330)

TVA response:	Letters, R. Gridley to NRC, March 31, 1988, and March 1, 1989; E. G. Wallace to NRC, August 16, 1990.
NRC action:	Letter, P. S. Tam to O. D. Kingsley (TVA), June 7, 1990. Inspection Report 50-390, 391/90-24 (December 17, 1990).
Implementation status:	Completed in February 1990 for Unit 1. No schedule for Unit 2.

Bulletin 88-03, GE HFA-Type Latching Relays (TAC 73955 and 73956)

TVA response:	Letters, R. Gridley to NRC, June 3, July 6, 1988.
NRC action:	Memorandum, P. S. Tam to Public Document Room, November 23, 1990.
Implementation status:	Full implementation expected by December 1990.

Bulletin 88-04, Mini-Flow Systems for Safety-Related Pumps (TAC 69991)

TVA response:	Letters, R. Gridley to NRC, February 27, 1989; M. J. Ray to NRC, June 29, 1989; E. G. Wallace to NRC, December 20, 1990.
NRC action:	Letter, S. C. Black to O. D. Kingsley (TVA), May 24, 1990.
Implementation status:	Full implementation expected by March 1991.

Bulletin 88-05, Non-Conforming Piping Material (TAC 68848 and 68849)

TVA response:	To come before Unit 1 fuel load.
NRC action:	NRC review needed before issuance of an operating license.
Implementation status:	Full implementation expected before fuel load.

Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Systems (TAC 69706)

TVA response:	Letter, M. O. Medford to NRC, August 6, 1990.
NRC action:	To come.
Implementation status:	Full implementation expected before initial criticality.

Bulletin 88-09, Thimble Tube Thinning in Westinghouse Reactors (TAC 72693 and 72694)

TVA response:	To come.
NRC action:	To come.
Implementation status:	Full implementation expected before first refueling outage.

Bulletin 88-10, Molded Case Circuit Breakers (TAC 71373 and 71374)

TVA response:	Letters, C. H. Fox to NRC, April 11, 1989; R. H. Shell to NRC, August 30, 1989; M. J. Ray to NRC, November 8, 1989.
NRC action:	Memorandum, P. S. Tam to F. J. Hebdon, September 27, 1990 (available in PDR).
Implementation status:	Full implementation expected before fuel load.

Bulletin 88-11, Pressurizer Surge Line Thermal Stratification (TAC 72181 and 72182)

TVA response: To come.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), July 16, 1990.

Implementation status: Full implementation expected before fuel load.

Bulletin 89-01, Failure of Westinghouse Steam Generator Mechanical Plugs (TAC 73220 and 73221)

TVA response: Letter, R. H. Shell to NRC, June 16, 1989.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), June 22, 1990.

Implementation status: Fully implemented (letter, E. G. Wallace (TVA) to NRC, November 13, 1990).

Bulletin 89-02, Stress Corrosion Cracking of Anchor-Darling Check Valve Bolting (TAC 74331 and 74332)

TVA response: Letter, M. J. Ray to NRC, April 25, 1990.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), June 22, 1990; Inspection Report 50-390/90-20 (September 25, 1990).

Implementation status: Fully implemented for Unit 1; Unit 2 to come.

Bulletin 89-03, Potential Loss of Required Shutdown Margin During Refueling (TAC 75474 and 75475)

TVA response: Letter, R. H. Shell to NRC, June 19, 1990.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), June 22, 1990.

Implementation status: Full implementation expected before fuel load.

Bulletin 90-01, Loss of Fill Oil in Rosemount Transmitters (TAC 76631 and 76632)

TVA response: To come.

NRC action: To come.

Implementation status: To come.

1.14.2 Generic Letters

Generic Letter 78-03, Cavity Annulus Seal Ring

TVA response: Letter, J. E. Gilleland to NRC, August 21, 1978.

NRC action: Watts Bar SER (NUREG-0847), Section 4.2.3.

Implementation status: Complete.

Generic Letters 78-16, 81-07, 83-42, and 85-11, Control of Heavy Loads (TAC 77560)

TVA response: Letters, L. M. Mills to NRC, October 28, 1981; August 25, 1982; February 6, March 14, and March 20, 1984; D. S. Krammer to NRC, September 24, 1984; J. A. Domer to NRC, January 16, 1985; January 24, 1986; more later.

NRC action: SSER 3 (Section 9.1.4); more later.

Implementation status: See proposed License Condition 39.

Generic Letter 79-03, Offsite Dose Calculation Manual

TVA response:
NRC action: The Offsite Dose Calculation Manual is being addressed under licensing action TAC 77553.

Implementation status: Per implementation schedule of TAC 77553.

Generic Letter 79-20, Cracking in Feedwater Lines

TVA response: Letter, L. M. Mills to NRC, December 1, 1983.

NRC action: Inspection Report 50-390, 391/85-08.

Implementation status: Implemented.

Generic Letter 79-25, Information Required To Review Corporate Capabilities

TVA response:
NRC action: Addressed in the Radiological Emergency Plan, which is tracked by proposed License Condition 27.

Implementation status: Full implementation before fuel load.

Generic Letter 79-36, Adequacy of Station Electric Distribution System Voltages

TVA response: Letter, L. M. Mills to NRC, October 9, 1981.

NRC action: Addressed in Watts Bar SER (NUREG-0847) and SSERs, Section 8.3.1.2; remaining actions tracked as Confirmatory Issue 28; Inspection Report 50-390, 391/84-90.

Implementation status: To come.

Generic Letter 79-40, Followup Actions Resulting From the NRC Staff Reviews Regarding the TMI-2 Accident

TVA response:
NRC action:
Generic Letter 79-40 was superseded by NUREG-0737. The staff has reviewed Watts Bar against NUREG-0737 (see Table 1.1 of the Watts Bar SER (NUREG-0847).

Generic Letters 79-46 and 79-54, Containment Purging and Venting During Normal Operation

TVA response: Letter, L. M. Mills to NRC, April 26, 1983.

NRC action: Addressed in SSER 3 and SSER 5, Section 6.2.4; actions tracked as proposed License Condition 8, which was resolved in SSER 5.

Implementation status: Not applicable.

Generic Letter 79-52, Radioactive Release at North Anna Unit 1 and Lessons Learned

TVA response: Letter, R. Gridley to NRC, March 27, 1986.

NRC action: Reviewed in Watts Bar SER (NUREG-0847) and tracked as proposed License Condition 24, which is resolved in SSER 6.

Implementation status: Full implementation when Offsite Dose Calculation Manual is issued.

Generic Letters 79-62 and 79-66, ECCS Calculations on Fuel Cladding

TVA response:
NRC action:
Superseded by GL 86-16.

Implementation status: Not applicable.

Generic Letters 79-63, 79-65, 79-67, 80-34, 80-60, 80-94, 81-04, Emergency Preparedness Plans (TAC 63656)

TVA response:
NRC action:
TVA responded to some of these generic letters, but the central issue, the plant emergency preparedness plan, is under review, and is tracked by proposed License Condition 27 (see Section 1.9 of the Watts Bar SER (NUREG-0847) and SSERs).

Implementation status: Full implementation expected before fuel load.

Generic Letters 80-02 and 83-26, Quality Assurance Requirements Regarding Diesel Generator Fuel Oil

TVA response:
NRC action:
Plant Technical Specifications are being developed, and when issued, will reflect recommendations of this generic letter where appropriate. TVA addressed the issues in FSAR Section 9.5.4.2.

Implementation status: Full implementation when Technical Specifications requirements are imposed.

Generic Letters 80-05, 80-13, 80-59, 80-82, and 84-24, Qualification of Safety-Related Electrical Equipment (TAC 63591)

TVA response:
NRC action: Plant Equipment Qualification Program is under review, and is tracked by Outstanding Issue 4(b) and special program on equipment qualification. See Section 1.7 for status.

Implementation status: Full implementation by July 1991.

Generic Letter 80-14, LWR Primary Coolant System Pressure Isolation Valves

TVA response:
NRC action: Guidance in this generic letter will be incorporated into the plant Technical Specifications, currently under development. TVA addressed this issue in FSAR 5.2.7.4.

Implementation status: Full implementation when Technical Specifications requirements are imposed.

Generic Letter 80-15, Request for Additional Management and Technical Resources Information

TVA response:
NRC action: Information only. No response needed.

Implementation status: Not applicable.

Generic Letter 80-20, Actions Required From OL Applicants of NSSS Designs by W and CE Resulting From NRC B&O Task Force Review of TMI-2 Accident

TVA response: Letter, L. M. Mills to NRC, October 28, 1981.

NRC action: Watts Bar SER (NUREG-0847), Section 10.4.9.

Implementation status: Complete.

Generic Letter 80-26 and 87-07, Qualifications of Reactor Operators

TVA response:
NRC action: Incorporated guidance in the FSAR Chapter 13. The staff has documented review results of Section 13 in the Watts Bar SER (NUREG-0847) and SSERs.

Implementation status: Complete.

Generic Letter 80-30, Clarification of the Term "Operable" as It Applies to Single-Failure Criterion for Safety Systems Required by Technical Specifications

TVA response:
NRC action: Guidance being incorporated in the plant Technical Specifications.

Implementation status: Open, pending issuance of the plant technical specifications.

Generic Letters 80-37, 80-61, 80-72, 81-10, 81-32, 82-05, 82-10, 82-33, 83-10, 83-35, 83-37, 85-12, and 89-06, TMI-2 Requirements

TVA response:
NRC action:
All TMI-2 accident requirements have been addressed in the Watts Bar SER (NUREG-0847). Incomplete ones are tracked as outstanding and confirmatory issues, or as proposed license conditions.

Implementation status: See Watts Bar SER and SSERs for individual issues.

Generic Letter 80-57, Further Commission Guidance for Power Reactor Operating Licenses Regarding NUREG-0694, "TMI-Related Requirements for New Operating Licenses"

TVA response:
NRC action:
Information only. No response needed.

Implementation status: Not applicable.

Generic Letter 80-77, Westinghouse Standard Technical Specifications

TVA response:
NRC action:
Guidance will be incorporated in the plant Technical Specifications.

Implementation status: Open, pending issuance of plant technical specifications.

Generic Letter 81-01, Quality Assurance (Qualification of Inspection, Examination, Testing and Audit Personnel) (TAC 76972)

TVA response: Letter, L. M. Mills to NRC, August 3 and August 28, 1981.

NRC action: Staff review of FSAR Section 17 is documented in the Watts Bar SER (NUREG-0847) and SSERs.

Implementation status: Complete.

Generic Letters 81-12 and 86-10, Fire Protection Requirements (TAC 63648)

TVA response:
NRC action:
All fire protection issues are covered by the corrective action program on fire protection (see Section 1.13.1) and proposed License Condition 20 (see Section 1.9).

Implementation status: See Watts Bar SSER (NUREG-0847), Section 9.5.1.

Generic Letter 81-14, Seismic Qualifications for Auxiliary Feedwater Systems

TVA response: FSAR Section 10.4.9.

NRC action: See Watts Bar SER (NUREG-0847) and SSERs, Section 10.4.9.

Implementation status: Complete.

Generic Letter 81-21, Natural Circulation Cooldown (TAC 63603)

TVA response:
NRC action: Addressed as Confirmatory Issue 15 (see Section 1.8 of the Watts Bar SER and SSERs). TVA responded by letter, M. R. Wisenberg to NRC, December 3, 1981; more to come.

Implementation status: Open.

Generic Letter 81-28, Steam Generator Overfill

TVA response:
NRC action: Information only; no response needed.

Implementation status: Not applicable.

Generic Letter 81-29, Simulator Examinations

TVA response: Letter, L. M. Mills to NRC, October 7, 1981.

NRC action: No plant-specific document issued.

Implementation status: Not applicable.

Generic Letter 82-01, New License Applications Survey

TVA response: Letter, L. M. Mills to NRC, March 9, 1982.

NRC action: No plant-specific document issued.

Implementation status: Not applicable.

Generic Letter 82-02, Commission Policy on Overtime

TVA response:
NRC action: Guidance in this generic letter will be reflected in the plant Technical Specifications.

Implementation status: Fully implemented when the Technical Specifications are in effect.

Generic Letter 82-28, Inadequate Core Cooling Instrumentation System (TAC 77132 and 77133)

TVA response:
NRC action: Initial TVA response submitted by letter, L. M. Mills to NRC, June 29, 1983. Addressed as proposed License Condition 3 (see Watts Bar SER (NUREG-0847) and SSERs, Section 1.9). Review is ongoing.

Implementation status: Open.

Generic Letters 83-01, 83-40, 85-04, 85-18, 86-14, 87-14, 88-13, 89-12, and 90-07, Operator Licensing Examination Schedule

TVA response: NRC action:	These generic letters address yearly operator examination schedules. Licensees' responses are factored into the staff's master examination schedule.
Implementation status:	Not applicable.

Generic Letters 83-28, 85-06, 85-09, and 90-03, Salem ATWS Event

TVA response: NRC action:	See Section 15.3.6 of this SSER.
Implementation status:	See documents referenced in Section 15.3.6.

Generic Letter 83-30, Deletion of an STS Surveillance Requirement During Diesel Generator Testing

TVA response: NRC action:	Plant Technical Specifications are being developed, and guidance in this generic letter will be incorporated.
Implementation status:	Not applicable.

Generic Letter 84-14, Replacement and Requalification Training Program

TVA response:	FSAR Amendment No. 61, Section 13.2.
NRC action:	See Watts Bar SER (NUREG-0847) and SSERs, Section 13.2.
Implementation status:	Complete.

Generic Letter 84-15, Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability

TVA response: NRC action:	Technical Specifications are being developed; guidance in this generic letter will be considered.
Implementation status:	Full implementation when Technical Specifications requirements are imposed.

Generic Letter 84-16, Adequacy of On-Shift Operating Experience for Applicants

TVA response:	Letter, J. A. Domer to NRC, June 12, 1985.
NRC action:	Letter, T. M. Novak to H. G. Parris (TVA), July 29, 1985.
Implementation status:	To come.

Generic Letter 84-17, Annual Meeting To Discuss Recent Developments Regarding Operator Training, Qualifications, and Examinations

TVA response: NRC action:	Information only.

Generic Letter 85-02, Staff-Recommended Actions Stemming From NRC Integrated Program for the Resolution of USIs Regarding Steam Generator Tube Integrity

TVA response:	Letter, J. A. Domer to NRC, June 17, 1985.
NRC action:	No plant-specific document issued.
Implementation status:	Full implementation expected by March 1992.

Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related

TVA response:	Letters, R. H. Shell to NRC, October 11, 1985; M. J. Ray to NRC, February 28, 1989; R. H. Shell to NRC, August 30, 1989.
NRC action:	Letter, S. A. Black to O. D. Kingsley (TVA), December 28, 1989.
Implementation status:	Full implementation expected by fuel load.

Generic Letter 85-19, Reporting Requirements on Primary Coolant Iodine Spikes

TVA response: NRC action:	Guidance will be incorporated in the plant Technical Specifications or other appropriate document.
Implementation status:	Full implementation when Technical Specifications requirements are imposed.

Generic Letter 86-04, Policy Statement on Engineering Expertise on Shift

TVA response:	Letter, R. L. Gridley to NRC, May 29, 1986.
NRC action:	To come.
Implementation status:	To come.

Generic Letters 87-04 and 87-10, FBI Criminal History Rule

TVA response:	Letter, R. Gridley to NRC, April 17, 1987.
NRC action:	No plant-specific document issued.
Implementation status:	To come.

Generic Letter 87-06, Leak Tight Integrity of Pressure Isolation Valves

TVA response:
NRC action: Guidance will be incorporated in the plant Technical Specifications.

Implementation status: Full implementation expected when Technical Specifications are in effect.

Generic Letters 87-12 and 88-17, Loss of Decay Heat Removal Capability (TAC 69792)

TVA response: Letters, R. Gridley to NRC, October 2, 1987, and January 6, 1989; R. Gridley to NRC, February 2, 1989; M. J. Ray to NRC, May 31, 1989.

NRC action: Letters, S. D. Richardson to O. D. Kingsley (TVA), December 5, 1988; P. S. Tam to O. D. Kingsley (TVA), June 19 and October 2, 1990.

Implementation status: Full implementation expected by March 1991.

Generic Letter 88-02, Integrated Safety Assessment Program II

TVA response: Letter, R. Gridley to NRC, March 15, 1988.

NRC action: No plant-specific document issued.

Implementation status: Not applicable.

Generic Letter 88-03, Steam Binding of Auxiliary Feedwater Pumps (TAC R00378)

TVA response: Letter, R. Gridley to NRC, June 3, 1988.

NRC action: Letter, S. C. Black to S. A. White (TVA), July 20, 1988.

Implementation status: Complete.

Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (TAC 77157 and 77158)

TVA response: Letter, R. Gridley to NRC, June 1, 1988.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), August 8, 1990.

Implementation status: Full implementation expected before fuel load.

Generic Letter 88-11, Radiation Embrittlement of Reactor Vessel Materials, RG 1.99 Rev. 2 (TAC 71567 and 71568)

TVA response: Letter, R. Gridley to NRC, December 9, 1988.

NRC action: Letter, S. C. Black to O. D. Kingsley (TVA), June 29, 1989.

Implementation status: Full implementation expected before fuel load.

Generic Letter 88-12, Removal of Fire Protection Requirements From Technical Specifications

TVA response:
NRC action: Guidance in this generic letter will be addressed during development of the plant Technical Specifications.

Implementation status: Not applicable.

Generic Letter 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment (TAC 71738 and 71739)

TVA response: Letters, R. Gridley to NRC, February 23 and May 12, 1989; E. G. Wallace to NRC, July 12, 1990.

NRC action: Letter, P. S. Tam to O. D. Kingsley (TVA), July 26, 1990.

Implementation status: Full implementation expected during hot functional testing.

Generic Letter 88-20, Individual Plant Examinations for Severe-Accident Vulnerabilities (TAC 74488)

TVA response: Letters, M. J. Ray to NRC, October 30, 1989; E. G. Wallace to NRC, May 25, 1990.

NRC action: Letters, S. C. Black to O. D. Kingsley (TVA), November 9, 1989, and January 12, 1990.

Implementation status: Expect TVA to complete evaluation by September 1992.

Generic Letter 89-04, Guidance on Developing Acceptable In-service Testing Programs (TAC 74801)

TVA response:
NRC action: TVA committed to submit an in-service test program, in accordance with guidance of this generic letter, 6 months before issuance of the operating license. This is tracked as Outstanding Issue 3.

Implementation status: Open.

Generic Letter 89-07, Safeguards Contingency Plan for Surface Vehicle Bombs (TAC 74748)

TVA response: Letter, E. G. Wallace to NRC, October 31, 1989.

NRC action: Publicly available internal memorandum, P. S. Tam to Document Control Desk, June 26, 1990.

Implementation status:	Full implementation expected 2 months before fuel load.

<u>Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning (TAC 73561 and 73562)</u>

TVA response:	Letter, R. H. Shell to NRC, July 19, 1989.
NRC action:	Publicly available internal memorandum, P. S. Tam to Document Control Desk, June 26, 1990.
Implementation status:	Full implementation expected by fuel load.

<u>Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance (TAC 75736 and 75737)</u>

TVA response:	Letter, M. J. Ray to NRC, December 21, 1989.
NRC action:	Letter, F. J. Hebdon to O. D. Kingsley (TVA), September 14, 1990.
Implementation status:	Full program implementation expected before fuel load.

<u>Generic Letter 89-13, Service Water System Problems (TAC 74082 and 74083)</u>

TVA response:	Letter, M. J. Ray to NRC, January 26, 1990.
NRC action:	Letter, P. S. Tam to O. D. Kingsley (TVA), July 9, 1990.
Implementation status:	Full implementation expected before hot functional testing.

<u>Generic Letter 89-19, Safety Implication of Control Systems (TAC 75017 and 75018)</u>

TVA response:	Letter, M. J. Ray to NRC, March 22, 1990.
NRC action:	Letter, P. S. Tam to O. D. Kingsley (TVA), October 24, 1990.
Implementation status:	Complete.

<u>Generic Letter 90-06, (1) Power-Operated Relief Valve and Block Valve Reliability, (2) Additional Low-Temperature Overpressure Protection for Light-Water Reactors (TAC 77393, 77394, 77469, 77470)</u>

TVA response:	Letter, E. G. Wallace to NRC, December 21, 1990.
NRC action:	Letter, P. S. Tam to O. D. Kingsley (TVA), January 9, 1991.
Implementation status:	Guidance will be incorporated in the Technical Specifications.

Generic Letter 90-08, Simulation Facility Certification

TVA response: Letter, E. G. Wallace to NRC, February 28, 1991.

NRC action: To come.

Implementation status: To come.

Generic Letter 90-09, Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions

TVA response: / NRC action: Not needed. Action is voluntary for operating reactor licensees. For Watts Bar, the guidance of this generic letter will be incorporated in the Technical Specifications.

Implementation Status: Not applicable.

Generic Letter 91-01, Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications

TVA response: / NRC action: Not needed. Action is voluntary for operating reactor licensees. For Watts Bar, the guidance of this generic letter will be incorporated in the Technical Specifications.

Implementation Status: Not applicable.

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

The staff has reviewed the FSAR through Amendment No. 64. The review work in this section was tracked by TAC 77325. The evaluation below (Sections 3.2.1 through 3.10) was transmitted to the applicant (letter, P. S. Tam to O. D. Kingsley (TVA), January 4, 1991).

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

The staff identified an issue regarding the seismic classification of structures, systems, and components at Watts Bar. In Amendment No. 50, the applicant incorrectly applied position 3 of Regulatory Guide 1.29 by seismically qualifying mechanical systems comprising portions that are Category I and portions not seismically qualified through the second change of direction beyond the defined boundary (such as a valve). Regulatory Guide 1.29, position 3, states that the seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that share boundaries between seismic Category I and non-seismic Category I should be designed to seismic Category I requirements. Subsequently, in Amendment No. 64, the applicant revised the seismic classification to agree with position 3. Therefore, the staff finds this revision acceptable.

The staff has also identified an issue regarding the seismic classification of the safety-related conduits and cable trays. The cable trays and conduit are designated by TVA as "seismic Category I(L)" (limited structural integrity) and are only designed and constructed to preclude failure which could reduce the ability of Category I structures, systems, or components to perform their intended function. Thus there are no seismic Category I cable trays and conduit at Watts Bar. However, the supports for safety-related cable trays and conduit in Category I structures are designated as seismic Category I.

The staff does not find TVA's safety classification and seismic qualification of cable trays and conduit acceptable. Regulatory Guide 1.29, position C.1.q clearly states that Class 1E electrical systems are to be designated as seismic Category I. "Systems" include the cable trays, conduit, supports, and switchgear, not just the cable. Furthermore, the NRC Standard Review Plan (NUREG-0800, dated July 1981), Section 3.7.2, states that non-Category I structures are to be analyzed and designed to prevent their failure under safe shutdown earthquake (SSE) conditions in such a manner that the margin of safety of these structures is equivalent to that of Category I structures. TVA's approach to classify its cable trays and conduit as seismic Category I(L) is considered a newly identified outstanding issue (Outstanding Issue 18). The staff communicated this concern to the applicant (letter from P. S. Tam to O. D. Kingsley (TVA), November 29, 1990).

3.2.2 System Quality Group Classification

3.2.2.1 Class A

The applicant defines class A quality standards that are required for pressure-containing components of the reactor coolant pressure boundary as reactor coolant pressure boundary components whose failure could cause a loss of reactor coolant which would not permit an orderly reactor shutdown and cooldown assuming that makeup is only provided by the normal makeup system. Branch piping 3/8-inch (inside) diameter (ID) and smaller, or protected by a 3/8-inch diameter or smaller orifice, is exempted from class A. The applicant has also stated that branch piping for the pressurizer steam space instrumentation nozzles (0.83-inch ID) is also exempted from class A.

The staff reviewed the applicant's basis for the premise that a break in the steam space can be made up with normal charging. As part of an audit held at the site between November 5 and 9, 1990, the staff reviewed a letter sent by Westinghouse to TVA, "Pressurizer Class Breaks," WAT-D-6345, containing TVA Calculation No. NEB 850118604, and dated January 18, 1985, which determined the maximum steam leakage from the pressurizer at 2250 psi to the containment through a 0.83-inch ID instrumentation nozzle. The results of the calculation indicated that the normal makeup system can provide an equivalent makeup flow rate and that a flow restrictor is not required in the pressurizer steam space instrumentation nozzle. On the basis of its review, the staff concludes that the calculation in the Westinghouse letter provides adequate justification for exempting the pressurizer steam space instrumentation nozzles from the class A quality group, and the quality group classification of the pressurizer steam space instrumentation nozzles is acceptable.

3.2.2.5 Relationship of Applicable Codes to Safety Classification for Mechanical Components

The applicant has described the use of paragraphs from editions and addenda of Section III of the American Society of Mechanical Engineers' Boiler and Pressure Vessel Code (ASME Code) that are later than the code of record for the application. In Amendment No. 64, the FSAR was revised to describe the controls TVA places on the use of later editions and addenda of the ASME Code as it relates to the design of components for which TVA is the designer. The applicant's controls ensure that later editions and addenda have been incorporated by reference into 10 CFR 50.55a and that all related requirements necessary to support the use of specific paragraphs in later editions and addenda are met in accordance with ASME Code paragraph NA-1140. The use of later provisions of the ASME Code is permitted by paragraph NA-1140 of the 1971 edition with addenda through summer 1973 (code of record). On this basis, the staff finds that the applicant's use of later paragraphs of the ASME Code is acceptable.

The use of ASME Code cases for the design or evaluation of plant components are required to be approved by the staff on a case-by-case basis. Any additional requirements or limitations shall be satisfied in accordance with Regulatory Guide 1.84 or Regulatory Guide 1.85.

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

In Amendment No. 64, FSAR Section 3.6A.2.1.2 was revised so that circumferential ruptures and longitudinal splits are no longer postulated by the applicant to satisfy a minimum number of intermediate breaks in high-energy class 1, 2, and 3 piping and piping containing high and moderate energy interfaces. The applicant proposes to eliminate from design considerations those breaks generally referred to as "arbitrary intermediate breaks," which are defined as those break locations which, based on piping stress and analysis results, are below the stress and fatigue limits specified in Branch Technical Position (BTP) MEB 3-1 (Revision 1) but are selected to provide a minimum of two postulated breaks between the terminal ends of a piping system. The FSAR change is consistent with Revision 2 to the BTP MEB 3-1 of Standard Review Plan (SRP) Section 3.6.2 in accordance with Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," dated June 19, 1987, and is thus acceptable to the staff.

The staff has identified an issue regarding the determination of intermediate break locations based on high stress limits. In FSAR Section 3.6A.2.1.2 (item 1A), the applicant has established a pipe stress limit of 3.0 S_m for the stress intensity range (S_n) as a criterion for postulating intermediate break locations for high energy Class 1 piping runs. This limit is consistent with the 1971 edition of the ASME Code, Section III, paragraph NB-3653.1, in which S_n is calculated according to equation 10 which sums stresses due to pressure, thermal and earthquake cyclic moments, gross structural or material discontinuity, and a linear thermal gradient (delta T1). Because the applicant's equation does not include a factor for delta T1, a pipe rupture limit of 2.4 S_m should be followed to account for the lower S_n value consistent with SRP, Revision 1, dated July 1981. In a letter from O. D. Kingsley to the NRC, dated December 18, 1990, TVA agreed to either include the delta T1 term in equation 10 or reduce the break postulation limit to 2.4 S_m. The staff finds that TVA's commitment adequately resolves this issue consistent with the guidelines of the SRP, and is thus acceptable.

The applicant has given the staff information regarding the analysis of jet impingment loads from postulated breaks. In FSAR Section 3.6A.1.1.2, test data and analysis developed in NUREG/CR-2913, "Two Phase Jet Loads," dated January 1983, are used to establish the criterion that unprotected components located more than 10 diameters from a pipe break are without further analysis assumed undamaged by a jet of steam or subcooled liquid that flashes at the break. The staff has previously reviewed the methodology used in NUREG/CR-2913 for determining the effects of such a jet on components at a distance greater than 10 diameters and has found it acceptable. Similar application of this criterion has been approved for other plants and is, therefore, acceptable for Watts Bar.

3.7 Seismic Design

This section was added to the FSAR to describe set A, set B, and set C seismic analyses (explained below). The original analyses and design of seismic Category I structures were performed in accordance with "set A criteria," the original design-basis criteria for Watts Bar.

In response to issues identified between 1987 and 1989, seismic reanalyses of certain structures were performed. Evaluations of these existing structures used the site-specific response spectra (SSRS) developed for Watts Bar and in conformance with the current Standard Review Plan (SRP, NUREG-0800) criteria. The criteria used for this evaluation are called "set B criteria." SRP Revision 1 (1981), as updated according to the provisions of SRP Revision 2 (1989), formed the basis for the set B analysis. Specific evaluations for soil-supported structures were performed for: (1) the requirement of varying the soil shear modulus by +100 percent and -50 percent from the best estimate and (2) the limiting of the hysteretic soil damping ratio to a maximum value of 15 percent. Although these requirements have not been incorporated in Amendment No. 64 of the FSAR, TVA submitted the marked-up copy of the related pages of the FSAR changes (letter from E. G. Wallace to the NRC, December 18, 1990). The seismic responses, including the amplified response spectra obtained from the set B analysis, are only to be used for evaluating existing seismic Category I structures, systems, and components, and for validating the existing design calculations.

To develop seismic loads for new designs and modifications, the Category I structures evaluated to set B criteria were reananlyzed using the original criteria with upgraded seismic models, including soil-structure interaction. This analysis is called "set C analysis." As discussed in the letter from E. G. Wallace (TVA) to the NRC, dated May 9, 1990, set C analysis does not stand by itself. The purposes of this analysis are to calculate structural responses, which represent the results based on the original design basis with the structural model upgrading, and to combine set C analysis results with the set B results.

The envelope of the seismic responses from the set B and set C analyses are to be used for the reanalysis of Category I piping systems, and all new designs and modifications of Category I structures, systems, and components.

For certain structures, TVA identified no seismic issues during 1987-1989. Therefore, TVA did not perform set B and set C analyses for these structures. However, TVA stated that if these structures need to be evaluated in the future, such evaluations will use set B criteria. According to TVA, the underground electrical concrete conduit banks were being evaluated to the set B criteria. However, Amendment No. 64 states that set B and set C analyses were not performed for the underground electrical concrete conduit banks. In its December 18, 1990, letter, TVA committed to revise the FSAR to be consistent with the design calculations completed for Watts Bar.

The addition of this section is consistent with the applicant's commitments made in the seismic corrective action program (CAP) plan for Watts Bar, Revision 2, (letter from E. G. Wallace (TVA) to the NRC, May 9, 1990) which the staff approved in Inspection Report (IR) 50-390, 391/89-21, dated May 10, 1990, and, therefore, is acceptable.

3.7.1 Seismic Input

3.7.1.1 Ground Response Spectra

FSAR Section 3.7.1.1 specifies the ground response spectra for use as the seismic input motions for the original design (set A), reevaluation (set B), and new design or modification (sets B+C) seismic analyses of Category I structures, components, and systems. Ground response spectra for the set A analysis are the modified Newmark response spectra specified previously in the FSAR. In Section 3.7.1.1.1 of the revised FSAR, the modified Newmark response spectra are redesignated as the "original site response spectra" and are shown in Figures 3.7-1 through 3.7-4 for damping ratios of 0.5 percent, 1 percent, 2 percent, and 5 percent, respectively. The corresponding peaks of the ground acceleration (or zero period accelerations) for the operating basis earthquake (OBE) are 0.09 g and 0.06 g, horizontal and vertical motions, respectively. Section 3.7.1.1.1 of the revised FSAR specifies that the same original site response spectra be the seismic input motion for the set C analysis. This criterion is consistent with the one specified in the seismic CAP plan, Revision 2, which the staff approved as stated above, and in conformance with SRP requirements. Therefore, the changes made in this section are acceptable.

FSAR Section 3.7.1.1.2 specifies the site-specific response spectra that are to be the seismic input criteria for the set B analysis. The staff accepted the site-specific response spectra in its June 1982 safety evaluation report (SER, NUREG-0847). The associated peaks of the ground acceleration for the horizontal and vertical components of the site-specific safe shutdown earthquake (SSE) are 0.215 g and 0.18 g, respectively, and the corresponding peaks of the ground acceleration for the site-specific OBE are 0.09 g and 0.06 g, respectively. Using the site-specific response spectra as input motion for the set B analysis is acceptable because it is consistent with the criterion specified in the seismic CAP plan, Revision 2 (mentioned above).

On the basis of the evaluation discussed above, the staff concludes that Section 3.7.1.1 of the FSAR, as supplemented by the revisions committed to in TVA's December 18, 1990, letter, is acceptable.

3.7.1.2 Design Time Histories

FSAR Section 3.7.1.2 specifies the artificial ground motion (acceleration) time histories that are compatible with the ground response spectra described in Section 3.7.1.1 of the revised FSAR. In its 1982 SER, the staff accepted the four artificial acceleration time histories developed for the original site response spectra. Figures 3.7-1 through 3.7-4 compare the averaged OBE response spectra obtained from the four artificial acceleration time histories for 0.5 percent, 1 percent, 2 percent, and 5 percent damping, respectively, with the original ground response spectra for the site.

These four spectrum plots duplicate the plots contained in the previous FSAR, and they start at the period of 0.05 second. However, this starting period for spectrum plots is different from the 0.03-second starting period listed in Table 3.7-1 of the revised FSAR. In addition, as shown in Figures 1 through 4 of TVA Design Criteria Document WB-DC-20-24, Revision 5, other than the ratio of 2 between the SSE and OBE, the OBE-averaged spectra in the FSAR differ in shape from the SSE-averaged spectra at both short- and long-period ends of the spectra. In its December 18, 1990 letter, the applicant stated that the SSE-averaged spectra in WB-DC-20-24, Revision 5, contain some plotting errors, and these errors will be corrected. In the same letter, TVA also committed to replace FSAR Figures 3.7-1 through 3.7-4 with the corrected spectrum plots. Because the SSE-averaged spectra in WB-DC-20-24 start at a period of 0.03 second, this TVA commitment will simultaneously resolve the discrepancies in: (1) the starting period between FSAR Table 3.7-1 and FSAR Figures 3.7-1 through 3.7-4 and (2) the shape of the averaged spectra between the revised FSAR and WB-DC-20-24.

The applicant developed three components for the artificial ground motion (acceleration) time history for the SSE site-specific response spectra, as specified in Section 3.7.1.2.2 of the revised FSAR. Two of these histories represent the horizontal components and one represents the vertical component of the ground motion. The SRP requires that the three components of the artificial time history be statistically independent of each other and that their response spectra envelope the site-specific response spectra for all damping ratios to be used in the seismic analyses (analyses of structures, systems, and components). In addition, because only a single set of time-history components was developed, the SRP requires that the power spectrum density function (PSDF) of each time-history component envelopes the 80-percent level of the target PSDF within the frequency range of interest. The FSAR states that all three components of the artificial time history satisfy the statistical independence and response spectrum enveloping requirements of the SRP, and that FSAR Figures 3.7-4a through 3.7-4c show a comparison of the 7-percent damping response spectrum of each time history to the site-specific response spectrum. FSAR Figures 3.7-4d through 3.7-4f also show a comparison of the PSDF of each time-history component with the corresponding 80-percent level of the target PSDF. The PSDF of the time-history components envelopes the 80-percent level of the target PSDF throughout the frequency range of 0.3 to 33 Hz, except for a slight local dip at low frequencies of from 0.5 to 0.7 Hz for the second horizontal component (H2) and from 0.40 to 0.42 Hz and from 1.2 to 1.6 Hz for the vertical component. In accordance with IR 50-390, 391/89-21, the staff previously reviewed TVA Calculation B26-890427-012 and concluded that the three time-history components satisfy the SRP requirements of statistical independency and spectrum enveloping for the damping ratios of 1 percent, 2 percent, 3 percent, 4 percent, 5 percent, and 7 percent. Figures 3.7-4a through 3.74c of the revised FSAR are adopted from this TVA calculation. The deficiency in the FSAR is that it does not present the results of spectrum comparison for damping ratios other than 7 percent. To resolve this deficiency, the applicant committed to include in the next FSAR revision the spectrum comparison results from TVA Calculation B26 890427 012 for the damping ratios of 1 percent, 2 percent, 3 percent, 4 percent, 5 percent, and 7 percent (letter from E. G. Wallace (TVA) to the NRC, December 18, 1990). The staff found the TVA commitment to be an adequate resolution. FSAR Figures 3.7-4d through 3.7-4f, which show the PSDF comparison, are adopted from TVA Calculation B26 890929 100. The staff previously reviewed this TVA calculation and considered the slight local dip on the PSDF of the H2 and vertical time-history components at certain low frequencies to be inconsequential because the response spectra still envelope the target spectra at these frequencies and because such low frequencies are outside the general frequency range of interest. Thus, on the basis of the discussion in IR 50-390, 391/89-21, the staff concluded that the three components of the site-specific artificial time history also satisfy the SRP requirement for PSDF matching.

On the basis of the evaluation discussed previously, the staff concludes that FSAR Section 3.7.1.2, as supplemented by the revisions that the applicant committed to in its December 18, 1990, letter, is acceptable.

3.7.1.3 Critical Damping Values

FSAR Tables 3.7-2, 3.7-2A, and 3.7-2B present the damping values used for seismic analyses of Category I structures, systems, and components. Although these tables include damping values for soil, structures, systems, and components, this section only covers the technical evaluation of damping values used in structures. For damping values used in soil, this report provides technical evaluations in the appropriate sections relating to the seismic analysis of soil-supported structures. Similarly, the damping values used for systems and components are covered in sections relating to those items.

Tables 3.7-2A and 3.7-2B were included in FSAR Amendment No. 64 inadvertently. The applicant has committed, in its December 19, 1990, letter, to delete these two tables from this amendment. Also, the applicant has committed to revise the text of FSAR Amendment No. 64 to eliminate any references to these two tables.

FSAR Table 3.7-2 shows that the damping values for structures used for the set C analysis are the same as those used for the set A analysis. Because set A criteria represent the original design criteria and have already been approved by the NRC, the staff accepts the damping values used for structures in set C analyses. In Table 3.7-2, the applicant also proposed to use the damping values specified in Regulatory Guide (RG) 1.61 for set B analyses. As discussed in IR 50-390, 391/89-21, the use of RG 1.61 damping values is acceptable to the staff for set B analyses.

The set A analysis was never performed for the additional diesel generator building (ADGB), because the design was completed after the issuance of the SER. Although the applicant performed a set B analysis, the damping values used, which are consistent with the RG 1.61 requirements, are not shown on Table 3.7-2. In its December 19, 1990, letter, the applicant committed to revise Table 3.7-2 to include the damping values used (4% for OBE and 7% for SSE) for the ADGB set B analysis. For the set C analysis, 5-percent damping was used for both OBE and SSE, which is the same damping value specified for "other concrete structures" in the previous FSAR. This damping value is also acceptable to the staff because this damping value is consistent with RG 1.61 criteria for SSE and for OBE. This damping is a slightly higher value but the enveloping requirement for loads derived from set B and set C calculations should compensate for the effect of the damping value.

3.7.1.4 Supporting Media for Seismic Category I Structures

The values of shear wave velocity used in the structural seismic response calculations for either soil or rock foundation materials are presented in a revised Table 3.7-3 for the set A calculations. These values correspond to the values previously accepted in the SER as appropriate for the wave speeds of these foundation materials. For the set B and set C calculations, values of shear wave velocity were generated from soil column analyses (SHAKE-type computations) for the specific foundation configurations under each structure as well as for the input ground motions specified at the top of bedrock in these analyses. The use of the SHAKE computer code to calculate the strain-dependent soil properties is acceptable to the staff, because this computer code has been used to license other nuclear plants. A range of values of shear moduli was computed for each soil type in the soil column to account for the variation in properties (upper bound, best estimate, and lower bound). These analyses considered initial low strain values and degradation with strain level appropriate for these soil types which accounted for depth in the soil column and the results of field and laboratory data. These approaches are considered acceptable for the set B and set C calculations because they agree with the guidelines of the SRP.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Category I Rock-Supported Structures--Original Analyses (Set A)

A number of modifications have been made to the general description of this FSAR section. They are all relatively minor changes in wording. These changes are considered acceptable because they do not alter the content of the paragraph but only serve to clarify the text.

<u>Shield Building</u>

This section of the FSAR presents a discussion of the original (set A) seismic analysis of the shield building. The analysis proposed by the applicant did not involve a technical change to the FSAR.

<u>Interior Concrete Structure</u>

This FSAR section, discussing the set A seismic analysis of the interior concrete structure, does not require technical change. However, the elastic modulus of concrete as shown in Table 3.7-6 is in error because it is different from the value specified in Table 3 of the applicant's seismic design criterion WB-DC-20-24, Revision 5. The applicant has agreed to correct the elastic modulus of concrete shown in FSAR Table 3.7-6 to be consistent with WB-DC-20-24, Revision 5. The staff reviewed the applicant's commitment and found it acceptable.

<u>Steel Containment Vessel</u>

This section of the FSAR discusses the set A seismic analysis of the steel containment vessel. Because the FSAR has been reorganized, the text, Tables 3.7-5A and 3.7-5B, and Figures 3.7-7B and 3.7-7C simply duplicate the corresponding information in Section 3.8.2.4 of the previous FSAR. Therefore, this topic does not require technical change in the revised FSAR. However, it is not clear to the staff how the two different sets of mass eccentricities listed in Table 3.7-5B were utilized in the set A seismic analysis. The applicant confirmed that the first set of mass eccentricities was used in the production analysis, and that the second set of mass eccentricities (shown in the last column of Table 3.7-5B) was utilized to study the sensitivity of the response of containment to the accidental torsion resulting from an assumed mass eccentricity equal to approximately 5 percent of the diameter of the containment. This accidental eccentricity was also considered in the final design calculations. For the purposes of clarification and consistency, the applicant committed, in its December 18, 1990, letter, to delete the second set of mass eccentricities from Table 3.7-5B. The staff found this corrective action acceptable.

<u>North Steam Valve Room</u>

This section of the FSAR discusses the set A seismic analysis of the north steam valve room. It duplicates the previous FSAR wording except for the deletion of the text referencing another section of the FSAR regarding the soil spring calculation procedure which was used in set A calculations to account for the soil-structure interaction effects. Since there is no justification for such deletion, the applicant agreed, in its December 18, 1990, letter, to revise the FSAR by cross-referencing Section 3.7.2.1.3 for the soil spring calculation procedure for the north steam valve room. The staff found the corrective action taken by the applicant acceptable.

3.7.2.1.2 Category I Rock-Supported Structures--Evaluation and New Design or Modification Analyses (Set B and Set B+C)

This section was added to the FSAR to describe the set B and set B+C seismic analysis performed for the Category I rock-supported structures. SRP Revision 1 (1981), updated to the provisions of SRP, Revision 2 (1989), formed the basis for the set B and set B+C criteria. Specific evaluations were performed for: (1) the requirement of varying the soil shear modulus by +100 percent and -50 percent from the best estimate and (2) limiting the hysteretic soil damping ratio to a maximum value of 15 percent. Although this statement was not included in Amendment No. 64 to the FSAR, TVA committed, in its December 18, 1990, letter, to include it in the next revision of the FSAR.

In the fifth paragraph on page 3.7-7a, Amendment No. 64 made reference to Table 3.7-2B for the structure damping values in set B and set C analyses. As previously discussed in Section 3.7.1.3, the applicant committed to delete Tables 3.7-2A and 3.7-2B and to revise Table 3.7-2. To be consistent with this corrective action, the applicant also committed to replace any reference in the FSAR to Table 3.7-2B with a reference to Table 3.7-2. This commitment is acceptable to the staff. Staff review of the applicant's set B and set B+C analyses for each individual rock-supported Category I structure is discussed in the material that follows.

Reactor Building*

For the reactor building, rock-structure interaction was included in the seismic analysis using the SASSI computer code. The use of the SASSI computer code is acceptable to the staff (letter from S. Black (NRC) to O. D. Kingsley (TVA), October 31, 1989). Tables and figures illustrating the properties and configurations of the individual structure models are:

- shield building--Table 3.7-4A, Figure 3.7-5A
- interior concrete--Tables 3.7-6A and 3.7-6B, Figures 3.7-8A and 3.7-8B
- steel containment--Table 3.7-5C, Figure 3.7-7A

Except for the vertical modeling of the dome, the structural models for both the shield building and steel containment vessel are essentially the same as those in the set A analysis. A completely new three-dimensional model was developed for the interior concrete. The structural modeling techniques, as described in the FSAR and the tables/figures listed above, are consistent with those contained in the TVA calculations which have previously been reviewed and accepted by the staff in IR 50-390, 391/89-21. Staff review of the FSAR identified two concerns, however. The first concern is the statement in the FSAR that, for the shield building, the beam element properties for the set B/set C structural model are the same as those used in the set A analysis. This contradicts Table 3 of TVA seismic design criteria document WB-DC-20-24, Revision 5, which shows

*Including shield building, interior concrete, and steel containment vessel.

that the concrete modulus in the set A analysis differs from that used in set B/set C analyses. Because the set B and set C calculations were based on WB-DC-20-24, the applicant committed, in its December 18, 1990, letter, to revise the FSAR statement to be consistent with WB-DC-20-24, Revision 5. The staff found the applicant's corrective action sufficient to resolve the first concern. The second concern is the FSAR statement that except for the single-degree-of-freedom vertical dome model, the model configuration, lumped masses, and elastic beam element properties for the steel containment vessel are the same as those used in the set A analysis. The set A analysis model as shown in Table 3.7-5B includes mass eccentricities although, as discussed previously, the applicant committed, in its December 18, 1990, letter, to delete the last column in Table 3.7-5B for the purpose of clarification. However, this change appears to contradict the statement in the FSAR that: "The dynamic model for the SCV set B and set C analyses is represented by a 3-D [three-dimensional] lumped-mass, concentric single-stick model," which implies that mass eccentricities were excluded from the set B and set C analysis model. Table 3.7-5C, in which the mass and member properties of the model are shown, does not show any mass eccentricities either. The applicant should verify whether or not mass eccentricities were actually included in the set B and set C analysis model for the steel containment. Therefore, the second staff concern remains unresolved and will be tracked as Outstanding Issue 19(f).

Auxiliary Control Building

For the set B and set C analyses, the auxiliary control building (ACB) was represented by a three-dimensional, lumped-mass model with a fixed base as depicted in FSAR Figure 3.7-9A. The stiffness and mass properties were unchanged from the original set A analysis except for the concrete shear modulus. These properties are listed in FSAR Tables 3.7-9 and 3.7-10. To account for torsional effects, the eccentricities of the center of mass and center of rigidity were included in the model. The centers of mass and rigidity are as shown in FSAR Table 3.7-9A. An additional eccentricity equal to 5 percent of the maximum building plan dimension was used to calculate the torsional moments that result from accidental eccentricity. The staff confirmed that the information contained in the figures and tables mentioned in this paragraph were the same as shown in the TVA document with RIMS No. B26 89-0427-033. The staff reviewed this document (IR 50-390, 391/89-21) and found that the consideration of the torsional effects, including the accidental eccentricity, is consistent with the SRP guidelines.

The time-history analyses for set B criteria were based on structural damping values for concrete structures of 4 percent for OBE and 7 percent for SSE. The statistically independent north-south and east-west components of ground-motion time history were applied simultaneously to the horizontal model. Similarly, vertical time-history analysis was performed on the vertical model. Structural responses and amplified response spectra (ARS) were computed by combining the horizontal and vertical directions using the square-root-of-the-sum-of-the-squares (SRSS) method. ARS were obtained for both OBE and SSE, since the structural damping values were different.

The time-history analyses for set C criteria were based on set A structural damping of 5 percent for both OBE and SSE. The structural responses and ARS for OBE were computed by combining the two horizontal and vertical responses using the SRSS method. The responses for the SSE were obtained by multiplying the OBE results by a factor of two. This is acceptable since the damping value is the same in this case.

The changes in analysis criteria stated in this section are in accordance with the revised CAP on seismic analysis and TVA Design Criteria WB-DC-20-24, which were both accepted by the staff, as stated in IR 50-390, 391/89-21. Therefore, the FSAR revision discussed in this section is acceptable to the staff.

Essential Raw Cooling Water Intake Pumping Station

For set B and set C analyses, the lumped-mass model of the intake pumping station (IPS) was revised from the original set A analysis. To account for torsional effects, eccentricities between the centers of mass and rigidity were included in the set B and set C analyses. Since the IPS is supported on rock, the lumped-mass model was fixed at the base. Horizontal soil springs to account for the embedments were not included in these analyses since the addition of such springs were found to have a negligible effect on the natural frequency on the IPS. Also, the highest water level in the IPS was considered in the analyses, since the difference in the fundamental horizontal and vertical frequencies resulting from variations in the water level were insignificant.

The time-history analyses and the generation of ARS for set B and set C were performed in accordance with the method described for the auxiliary control building.

The criteria changes described in this section are in accordance with the revised CAP on seismic analysis (letter from E. G. Wallace (TVA) to the NRC, May 9, 1990) and TVA Design Criteria WB-DC-20-24, which were both reviewed and accepted by the staff. Therefore, the criteria changes for the seismic analyses of the IPS are acceptable.

North Steam Valve Room

A new structural model was developed for the set B and set C analyses, and rock-structure interaction effects were accounted for using the SASSI computer code. The analysis method and model properties as given in Tables 3.7-13A and 3.7-13B, and model configuration as shown in Figures 3.7-10A and 3.7-10B, are based on TVA calculations that were previously reviewed and accepted by the staff in IR 50-390, 391/89-21. This revised section of the FSAR is, therefore, acceptable.

On the basis of the evaluations discussed previously, Section 3.7.2.1.2 of the FSAR, as supplemented by the FSAR revisions, committed to in TVA's December 18, 1990, letter, is acceptable.

3.7.2.1.3 Category I Soil-Supported Structures--Original Analyses (Set A)

In the introductory paragraph to this section of the FSAR, several editorial changes have been made which clarify the text without changing the intent of the descriptions. They are considered acceptable since they are editorial only. Other changes made in this section of the FSAR are summarized below.

Diesel Generator Building

The changes presented in the description of the soil foundation under the diesel generator building serve to make it conform to the actual conditions existing under the structure. The remainder of the modifications to this section are editorial. These are all considered acceptable since they clarify the description.

Waste Packaging Area

The modifications made in the descriptions of the waste packaging area and refueling water tank and emergency raw cooling water (ERCW) pipe tunnels are again considered acceptable since they are only editorial, and clarify the descriptions of the analysis performed under the original set A evaluations.

Underground Electrical Conduit Banks

The modifications to the FSAR presented in this section primarily are concerned with describing two separate aspects of the evaluation of these facilities. The first primary modification contains a detailed description of the analyses performed to estimate bending and shear stresses induced in flexible buried systems by wave passage effects and is extracted directly from Section 5.2.4 of TVA's Design Document WB-DC-20-26. The modification serves to make the notation of this section compatible with the descriptions provided in TVA's Design Criteria Document WB-DC-20-26, Revision 6, but, in fact, does not significantly differ from the original description contained in the previously accepted FSAR. The only change lies in the notation to determine the peak acceleration of the surface ground motion, given the basement bedrock acceleration. The soil amplification through the soil layer was reviewed and accepted by the staff in IR 50-390, 391/89-21. Therefore, this modification is considered acceptable, since it clarifies the analysis used for calculating the stresses in the conduit.

The second major modification made to this FSAR section is a detailed presentation of the analyses performed to estimate maximum values of axial stresses induced in buried systems due to the passage of surface seismic waves. These descriptions are also contained in Design Documents WB-DC-20-26 and WB-DC-40-31.5, Revision 3. These presentations are, in turn, based on the evaluations that have been presented in the open literature.* These references have been reviewed and evaluated and are considered to present descriptions of procedures which lead to conservative estimates of the maximum axial loads applied to the systems. On the basis of the review of the design calculations in IR 50-390, 391/89-21, and the review of the open literature,* the analysis methods contained in the design criteria documents are considered acceptable, and the modifications to the FSAR appropriate.

Class 1E Electrical Systems Manholes and Handholes

The modification in this FSAR section is editorial and is considered acceptable.

Miscellaneous Yard Structures

The editorial change in this FSAR section is minor; it clarifies the description of the structures of interest and is acceptable.

*See Newmark, 1968, 1972; Yeh, 1974; Shah and Chu, 1974; Goodling, 1978, 1979, 1983; ASCE, 1983; Iqbal and Goodling, 1975; and Westinghouse, 1985.

Structure Interaction Analysis--Waste Package Area (WPA), Condensate Demineralizer Waste Evaporator (CDWE) Building, and Auxiliary Control Building (ACB)

This FSAR section was added to the description of the set A calculations to summarize the results of the evaluation of the adequacy of seismic gaps between structures. The results of the evaluations of these structures indicate that the gaps provided are adequate to eliminate concern for this issue. On the basis of previous evaluations of the seismic calculations performed during site audit (see publicly available memorandum from P. S. Tam to the NRC Document Control Desk, October 19, 1990), the conclusions presented are reasonable. This addition to the FSAR revision is, therefore, considered acceptable.

3.7.2.1.4 Category I Soil-Supported Structures Evaluation and New Design/ Modification Analysis (Set B and Set B+C)

This section was added to the FSAR to describe those analyses performed for the set B and set C analyses for soil-supported structures. The analyses performed made use of the SHAKE and SASSI computer programs to determine structural response, including the effects of soil-structure interaction. In this approach, the ground motions for each case considered were specified at the level of the top of bedrock and were transmitted through the soil column to account for soil amplification effects on the free-field motions. The effects of strain-dependent shear modulus degradation and equivalent soil damping were suitably accounted for in these calculations using appropriate properties for the particular materials in the overburden. In conformity with the requirements of the Standard Review Plan (SRP, NUREG-0800), calculations were performed for upper-bound, best-estimate, and lower-bound soil properties to include the effects of potential soil variability in the analyses, with enveloping of calculated responses used to arrive at design acceleration response spectra for each input control motion.

In the calculations reviewed during various audits conducted at the site (see publicly available memorandum from P. S. Tam to the NRC Document Control Desk, October 19, 1990), the staff noted that the range of variability included in the analyses was from 1/2 to 3/2 of the best-estimate, low-strain shear moduli, which is less than the SPR-recommended range of 1/2 to 4/2 the best-estimate properties. However, the procedure used to broaden the computed amplified response spectra by ±15 percent in addition to the variability in soil properties considered was shown for this particular site to conservatively envelope the effect of variation of properties normally considered.

In addition, some calculations using the lower bound soil properties led to effective soil hysteretic damping ratios which exceeded the limits of the current version of the SRP. Additional computations were performed which for this site indicate that these exceedances do not lead to significant changes to the computed structural responses (axial forces, shear and bending moments) and amplified response spectra. The FSAR adequately describes the calculations conducted for these soil-supported structures.

On the basis of the detailed audits conducted, the descriptions provided in this FSAR section are acceptable.

3.7.2.1.5 Category I Pile-Supported Structures

This FSAR section was not changed since it refers to the calculations conducted for the set A criteria, including ground motions. Since the next section presents the results of additional computations conducted for the set B and set C calculations, the applicant committed to change the title of this section to indicate that it refers to the original set A calculations.

3.7.2.1.6 Category I Pile-Supported Structures Evaluation and New Design/ Modification Analyses (Set B and Set B+C)

The primary addition to this FSAR section concerns the description of the evaluation of the additional diesel generator building (ADGB) performed for the set B and set C criteria. The ADGB was designed subsequent to the other Category I structures and was added to the FSAR by Amendment No. 57. The reanalyses performed for this structure made use of the SHAKE and SASSI computer codes, as described above, but incorporated the effects of the pile foundations into the structural model. On the basis of staff's previous review (letter from S. C. Black to O. D. Kingsley (TVA), October 31, 1989) and the discussion in Section 3.7.2.1.4 above, the application of SASSI and SHAKE computer codes for Watts Bar soil-structure interaction calculation is acceptable to the staff. The seismic response analysis was performed with the CLASSI computer program for the upper-bound, best-estimate, and lower-bound soil columns. The use of the CLASSI computer code, which has been widely used to license many other nuclear power plants, is acceptable to the staff. Similar departures from the SRP, as described above for the other soil-supported structures, were noted in other licensee calculations, and the rationale for staff acceptance of the results of these calculations is also applicable for this structure. The description of the analyses presented in this section is considered adequate based on the detailed audits of the seismic calculations performed during the various site visits (publicly available memorandum from P. S. Tam to the NRC Document Control Desk, October 19, 1990).

3.7.2.2 Natural Frequencies and Response Loads for the Nuclear Steam Supply System

The previous FSAR included tables and figures to explicitly show information on natural frequencies, mode shapes, and response loads from the set A analysis of the nuclear steam supply system (NSSS). Amendment No. 64 deleted such information and made reference to a Westinghouse report instead (Westinghouse, 1985). The staff questioned the basis for this deletion. To resolve the staff concern, the applicant committed, in its December 18, 1990, letter, to reinstate the applicable portion from the previous FSAR Section 3.7.2.2 and replace the amendment. This commitment resolved the staff concern.

3.7.2.3 Techniques Used for Modeling

3.7.2.3.1 Other Than NSSS

This FSAR section addresses the criterion for determining whether or not a subsystem may be decoupled from the structure when developing the structural model. The criterion in the previous FSAR has been amended. The amended criterion is a function of the ratio in mass and frequency between the subsystem and the structure. The staff finds that the amended criterion is consistent with the one specified in TVA Seismic Design Criteria WB-DC-20-24, Revision 5, which, in general, conforms with the SRP guidelines, and thus concludes that FSAR Section 3.7.2.3.1 is acceptable.

3.7.2.3.2 For NSSS Analysis

The FSAR previously addressed the seismic analysis model of the reactor coolant system and included figures showing the configurations of the models for both the reactor coolant system and reactor pressure vessel. Amendment No. 64 deleted the description of the NSSS model. Instead, it made reference to Section 5.2.1.10.3 of the FSAR for the description of the NSSS analysis model and to the Westinghouse report (Westinghouse, 1985) for the Westinghouse-supplied model of the reactor coolant loop system. The staff questioned the basis for this deletion. To resolve the staff concern, the applicant committed, in its December 18, 1990, letter, to reinstate the previous FSAR Section 3.7.2.3.2 to replace the amendment. This commitment resolved the staff concern, and the proposed reinstated Subsection 3.7.2.3.2 is acceptable.

3.7.2.4 Soil-Structure Interaction

The primary modification in the description provided in the FSAR concerns descriptions of the procedures associated with the evaluation, new design, and modification analyses performed for the set B and set C analyses. The procedures and computer analyses used in these analyses are described in Sections 3.7.2.1.3 and 3.7.2.1.4 above. The added paragraphs are consistent with the previous descriptions provided and, based upon the previous review (IR 50-390, 391/89-21), are considered acceptable.

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.2 Evaluation and New Design or Modification Analysis

Except for set C analysis of the auxiliary control building, the amplified response spectra (ARS) for both set B and set C analyses were generated at the 75 frequency points specified in the revised FSAR Table 3.7-1 and at the natural frequencies of the foundation-structure system. These 75 frequency points are the same as those specified in SRP Table 3.7.1-1. For set C analysis of the auxiliary control building, the ARS were generated at the 55 period points as specified in the updated FSAR Table 3.7-1, and at the natural periods of the building. These 55 period points are the same as used for set A ARS generation. The frequencies or periods specified in the updated FSAR Table 3.7-1 for set B and set C ARS generation are identical to those specified in Table 5 of WB-DC-20-24, Revision 5, which the staff previously accepted in IR 50-390, 391/89-21. The staff, however, had a concern that FSAR Amendment No. 64 did not specify the time interval for set B and set C structural response analyses which generated the floor response time histories, and the time interval for generating ARS. In its December 18, 1990, letter, the applicant agreed to include the following time interval information for set B and set C analyses in the next amendment of the FSAR:

Structural response analysis method	Time interval for: Structural response analysis	Time interval for: ARS generation
Time domain	0.005 sec	0.005 and 0.0025 sec
Frequency domain	0.01 sec	0.010 to 0.0025 sec

The time interval for structural response analysis is consistent with the specification of TVA seismic CAP, Revision 2, which has been accepted by the staff (IR 50-390, 391/89-21). The time interval, DT, for ARS generation varies with the frequency, f, so that 1/(f*DT) equals or exceeds 10. It is a common industry practice and acceptable to NRC. The applicant's commitment thus resolved the previous staff concern.

Effect of the three earthquake components on ARS generation due to structural coupling was accounted for in the set B analysis with either one of the two following methods. With the first method, the three components of earthquake ground motion were input simultaneously to the structural response analysis, so that the floor response time history and the ARS generated thereof automatically included the structural coupling effect, if any. With the second method, one component of earthquake ground motion was input to the structure analysis at a time and the ARS was generated; co-directional ARS due to structural coupling were then combined by the square-root-of-the-sum-of-the-squares (SRSS) rule. In set C analysis, only the second method was used to account for the effect of three earthquake components on ARS due to structural coupling. The methods discussed previously are consistent with those specified in Tables 4 and 5 of TVA seismic CAP, Revision 2, for set B and set C analysis, respectively.

ARS were generated for a constant damping value of 1, 2, 4, 5, and 7 percent for the OBE condition, and 2, 3, 5, and 7 percent for the SSE condition. In addition, ARS for the ASME Code Case N411 variable damping were generated for both the OBE and SSE conditions. To account for the uncertainty in structural modeling, the frequency shift due to the soil property variation, and analysis technique, the peaks of the final set B and set C ARS were broadened by ±15 percent and ±10 percent of the corresponding structural frequencies, respectively, for all Category I structures except the ERCW pipe tunnels. The final set B+C ARS for use in the new design or modification were then obtained from enveloping the final set B and set C ARS.

The spectral damping values and the procedure for generating the final set B and set B+C ARS are consistent with the corresponding criteria specified in TVA's Design Criteria Document WB-DC-20-24, Revision 5, and are hence acceptable.

3.7.2.6 Components of Earthquake Motion

3.7.2.6.1 Original Analysis (Set A)

There is no technical amendment to this section of the FSAR.

3.7.2.6.2 Evaluation and New Design/Modification Analyses (Set B and Set C)

This new FSAR section addresses the technique for spatial combination of effects from the three earthquake components in the set B and set C analyses of structures:

(1) When response spectrum method of structural analysis is used, co-directional maximum responses from the three earthquake components are combined with the square-root-of-the-sum-of-the-squares (SRSS) technique.

(2) When the time-history method of structural analysis is used, either the co-directional maximum responses are combined with the SRSS technique or, as an option in the set B structure analysis, the co-directional concurrent responses are combined algebraically at each time step to produce a time history of the combined response.

The spatial combination techniques described above for structural analyses were found to be consistent, in general, with the SRP requirements and the staff concludes that FSAR Section 3.7.2.6.2 is acceptable.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Other Than NSSS

3.7.2.7.1.1 Original Analysis (Set A)

There is no technical amendment to this section of the FSAR.

3.7.2.7.1.2 Evaluation and New Design or Modification Analyses

This new FSAR section addresses the technique for combining modal responses for set B and set C analyses of structures, systems, and components other than the NSSS. For the response spectrum method of analysis, modal responses are combined in accordance with Regulatory Guide 1.92, Revision 1. For the time-history method of analysis, modal responses at each time step are combined algebraically. This is consistent with the criterion specified in TVA seismic CAP, Revision 2, and the staff concludes that FSAR Section 3.7.2.7.1 is acceptable.

3.7.2.7.2 NSSS System

There is no technical amendment to this section of the FSAR.

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

There is no technical amendment to this section of the FSAR.

3.7.2.9 Effects of Parameter Variations on Floor Spectra

In this FSAR section, the applicant proposed to broaden the spectral peaks of the ARS by ±10 percent based on the corresponding frequencies to account for the uncertainties owing to variations in material properties of the structure and soil foundation, and owing to approximations in structural modeling technique. The ±10 percent peak broadening deviates from the percentage actually applied in the Watts Bar seismic analysis of Category I structures for generating the ARS. As was found during site audit (publicly available memorandum from P. S. Tam to the NRC Document Control Desk, October 19, 1990), the computed floor response spectra were smoothed and peaks associated with the structural frequencies were broadened ±10 percent for set A and set C analyses. For set B analysis, the peaks were broadened ±15 percent. The ±10 percent peak broadening is consistent with the criteria specified in the TVA seismic CAP. The ±15 percent broadening of the peaks for a set B analysis is in accordance with Regulatory Guide 1.122. The applicant committed to revise the FSAR to state that the ±10 percent broadening is for set A and set C analyses (TVA letter to the NRC, dated

December 18, 1990). As for the set B analyses, the rule of ±15 percent will be applied. Therefore, the technique used for peak broadening of the floor response spectra is acceptable.

The FSAR also states that: "As an option, response spectra peak shifting as defined in ASME Code Case N-397 was used in some cases." Because this code case has never been used for accounting for the structural parameter variation, the applicant committed (letter from TVA to the NRC, dated December 18, 1990) to remove this statement from the FSAR.

3.7.2.10 Use of Constant Vertical Load Factors

3.7.2.10.1 Other Than NSSS

3.7.2.10.1.1 Original Analysis (Set A)

There are no technical changes to this section of the FSAR.

3.7.2.10.1.2 Evaluation and New Design or Modification Analyses

This new FSAR section is unspecific about whether or not constant vertical load factors were used in set B and set C analyses. The applicant agreed, in a letter dated December 18, 1990, to add a statement to the FSAR that "Constant vertical load factors were not used for either set B or set C analysis." In addition, because this section is only applicable to structures and not to systems and components, the applicant agreed to delete the words "system and components" from the text. The staff found the applicant's corrective actions acceptable.

3.7.2.10.2 For NSSS

This FSAR section was not changed.

On the basis of the findings discussed previously, the staff concludes that Section 3.7.2.10, as supplemented by the FSAR revision committed to by the applicant, in its letter dated December 18, 1990, is acceptable.

3.7.2.11 Methods Used to Account for Torsional Effects

The only technical change made to this FSAR section is the statement that: "For set B and set C analyses, modeling of torsional effects was refined by three-dimensional modeling." The seismic models used for set B and set C analyses were reviewed; the torsional effects were properly included and the accidental eccentricity equal to 5 percent of the maximum structural dimension was considered (IR 50-390, 391/89-21). Therefore, the staff finds the change to the FSAR acceptable.

3.7.2.12 Comparison of Responses--Set A versus Set B

This FSAR section is new and compares the responses from set A and set B analyses. The purposes of making these comparisons were to validate the original (set A) design calculations based on the set B analysis results on the existing plant structures, and to identify any features that required detailed reevaluation or upgrading. Currently, these comparisons and evaluations are being performed on a building-by-building basis. As committed to by the applicant, in its December 18, 1990, letter, this section of the FSAR will be revised once these evaluations are completed. The current editorial revisions in this section are acceptable to the staff; however, staff acceptance of this FSAR section remains open until the final review of the results of these comparisons. This will be tracked as Outstanding Issue 19(g).

3.7.2.14 Determination of Category I Structure Overturning Moments

3.7.2.14.2 Evaluation and New Design or Modification Analysis

This section was added to the FSAR to state that moments, shears, and vertical forces for set B and set C analyses were determined by the time-history modal analysis method. This statement is acceptable to the staff since the staff reviewed and found acceptable both set B and set C seismic analyses, as documented in IR 50-390, 391/89-21.

3.7.2.15 Analysis Procedure for Damping

There are two technical changes to this section of the FSAR regarding the method for determining the modal damping value when elements with different damping ratios are considered in one structural model. The first amendment is the deletion of the technique for set A analysis. According to the previous FSAR, the lowest element-associated damping value was taken to be the modal damping for the original analysis. The second technical amendment specifies the use of the strain energy method for determining the composite modal damping for set B and set C analyses. The staff questioned the basis for the first amendment, and the applicant agreed, in its December 18, 1990, letter, to reinstate the method for original analysis as specified in the previous FSAR. Regarding the method for set B and set C analyses, the staff found that the new FSAR contradicts TVA Seismic Design Criteria WB-DC-20-24, Revision 5, which states that "element associated damping shall be accounted for either directly or by the strain energy or composite modal damping approach." The applicant agreed, in its December 18, 1990, letter, to revise the FSAR to be consistent with the statement in WB-DC-20-24, Revision 5. The technique, documented in WB-DC-20-24, for determining the composite modal damping is, in general, consistent with the SRP. The staff found both commitments to be sufficient resolution and concludes that FSAR Section 3.7.2.15 is acceptable.

3.7.3 Seismic Subsystem Analysis

The staff has identified an issue regarding the number of earthquake stress cycles considered in the design of seismic subsystems. The applicant has stated that the number of equivalent peak stress cycles considered for the OBE and SSE are 20 cycles and 10 cycles, respectively. Previously, the total number of cycles considered for the OBE and SSE were 600 and 300, respectively, which was acceptable to the staff. As stated in the applicant's December 18, 1990, letter, the reduced number of cycles specified applies to non-NSSS Category I subsystem components and ASME Code Section III Class 1 piping and component fatigue analysis, and for the seismic testing of equipment. This criterion is not used in the qualification of cable tray, conduit, and HVAC systems. The number of equivalent peak stress cycles is based on the occurrence of two OBEs and one SSE during the design life of the plant (40 years). For each event, 10 cycles of maximum stress are considered based on the Standard Review Plan (SRP) and IEEE 344-1975.

For equipment and piping systems, the use of 10 peak stress cycles for the SSE is consistent with Sections 3.7.3 and 3.9.2 of the SRP and IEEE 344-1975 requirements and is, thus, acceptable to the staff. For the OBE case, the use of five OBE events for the entire design life of the plant is a guideline specified in Sections 3.7.3 and 3.9.2 of the SRP and IEEE 344-1975. The number of peak stress cycles for each OBE may be obtained from the actual time history, or a minimum of 10 peak stress cycles can be assumed. If 10 peak stress cycles are used for each OBE, then a total of 50 peak stress cycles for the entire design life of the plant would be required. Because the number of OBE events specified by the applicant does not meet the guidelines of the SRP and IEEE 344-1975, this issue remains open and will be identified as Outstanding Issue 19(a).

FSAR Section 3.7.3.3.1.1 was modified and a new section (3.7.3.3.1.3) was added to the FSAR to describe the mass modeling of piping, HVAC, conduit, and cable tray subsystems for seismic evaluation. The applicant stated that continuous or discrete mass models are developed for manual or computer analyses. The adequacy in selecting and locating lumped masses and the consideration of all significant modes of vibration were reviewed. As described in the applicant's December 18, 1990, letter to the NRC, in addition to the continuous mass, additional lumped masses are located at significant concentrated weights, such as heavy fittings or other in-line or attached commodities. A sufficient number of masses are included so that additional masses (or degrees of freedom) would not increase the predicted response by more than 10 percent. Alternatively, the number of masses are modeled to be at least twice as many as the number of modes with frequencies less than 33 Hz. For piping, the spacing is based on a 33-Hz frequency for spans between mass points with at least three mass points between supports in the same direction. The modeling methods described above are consistent with Section 3.7.2 of the SRP criteria for modeling of subsystems and, therefore, are acceptable.

FSAR Section 3.7.3.4.1 was revised to include the commitment that the frequencies of the subsystems are selected so that all significant modes of vibration are included in the analysis. Frequencies of simplified analysis models are determined by solutions of closed-form expressions. Frequencies of detailed analysis models are determined by computerized solutions. For HVAC, conduit, and cable tray systems the applicant's December 18, 1990, letter states that the FSAR will be revised to indicate that the number of modes included in the calculations are selected so that the inclusion of additional modes does not result in more than a 10-percent increase in responses. Alternatively, the dynamic analysis considers all modes up to 33 Hz and includes an additional check for any missing mass participation factors. These criteria are consistent with the guidelines stated in Section 3.7.2 of the SRP and are acceptable.

As stated in the applicant's December 18, 1990 letter, for piping systems, Section 3.4.5 of the Watts Bar piping design criteria requires that all modes below 33 Hz be included in the piping analysis. Also, the contribution of higher modes (usually calculated by the missing-mass method) are combined with those of lower modes by the square-root-of-the-sum-of-the-squares (SRSS) method. The staff's review of the applicant's letter and design criteria finds that the applicant's methodology is consistent with the guidelines of Section 3.9.2 of the SRP for selecting significant frequencies for simplified and computerized piping analyses and is, thus, acceptable.

In FSAR Section 3.7.3.5.1, the applicant revised its description of the equivalent static load method and stated that a multi-mode factor of 1.2 will be used for analysis of HVAC, conduit, and cable tray subsystems in lieu of the 1.5 factor previously used. A 1.5 multi-mode factor is in accordance with the guidelines of the SRP and was previously accepted by the staff in Section 3.7.3 of the Watts Bar SER. The justification for a 1.2 multi-mode factor was reviewed by the staff in an audit held on November 5-9, 1990, and is contained in Sargent & Lundy Calculation WCG-1-397, entitled "Two Degree of Freedom Comparison to a Couple System Response," dated February 21, 1990. The calculation uses the complete quadratic combination (CQC) method to combine the modal responses in the response spectrum analyses which yields varying results to the methods recommended in Regulatory Guide 1.92. Also, the staff found that the selected configurations in the study might not bound all of the installed configurations at Watts Bar. In a letter to the NRC, dated December 18, 1990, the applicant stated that additional calculations currently being performed in order to address the concerns regarding bounding configurations will be submitted for the staff's review when complete. Therefore, this item remains open and will be tracked by Outstanding Issue 19(b).

FSAR Section 3.7.3.6 was revised to describe the method used for the combination of the three components of earthquake motion for equipment, HVAC, conduit, and cable tray subsystems. Seismic input in each major horizontal direction is applied separately with the vertical input. Horizontal and vertical responses are combined by absolute summation and the larger of the two will be used for evaluation and design of the commodities. This method was utilized for all three seismic inputs--set A, set B, and set B+C. The staff's evaluation of the development of set A, B, and B+C seismic loads for original analysis/qualification, evaluation, and new design/modification is described in Section 3.7.2 of this SSER. The applicant's procedure for combining spatial components (one horizontal and vertical components) by the absolute sum method has been previously approved in Section 3.7.3 of the Watts Bar SER (NUREG-0847) and, therefore, is acceptable.

A new section (3.7.3.6.1) was added to the FSAR in Amendment No. 64 to provide specific requirements for piping subsystems with regard to the combination of the maximum directional responses caused by each of the three components of earthquake motion by SRSS. The applicant's procedures for combining spatial components of piping subsystem responses by the SRSS method are in accordance with the guidelines of Section 3.9.2 of the SRP and are, thus, acceptable.

In Amendment No. 64, FSAR Section 3.7.3.8.1 was revised to provide a more detailed description of the codes used for piping analysis. The staff raised an issue regarding the analysis of some classes of pipe using ANSI Standard B31.1. The applicant stated, in the letter to the NRC dated December 18, 1990, that for piping analysis, the use of ANSI B31.1 applies to nonnuclear safety piping only; this is acceptable to the staff.

In Amendment No. 64 to the FSAR, the applicant listed specific ASME Code cases it proposes to use in the design of piping systems. The code cases are:

- N-122, "Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1"

- N-313, "Alternate Rules for Half Coupling Branch Connections, Section III, Division 1, Class 2"

- N-318, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1"

- N-319, "Alternate Procedure for Evaluation of Stresses in Butt Weld Elbows in Class 1 Piping, Section III, Division 1"

- N-391, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1"

- N-392, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 or 3 Piping, Section III, Division 1"

- N-397, "Alternative Rules to the Spectral Broadening Procedures of N-1226.3 for Class 1, 2, and 3 Piping"

- N-411, "Alternate Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Systems, Section III, Division 1"

- N-463, "Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping That Exceed the Acceptance Standards of IWB-3514.2"

- 1606, "Stress Criteria for Section III, Class 2 and 3, Piping Subjected to Upset, Emergency, and Faulted Operating Conditions"

The staff asked the applicant to specify the particular revision and date of the code cases it intends to use in its piping analyses. The applicant committed to use those code cases that are endorsed by RG 1.84 and will revise its FSAR to include the specific revisions of the code cases. When this information is submitted, the staff will complete its review and evaluation of the acceptability of each code case. The staff's evaluation of Code Case N-411 is given below (in Section 3.7.3 of this SSER). The staff's evaluation of the remaining code cases will be tracked as Outstanding Issue 19(c).

The staff identified an issue regarding the applicant's simplified seismic analysis of equipment. The applicant stated that for equipment qualification the peak acceleration of the applicable floor response spectra is multiplied by a factor of 1.5 if natural frequencies are not determined. Lower load factors (between 1.0 and 1.5) are used only when justified by frequency analysis. Previously, a factor of 1.5 was used regardless of frequency. As stated in Section 3.7.3 of the Watts Bar SER, it was understood that for balance-of-plant (BOP) equipment, the peak acceleration value of the applicable response spectrum was increased by a factor of 1.5 and applied as an equivalent static load factor to the entire mass of the equipment being evaluated.

The applicant's letter dated December 18, 1990, stated that when the equipment's natural frequency is determined and there is only one mode below 33 Hz (as determined by test or analysis) the equivalent static loads can be determined by using a minimum factor of 1.0. The peak acceleration of the floor spectra is used (without any load factor) provided any one of three listed criteria is met. One of these criteria is if the fundamental frequency of the equipment is lower than the rigid frequency but its other frequencies are higher than the rigid frequency. Under this condition, the response would be indicative of a one-degree-of-freedom system for which a load factor of 1.0 would be appropriate. In addition, the staff finds the use of the peak acceleration from the floor response spectrum curve conservative. Thus, the staff concludes that the use of a 1.0 load factor coupled with the peak acceleration from the floor response spectrum curve, when there is only one mode below 33 Hz, is acceptable.

The staff reviewed the applicant's criteria for consideration of torsional effects of eccentric masses in piping analysis. In Amendment No. 64, the applicant included member stiffnesses in the analysis to simulate the flexibility of cantilevers. Previously, the cantilever members were assumed to be infinitely stiff. This revision is consistent with the guidelines of SRP Section 3.9.2, and is thus acceptable.

FSAR Section 3.7.3.15 refers to Tables 3.7-2 and 3.7-24 for specific values to be used for the critical damping of structures, systems, and components. In Amendment No. 64, Table 3.7-2 has been revised to include damping values specifically for cable tray, conduit, HVAC, and equipment subjected to set A, B, and B+C input loads. Previously, there were no damping values specifically given for these subsystems and components.

For conduit systems subjected to set A loads, FSAR Table 3.7-2 duplicates the damping value of 2 percent for the SSE from the previous FSAR. No value is presented for the OBE case because the design is based on SSE only. For sets B and B+C, damping values of 4 percent and 7 percent are used for the OBE and SSE, respectively. During an audit held between November 5 and November 9, 1990, the staff reviewed the justification for these damping values. The basis for the damping values for set B and set B+C are documented in a report of conduit tests performed by TVA, "Summary Test Report on Damping in Electrical Conduit," CEB-BN-1028, dated June 1987, as well as in test reports by ANCO Engineers, "Cable Tray and Conduit Raceway Seismic Test Program," Report No. 1053-21.1-4 (Volumes I-VI), and by Wyle Laboratories, "Seismic Qualification/Verification of Various Aluminum Electrical Conduit Configurations," Report No. 17743-1 (CEB-BN-1002), Volumes I and II, dated May 9, 1986.

For comparison purposes, Regulatory Guide 1.61 recommends for welded steel structures, damping values of 2 percent and 4 percent for OBE and SSE, respectively. For bolted steel structures, damping values of 4 percent and 7 percent for OBE and SSE, respectively, are recommended. Conduit systems at Watts Bar are primarily constructed of welded steel members (support frames) with some bolting-type connections. Typically, the bolting-type connections are the conduit clamp attachment to the support frame, concrete anchors when used, and the threaded fittings.

Having reviewed the test reports, the staff concluded that the results of the Wyle tests are of limited value since they were performed on aluminum conduit only, whereas, most conduits at Watts Bar are made of steel. For the ANCO tests that were performed at high acceleration levels (comparable to the SSE), much of the data suggests the use of approximately 4- to 5-percent damping based on a mean-value-minus-one standard deviation. The applicant had proposed the use of 7-percent damping for the SSE based on the TVA tests using the average value of damping times 0.85 to account for variation in the cable fill.

The staff has identified an issue regarding the use of 4-percent and 7-percent damping values for conduit systems. The staff determined there is insufficient basis for using average values of the damping test values, particularly since the scatter of test data ranged from 3 percent to 22 percent. A second concern is whether the use of the limited TVA test data sufficiently covers the variation in configurations and design parameters such as cable fill, span lengths, diameters, and supporting conditions. In a letter to the NRC dated December 18, 1990, the applicant provided some additional information which will require additional review. In that letter, the applicant also stated that the use of 4-percent and 7-percent damping values for OBE and SSE, respectively, has precedence at some other nuclear power facilities, such as Vogtle (4% OBE, 7% SSE), Byron and Braidwood (4% OBE, 7% SSE), Diablo Canyon (7%), and Grand Gulf (7% OBE, 7% SSE). However, it is not clear to the staff whether the bases to justify the use of higher damping values for these plants are applicable to Watts Bar. Therefore, on the basis of the two concerns noted above, this issue remains open and will be tracked by Outstanding Issue 19(d).

For the HVAC subsystems, the staff also identified an issue regarding the proposed damping ratios in FSAR Table 3.2-7. As a result, TVA agreed to apply the RG 1.61 damping values for bolted structures to companion angle ducts, and the RG 1.61 values for welded structures to welded ducts. The damping values for pocket-lock construction are the same as those previously accepted by the staff for the Sequoyah nuclear plant (letter from S. Black to O. D. Kingsley (TVA), dated August 25, 1989). In its December 18, 1990, letter to the NRC, the applicant agreed to revise the HVAC damping ratios in FSAR Table 3.2-7 as follows:

	Set B		Set B+C	
Duct construction	OBE	SSE	OBE	SSE
Companion angle	4%	7%	4%	7%
Pocket lock	7%	7%	7%	7%
Welded	2%	4%	2%	4%

The staff finds these values acceptable.

For cable tray systems, the damping ratios presented in FSAR Table 3.7-2 for the OBE and SSE, respectively, are 4 percent and 5 percent for set A, and 4 percent and 7 percent for sets B and B+C. To justify these values, the applicant stated that the values are consistent with the recommended values in Regulatory Guide 1.61 for bolted structures and with the results of tests conducted by ANCO as documented in, "Cable Tray and Conduit Raceway Seismic Test Program," Report No. 1053-21.1-4 (Volumes I-VI). The cable tray systems, as installed at Watts Bar, consist of cable tray assemblies bolted to each other and bolted to welded support frames which, in turn, are typically fixed with a bolted anchorage. The tray assemblies themselves have bolted support attachments, splice plates, and in some cases bolted cover plates. As such, the systems can reasonably be expected to exhibit the characteristics of bolted structures for which damping ratios of 4 percent and 7 percent for the OBE and SSE, respectively, are recommended in RG 1.61. Further, in the ANCO tests, with cable tray assemblies bolted directly to a relatively rigid shake-table frame, the minimum observed damping ratio was 7.5 percent for coated cables and 20 percent for uncoated cables, for 100-percent loaded trays at acceleration levels comparable to the OBE. These test conditions correspond with the actual cable system installations and the test results should be indicative of the results to be expected in the field. On the basis of these observations, the staff finds the applicant's damping ratios assigned to cable tray systems acceptable.

For equipment and components, the FSAR specifies damping values of 2 percent and 3 percent for the OBE and SSE, respectively. These damping values are applied to all three sets of seismic loads, sets A, B, and B+C. For seismic Category I piping analysis, the applicant specifies damping values of 2 percent and 3 percent for OBE and SSE, respectively, for piping of at least 12-inch diameter, and 1 percent (OBE) and 2 percent (SSE) for piping of less than 12-inch diameter. These damping values are applied to seismic load sets B and B+C. The damping values for set A are unchanged from the previous FSAR. Because these damping values for equipment and piping are in agreement with Regulatory Guide 1.61, the values are acceptable.

The applicant has also proposed to use damping values from ASME Code Case N-411 as an alternative for piping systems. ASME Code Case N-411 has been found acceptable by the staff, subject to certain limitations as specified in Regulatory Guide 1.84. In order to satisfy one of the limitations, the staff requires that the ASME Code Case N-411 damping values only be used in piping system response spectrum analyses where the Watts Bar seismic load set B+C is used. Subject to the above limitations, the staff finds the use of ASME Code Case N-411 for Watts Bar consistent with Regulatory Guide 1.84 and thus acceptable.

The staff reviewed the analysis of mounting for equipment and components. The applicant's criteria consider the flexibility of non-rigid supports to floor-mounted or wall-mounted equipment and components. For non-rigid supports, a coupled analysis of the equipment and/or component assembly and its support and/or anchorage is performed. For line-mounted equipment/components and their mountings, the subsystem response (e.g., piping response) at the equipment/component location is kept below the device qualification level. These methods adequately account for the potential amplification due to support flexibilities of equipment and components and are thus acceptable. To address the potential effects due to wall and floor flexibility on the amplified floor response spectra for the subsystem evaluations, TVA performed a separate study. The staff reviewed the study previously, as is documented in IR 50-390, 391/89-21, dated May 10, 1990, and in an audit report dated October 10, 1990 (publicly available memorandum from P. S. Tam to the NRC Document Control Desk, October 19, 1990), and it is acceptable.

The staff reviewed the loads and load combinations used in the design of HVAC ducts and duct supports. The staff identified two concerns regarding the loads: (1) LOCA and high-energy line break (HELB) pressure loads were not considered in the design of HVAC ducts inside the containment and (2) the definition of fluid-induced loads did not include loads that resulted from sudden damper closure.

The applicant agreed, in its December 18, 1990, letter to the NRC, to address the first concern by including a load in all the applicable duct load combinations which include accident pressure exterior to the duct resulting from jet impingement or compartmentalization pressure. Where possible, the duct will be protected from these effects. Otherwise, the duct shall be designed as necessary to withstand the forces from these effects consistent with allowable stress criteria.

The applicant responded to the second concern by stating that loads resulting from sudden damper closure are not considered because system operation precludes these loads. Fire dampers can only close when the fans are stopped and forced air flow is discontinued. Therefore, no pressure transients are expected to occur. In addition, other dampers, which close in response to an initiating accident event, have closure times ranging from approximately 4 to 16 seconds. These relatively slow closure rates preclude any significant loads due to pressure transients. The staff's review of the applicant's response finds the system operation adequately precludes the sudden damper closure load.

Therefore, the staff finds the applicant's methods for applying loads and load combinations to HVAC systems acceptable.

Since the original design of the structures, systems, and components at Watts Bar, a number of issues were raised by various sources. These sources include NRC inspection reports, Watts Bar reports (NCRs, CAQRs, PIRs, and SCRs), employee concerns, and internal and external reviews. Problems were identified in the areas of design, construction, and inspection/quality assurance of the plant features.

To resolve these issues, corrective action programs are being conducted by the applicant which will assure that Watts Bar plant features meet upgraded design criteria and licensing commitments. One phase of these validation programs consists of an engineering evaluation to validate the adequacy of the existing designs. The approach taken by the applicant in the corrective action programs is to validate the existing commodities by grouping the components having similar configurations and then evaluating the "worst case" or "critical case" and performing "bounding calculations."

The "worst case" approach involves identifying from actual installed configurations the most severe example of a given population. The worst-case approach is being used to validate such items as platforms, pipe whip restraints, concrete, and masonry walls.

The "critical case" approach uses actual or hypothetical configurations that combine attributes that have the greatest effect on the ability of the plant system or component in meeting allowable stresses. The critical cases combine the attributes from the various actual configurations in a given population. The critical-case approach is being used to validate conduit systems, cable tray systems, and HVAC systems.

Bounding calculations envelope the effects of varying parameters on a representative population. Initially, the features are grouped and the enveloping attributes are identified. Then, the bounding calculation determines the maximum stress for an actual or hypothetical condition. Bounding calculations may be performed to evaluate worst cases or critical cases. Presently, bounding calculations are used for the evaluation of small-bore pipe support variances, equipment seismic qualification, certain cable tray configurations, and other components.

The descriptions given above are based on the definition of the worst case, critical case, and bounding calculation as provided by the applicant in a letter from E. G. Wallace to the NRC, dated September 14, 1990, and as presented by the applicant during the November 5-9, 1990, site audit. Since all three approaches rely on either the actual configuration and attributes or the hypothetical combination of attributes, which is more severe, the staff considers the use of the worst case, critical case, and bounding calculation approach acceptable. The staff has not yet reviewed the procedures used to perform the walkthrough or the basis for grouping the configurations and identifying critical attributes. The implementation of these three methods will be reviewed and tracked by Outstanding Issue 19(e).

3.8 Design of Category I Structures

By Enclosure 4 to the letter dated January 4, 1991 (P.S. Tam to O.D. Kingsley), the staff expressed a number of concerns regarding (1) codes and load combinations and (2) stress allowables. The applicant has been asked to address these concerns. The staff will track its efforts by Outstanding Issue 19(j).

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The applicant performed a non-linear elastic-plastic analysis of the feedwater system inside the containment in order to evaluate the pressure boundary integrity of the feedwater piping for the feedwater water hammer that would occur if the check valve slammed shut following a postulated rupture at the main header in the turbine building. The applicant has proposed to use the rules in Appendix F of the ASME Code to develop acceptance criteria for the piping. However, as part of the piping evaluation, the applicant has also proposed assuming that certain supports fail when the loads exceed their calculated capacities. The staff considers this criterion and the applicant's proposed method of analysis an open issue requiring further staff review. This will be tracked as Outstanding Issue 20(a).

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

The staff identified an issue regarding the use of experience data as a method of seismic qualification of Category I(L) (limited structural integrity) piping. Presently, the staff does not permit the use of experience data to qualify safety-related piping systems for the plant design loading conditions. Category I(L) systems are systems whose failure could affect the functioning of a safety-related system. The applicant, in FSAR Section 3.2, stated that Category I(L) systems are seismically qualified to meet the intent of position 2 of Regulatory Guide 1.29. The applicant gave the staff information regarding the proposed methodology for using experience data. The staff has communicated its concern to the applicant in a letter from P. S. Tam to O. D. Kingsley (TVA), dated November 29, 1990, and is continuing its review of this item. This will be tracked as Outstanding Issue 19(h).

3.9.3.3 Design and Installation of Pressure-Relief Devices

The staff reviewed the design and installation of pressure-relief devices. The applicant has provided revised set pressures, accumulation pressures, and blowdown pressures for the Watts Bar main steam safety valves. The staff is continuing its review of the operating characteristics of these valves and will track this as Outstanding Issue 19(i).

3.9.3.4 Component Supports

The applicant proposed new criteria for service load combinations and associated stress limits for ASME Code Class 1, 2, and 3 piping supports in FSAR Section 3.9.3.4.2. For linear supports, the applicant had previously proposed load combinations and stress limits that were based on SRP Section 3.8.3. The applicant's new criteria for load combinations and associated stress limits are based on American Institute of Steel Construction (AISC) stress allowable criteria using the service level A, B, and C stress limit factors currently specified in Subsection NF of the ASME Code. The applicant has placed an additional restriction on the stress limits that, for all loading combinations, the tensile stresses shall not exceed nine-tenths of the material yield stress and the buckling loads shall not exceed two thirds critical buckling. For component standard supports, the applicant proposed load combinations and associated stress limits which are either based on criteria in Subsection NF of the ASME Code or, for those standard component supports not originally designed to Subsection NF of the ASME Code, on criteria in Manufacturers Standard Specification (MSS) SP-58. The staff finds that the specified service load combinations and associated stress limits for piping supports in systems classified as seismic Category I provide a conservative basis for the design of pipe support components to withstand the most adverse combination of loading events without loss of structural integrity.

The applicant proposed new support stiffness and deflection limits for seismic Category I piping supports in FSAR Section 3.9.3.4.2. The staff has asked the applicant to provide additional information in support of these changed criteria. Upon completion of its review of the additional information supplied by the applicant, the staff will report its findings in an SSER. This will be tracked as Outstanding Issue 20(b).

In Section 3.9.3.4 of the Watts Bar SER, the staff stated that the applicant had responded to IE Bulletin 79-02 for the Watts Bar facility in a letter dated July 7, 1980. At that time, the staff reviewed the applicant's response with respect to the flexibility of the pipe support baseplate and its effect on anchor bolt loads and determined that the staff needed additional information before it could accept TVA's justification of the use of rigid baseplate criteria. The staff sent a letter on June 28, 1985, to TVA requesting additional information concerning flexibility requirements in pipe support baseplate design using concrete expansion anchors. TVA responded in a letter dated August 22, 1985. At a recent audit at Watts Bar held between November 5 and November 9, 1990, the staff found that the TVA letter response of August 22, 1985, has been superseded as a result of the corrective action programs now being implemented at Watts Bar. The applicant has provided a partially revised response by its letter of January 31, 1991. Pending receipt of the applicant's revised response and subsequent review, this item remains open; the staff will continue to track it as Confirmatory Issue 9.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

For the seismic and dynamic qualification of equipment, the applicant refers to Regulatory Guide 1.48 and to IEEE Standard 344-1971 or IEEE Standard 344-1975, depending on when the equipment was procured. The applicant's December 18, 1990, letter to the NRC stated that Category I, Class 1E equipment is qualified in accordance with IEEE 344-1975 for equipment procured after September 1, 1974. Equipment procured before September 1, 1974, was seismically qualified in accordance with the requirements of IEEE 344-1971.

Although the SRP does not recommend adherence to IEEE 344-1975 for plants with construction permit applications docketed before October 27, 1972, the SRP does specify certain additional guidelines. These additional guidelines include describing the extent to which the seismic and dynamic qualification of mechanical and electrical equipment and their supports meet IEEE 344-1975, Regulatory Guide 1.100, and the criteria listed in SRP Section 3.10.II.1. In addition, the SRP states that it should be demonstrated that equipment has adequate margin to perform its intended design functions during seismic and dynamic events when considering the effects of possible multi-mode response and simultaneous vertical and horizontal excitations on equipment operability. The applicant has not yet demonstrated that it satisfies these SRP guidelines for equipment qualification. The applicant has committed only to IEEE 344-1971. Therefore, this issue remains open and will continue to be tracked as Outstanding Issue 4(a).

The staff identified an issue regarding the seismic design of cable trays and conduit at Watts Bar. FSAR Section 3.10.3.2.1 states that all cable trays and conduit are designated as "seismic Category I(L)." The staff's evaluation of the applicant's seismic classification of cable trays and conduit as Category I(L) is discussed in Section 3.2.1 of this SSER. The cable tray design criteria corresponding to this categorization includes: limiting the allowable vertical bending moment to 80 percent of the ultimate capacity of the tray, limiting the allowable horizontal moment to a value corresponding to a ductility factor of three, maintaining a minimum factor of safety of three for dead load effects alone, and designing for the load combination of dead load plus SSE. A description of the study performed by EQE Engineering to develop a program plan for the qualification of cable trays at Watts Bar, the design criteria for Category I cable tray supports and Category I(L) cable trays and backup calculations for the criteria, are included with the applicant's December 18, 1990, letter to the NRC. The applicant has not provided specific design criteria for conduit. Because the staff has not accepted the categorization of conduit and cable trays as I(L), as discussed in Section 3.2.1, the design criteria for these commodities remains an open item, and is tracked by the new Outstanding Issue 18.

The following did not come from the staff's review of FSAR Amendment No. 64:

SSER 1 and SSER 3 described a number of generic and specific concerns (Outstanding Issue 4(a)). The staff updated the resolution status of several of these issues in IR 50-390, 391/90-05 (May 10, 1990). Pages 28-39 of that report have been incorporated in this SSER as Appendix M. Appendix M resolved a number of generic and specific concerns, whose status is summarized in the sections that follow.

3.10.1 Generic Concerns

(1) Single-frequency and single-axis tests were performed to qualify electrical equipment. Status is updated in Appendix M, but this concern is still unresolved.

(2) This concern was resolved in SSER 3.

(3) In numerous cases, particularly for electrical cabinets, equipment is field mounted by welding but test mounted by bolting. This concern is resolved in Appendix M.

(4) Many safety-related equipment items, such as the insulation of motors, transformers, and other electrical devices, are age sensitive with respect to their seismic performance. This concern is resolved in Appendix M, with the environmental qualification (EQ) aspect to be addressed by the special program on EQ (SER Section 1.13.2).

(5) This concern was resolved in SSER 3.

(6) This concern was resolved in SSER 3.

(7) This concern was resolved in SSER 3.

3.10.2 Specific Concerns

(1)(a) The staff asked the applicant to demonstrate that the welded field mounting for the reactor trip switchgear is structurally as sound as the bolted lab mounting. This concern is resolved in Appendix M.

Concerns (1)(b), (1)(c), and (1)(d) were resolved in SSER 3.

(2)(a) The staff asked the applicant to demonstrate that the field mounting of the reactor protection system cabinet is as adequate as the lab mounting. This concern is resolved in Appendix M.

(2)(b) This concern was resolved in SSER 3.

(2)(c) The staff asked the applicant to evaluate the degree of amplification that occurred in the reactor protection system cabinet response motion during tests to clearly justify single-frequency testing. This concern is resolved in Appendix M.

(3) This concern was resolved in SSER 3.

(4) This concern was resolved in SSER 3.

(5)(a) The seismic analysis performed by the applicant for the main control boards assumed the panel was fixed at its base. This concern is resolved in Appendix M.

(5)(b) This concern was resolved in SSER 3.

(6) This concern was resolved in SSER 3.

(7)(a) This concern was resolved in SSER 3.

(7)(b) The staff asked the applicant to verify that the 125-V dc vital batteries will have spacers installed, as was done during the qualification tests. This concern is resolved in Appendix M.

(7)(c) This concern was resolved in SSER 3.

(8) This concern was resolved in SSER 3.

(9) This concern was resolved in SSER 3.

(10) This concern was resolved in SSER 3.

(11) This concern was resolved in SSER 3.

(12) This concern was resolved in SSER 3.

(13)(a) This concern was resolved in SSER 3.

(13)(b) The staff asked the applicant to justify using single-frequency, single-axis tests on the Barksdale pressure switch. This concern is resolved in Appendix M.

In summary, of all the concerns identified in SSER 1 and SSER 3, only concern 3.10.1(1) remains unresolved. Thus, Outstanding Issue 4(a) is mostly resolved; the staff will report on the status of the remaining action in a future SSER.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The staff has partially completed its review of TVA's application to replace the resistance temperature detector (RTD) bypass system with a new design. The staff issued the results of its review in a letter (S. C. Black, NRC, to O. D. Kingsley, Jr., TVA, dated June 13, 1989), stating that the Eagle-21 microprocessor system is acceptable for monitoring reactor coolant temperature. That document is hereby incorporated by reference. In addition, the applicant has incorporated the information for the approved new design in Final Safety Analysis Report (FSAR) Amendment No. 63, Sections 5.1, 5.3.2, 5.4, 7.1.3, 7.2, and 15.2. The staff will report results of this design change review in a future SSER. This issue will be tracked as Outstanding Issue 21.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.1 System Design

By letters dated September 19, 1985, September 17, 1986, and August 31, 1987, TVA informed the staff that TVA intended not to install the upper head injection (UHI) system at Watts Bar Nuclear Plant. The staff is reviewing those letters and FSAR Amendment No. 63 to assess whether such a design change is acceptable; the UHI design was originally approved in the SER in 1982. The staff's review effort will be tracked as Outstanding Issue 22.

11 RADIOACTIVE WASTE MANAGEMENT

11.7 NUREG-0737 Items

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

In SSER 5, the staff kept proposed License Condition 6(b) open pending the applicant's reevaluation of the implementation date to have the capability for continuous collection of postaccident plant gaseous effluents. By letter dated January 3, 1991, the applicant committed to have the procedural revision and upgrade of the radiation monitors by fuel load. This commitment will ensure the plant will have the capability for continuous collection of postaccident gaseous effluents by fuel load. This commitment is an improvement over the previous implementation schedule (i.e., before 5% power operation), and thus resolves the staff's concern stated in SSER 5. Thus, proposed License Condition 6(b) is considered resolved.

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

In SSER 5, the staff concluded that the applicant's leakage-reduction program is acceptable, and stated that proposed Licence Condition 24 will be resolved if the applicant accepts the inclusion of the waste gas disposal system (WGDS) in the leakage-reduction program.

By letter dated March 27, 1986, the applicant justified excluding the WGDS from the program (i.e., Section 6.8.5 of the then-proposed Watts Bar Technical Specifications). However, in Generic Letter (GL) 89-01, dated January 31, 1989, the staff requested that licensees and applicants relocate all radiological effluent technical specifications (RETS) to the respective plant offsite dose calculation manual (ODCM). Among line items to be relocated is the specification on the WGDS. Therefore, in accordance with the guidance of GL 89-01, the staff would exclude the WGDS from the Watts Bar Technical Specifications, but expects to see it included in the ODCM. The staff is reviewing the ODCM and is tracking it by TAC 77553. Proposed License Condition 24 will be considered fully resolved when the ODCM is issued accordingly.

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 as a result of 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 the bulk of the applicant's response to GL 83-28 and found the applicant's response acceptable for the following items:

- Item 1.1, Post-Trip Review (Program and Procedure) (letter from P.S. Tam, NRC, to O.D. Kingsley, TVA, dated Auguts 13, 1990)
- Item 1.2, Post-Trip Review (Data and Information Capability) (Inspection Report 50-390, 391/86-04, dated May 28, 1986)
- Item 2.1, Equipment Classification and Vendor Interface (Reactor Trip System Components) (letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 18, 1990)
- Item 2.2, Part 1, Equipment Classification Program (letter from S. C. Black, NRC, to O. D. Kingsley, TVA, dated June 1, 1989); Part 2 (letter from F. J. Hebdon, NRC, to O. D. Kingsley, TVA, dated September 7, 1990)
- Items 3.1.1 and 3.1.2, Post-Maintenance Testing of Trip System Components, (Inspection Report 50-390, 391/86-04, dated May 28, 1986)
- Items 3.1.3 and 3.2.3, Post-Maintenance Testing in Technical Specifications That Could Degrade Safety (letter form P. S. Tam, NRC, to O. D. Kingsley, TVA, dated July 2, 1990)
- Items 3.2.1 and 3.2.2, Post-Maintenance Testing of All Other Components (Inspection Report 50-390, 391/86-04, dated May 28, 1986)
- Items 4.1, Trip System Reliability (Vendor-Related Modifications) (Inspection Report 50-390/84-53, dated August 1, 1984)
- Item 4.3, Shunt Attachment to Reactor Trip Breaker (SSER 3 Section 15.3.6, and letter from P. S. Tam, NRC, to O. D. Kingsley, TVA, dated June 18, 1990; resolution of this issue eliminated proposed License Condition 40)

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

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

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

16 TECHNICAL SPECIFICATIONS

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

18 HUMAN FACTORS ENGINEERING*

18.1 Detailed Control Room Design Review**

Item I.D.1, "Control Room Design Review," of Task I.D., "Control Room Design," of the "NRC Action Plan Developed As a Result of the TMI-2 Accident" (NUREG-0660), states that operating reactor licensees and applicants for operating licenses will be required to perform a detailed control room design review (DCRDR) to identify and correct design discrepancies. The objective, as stated in NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent accidents, or to cope with them should they occur, by improving the information provided to the operators. Supplement 1 to NUREG-0737 confirmed and clarified the DCRDR requirement made in NUREG-0660. In response to Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct its DCRDR according to a schedule negotiated with the NRC.

In addition, the DCRDR was identified as a special program under TVA's Watts Bar Nuclear Performance Plan, which the staff evaluated in NUREG-1232, Volume 4. (For background information, see Section 1.13 of this SSER.) The evaluation that follows, picks up where NUREG-1232, Volume 4, left off.

18.1.1 Chronology of Major Events

The Tennessee Valley Authority (TVA or applicant) has conducted a detailed control room design review for Watts Bar Nuclear Plant, Unit 1. The staff performed an onsite audit from August 21 through 23, 1990, to assess the status of the DCRDR.

A partial chronology of the DCRDR for Watts Bar Nuclear Plant, Unit 1, is given below.

October 2, 1987	The applicant submitted DCRDR Summary Report to the NRC.
November 14-18, 1988	The NRC conducted an in-progress DCRDR audit.
April 28, 1989	The NRC forwarded a DCRDR safety evaluation to the applicant (included as Appendix L in SSER 6).
March 28, 1990	The applicant submitted a supplemental DCRDR summary report to the NRC, noting that the commitment by letter dated February 23, 1990, to correlate the Watts Bar Safety Evaluation Report (NUREG-0847), Appendix D, commitments to the DCRDR human engineering discrepancies (HEDs) was completed and available for review on site.

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

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

August 21-23, 1990	The NRC conducted a DCRDR audit.
October 24, 1990	The applicant submitted a list of and schedule for correcting all remaining Category 1 and 2 HEDs, and reaffirmed its commitment to correct all Unit 1 HEDs before fuel load.

This evaluation is based on the documentation and events listed above.

18.1.2 Evaluation

The staff's safety evaluation dated April 28, 1989, is incorporated in this SSER as Appendix L. The following sections, 18.1.2.1 through 18.1.2.9, correspond to Sections 2.1 through 2.9 of that safety evaluation, and provide supplemental or revised information.

18.1.2.1 Establishment of a Qualified Multidisciplinary Review Team

In the April 28, 1989, evaluation, the staff concluded that the applicant satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.2 System Function and Task Analysis To Identify Control Room Operator Tasks and Information and Control Requirements During Emergency Operations

Three specific human factors concerns were identified in Section 2.2 of the evaluation dated April 28, 1989. The staff's concerns included the following items for which no task analysis had been conducted:

(1) the six critical safety function trees
(2) the "symptoms" sections of the emergency procedures
(3) six emergency contingency actions (ECAs)

During the August 21-23, 1990, onsite audit, the NRC team found that the applicant had conducted a task analysis of the items listed above. The applicant conducted a supplemental task analysis using the same process used and approved by the staff for the original 1986 task analysis of draft emergency procedures.

The applicant's analysis of the six new plant-specific ECAs did not include a technical justification for deviations from the generic emergency response guideline (ERG) steps. However, these deviations were accounted for on Action-Information Requirements Detail (AIRD) forms used in the data recording.

Justification and documentation of step deviations (use of generic ERG steps or use of plant-specific procedure steps) was deferred to an upcoming emergency operating procedures verification and validation program. As described in Administrative Instruction (AI)-3.4, "Plant Operating Instructions," all deviations will be documented on deviation worksheets and resolved.

In summary, the applicant conducted a supplemental task analysis of the items identified in the April 28, 1989, safety evaluation. The staff concluded that the combination of the supplemental task analysis, along with the 1986 task analysis, satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.3 Comparison of Display and Control Requirements With Control Room Inventory

In the April 28, 1989, safety evaluation, the staff indicated that the applicant should compare the operator information and control requirements identified during the upgraded task analysis to the control room inventory to determine the availability and suitability of controls and displays. The staff evaluated the supplemental comparison of display and control requirements to the control room inventory for the additional task analysis activity discussed in the previous section. The staff concludes that this activity satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.4 Control Room Survey

In the April 28, 1989, safety evaluation, the staff concluded that the applicant satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.5 Assessment of Human Engineering Discrepancies To Determine Which Are Significant and Should Be Corrected

In the April 28, 1989, safety evaluation, the staff indicated that the applicant must demonstrate how each of the Watts Bar SER (NUREG-0847), Appendix D, commitments were satisfied. The applicant's March 28, 1990, supplemental summary report noted that the commitments were correlated with DCRDR HEDs and that documentation of this activity was available for review on site. On August 21-23, 1990, the audit team evaluated the applicant's documentation and determined that the commitments were either satisfactorily resolved or referred to a DCRDR HED for closure. The staff finds that the applicant's request by letter dated March 7, 1989, that the commitments be superseded by the Watts Bar summary report, is acceptable and that this issue has been resolved.

The April 28, 1989, safety evaluation identified two areas requiring additional HED assessment activity. First, any HEDs arising out of the supplemental task analysis and control room inventory activity needed to be assessed for safety significance. Second, HEDs 082 and 199 needed to be reassessed.

HED 082 indicated that set point adjustments on certain controllers can be changed accidentally by brushing up against the set point controls. The staff's concern with HED 082 was that relocation of the subject controllers might not sufficiently correct the HED. The applicant responded to this concern by implementing a formal process, described below under Section 18.1.2.7, requiring that HED 082 be reassessed. The applicant's response has satisfactorily addressed the staff's concern.

HED 199 indicated that certain valves could be manually opened with the phase A signal not reset. The staff requested additional justification for not correcting this HED. By letter dated March 28, 1990, the applicant provided eight justifications for the current design. One justification was as follows: "Since, according to the WBN [Watts Bar] FSAR, the plant has redundant isolation barriers for each of these penetrations, containment penetration isolation would be maintained, even when one valve is opened." The staff finds the applicant's justifications satisfactory for resolving this issue.

The applicant identified two additional HEDs (207 and 209) during the supplemental task analysis and inventory activity. These HEDs were assessed and documented in the applicant's March 28, 1990, submittal.

The staff concludes that the applicant satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.6 Selection of Design Improvements

In the April 28, 1989, safety evaluation, the staff concluded that the applicant satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.7 and 18.1.2.8 Verification That Selected Improvements Will Provide the Necessary Corrections Without Introducing New HEDs

In the April 28, 1989, safety evaluation, the staff found that no formal process existed to verify that the selected design improvements would result in the implementation of effective corrective actions and not introduce new HEDs. In order to satisfy this concern, the applicant implemented a formal process through Administrative Instruction (AI)-1.89, "Closing Out Control Room Human Engineering Concern Discrepancies." The staff reviewed AI-1.89 and determined that it satisfied this Supplement 1, NUREG-0737, requirement.

18.1.2.9 Coordination of Control Room Improvements With Changes From Other Programs Such as Safety Parameter Display System (SPDS), Operator Training, Regulatory Guide 1.97 Instrumentation, and Upgraded Emergency Operating Procedures

In the April 28, 1989, safety evaluation, the staff concluded that the applicant satisfied this Supplement 1, NUREG-0737, requirement.

18.1.3 Conclusions

The staff concludes that the DCRDR program implemented at the Watts Bar Nuclear Power Plant, Unit 1, satisfies the DCRDR programmatic requirements of Supplement 1, NUREG-0737. Any changes to commitments related to the DCRDR made by the applicant must be submitted to the staff for approval. Before startup, the staff plans to confirm by audit that the corrective actions which the applicant has committed to perform as a result of the DCRDR (i.e., all HEDs corrected before fuel load of Unit 1) have been completely and properly implemented. The staff will continue to track completion of all remaining DCRDR actions by proposed License Condition 37.

18.2 Safety Parameter Display System*

Item I.D.2, "Plant Safety Parameter Display Console," of Task I.D., "Control Room Design," of the "NRC Action Plan Developed As a Result of the TMI-2 Accident" (NUREG-0660), states that operating reactor licensees and applicants for operating licenses will be required to install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. Supplement 1 to NUREG-0737 confirmed and

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

clarified the SPDS requirement in NUREG-0737 that each licensee or applicant is required to submit a safety analysis describing the basis on which the selected variables are sufficient to assess the safety status of each identified function for a wide range of events, including symptoms of severe accidents. Licensees and applicants were also required to submit their specific implementation plans for an SPDS. TVA initially responded by letter dated April 15, 1983. The staff has been reviewing the Watts Bar SPDS for several years, as signified by requests for information dated September 14, 1984, and July 30, 1986.

18.2.1 Background

On April 12, 1989, the staff issued Generic Letter 89-06 (GL 89-06), "Task Action Plan Item I.D.2--Safety Parameter Display System--10 CFR 50.54 (f)," along with NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems." GL 89-06 asked all licensees to furnish one of the following:

(1) certification that the SPDS fully meets the requirements of NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342

(2) certification that the SPDS will be modified to fully meet the requirements of NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342 (The licensee or applicant was also asked to provide the implementation schedule for the modifications.)

If a certification could not be furnished, the licensee was asked to discuss the reasons for that finding and to report the compensatory action the licensee intends to take or has taken.

NUREG-1342 describes methods used by some licensees and applicants to implement SPDS requirements in a manner found acceptable by the staff. NUREG-1342 also describes SPDS features that the staff finds unacceptable and gives the reasons for finding them unacceptable.

By letter dated July 11, 1989, the applicant responded to GL 89-06 indicating that: (1) the Watts Bar, Unit 1, SPDS will be "operational" before startup following the first refueling outage and (2) a "functional" SPDS will be installed before fuel load. The applicant noted that the "functional" SPDS will comply with NUREG-0737, Supplement 1, with these exceptions: documented availability, resolution of operator comments during the first cycle, and verification of displayed data with main control room indications. The applicant committed to provide a supplemental response to GL 89-06 addressing certification of compliance with requirements of NUREG-0737, Supplement 1, within two months after the Unit 1 SPDS has become operational.

As stated in SSER 5, the staff conducted an onsite audit between August 21 and 23, 1990. The audit team consisted of a team leader from NRC's Human Factors Assessment Branch and two contractors from Science Applications International Corporation (SAIC). The audit was performed to assess the status of the SPDS with regard to eight Supplement 1, NUREG-0737, requirements. In addition, this report reflects the results of conference calls conducted on August 29 and September 25, 1990, and February 13, 1991, between the staff and TVA personnel to continue discussions regarding the Watts Bar Unit 1 SPDS. By letter dated November 1, 1990, the applicant responded to the NRC audit concerns. The audit agendum, attendee list, and slides prepared by TVA are publicly available documents located on microfiche 55983-023 in the Public Document Room (PDR) (Accession No. 9011300011).

18.2.2 Evaluation

The staff's evaluation of the Watts Bar Unit 1 SPDS follows. The evaluation is based on the previously identified documentation and the audit conducted in August 1990.

18.2.2.1 Concise Display of Critical Plant Variables to Control Room Operators

The evaluation of the concise display requirement included a review of physical location of displayed information and technical information organization within the display screens. Appropriate physical display grouping and technical information organization were the criteria used to judge whether the concise display requirement was satisfied.

The SPDS design is based on the six critical safety function trees in the emergency operating procedures. The organization of the critical safety function tree and its presentation are concise. The audit team found that the applicant's proposed design meets the NUREG-0737, Supplement 1, requirement.

18.2.2.2 Located Convenient to Control Room Operators

The evaluation of SPDS workspace location included an assessment of how the SPDS displays and controls support the operator's needs during emergency operations. This includes a determination of who is defined by the applicant as a user of the SPDS.

The applicant defines the primary users of the SPDS during an accident as the assistant shift operating supervisor (a licensed senior reactor operator responsible for all operations associated with that unit) and the shift technical advisor located at the unit operator work stations where there are two SPDS terminals. Although both terminals are expected to be operational, only one is required to be operational.

The applicant's proposal meets the NUREG-0737, Supplement 1, requirement.

18.2.2.3 Continuous Display of Plant Safety Status Information

The audit team evaluated the SPDS to determine if it continuously displays information about the five critical plant variables identified in Supplement 1 to NUREG- 0737. Two concerns were identified. First, the critical safety function status boxes were displaced during the display of prompt and error messages. Second, two new critical safety function trees were added in order to meet the continuous display requirement. These function trees were inconsistent with the critical safety function trees in the emergency procedures.

In order to address the staff's concerns, the applicant revised the SPDS design. In the revised design, the SPDS will not overwrite any of the status boxes (located at the bottom left corner of all displays) under any condition. This will ensure that the status boxes are continuously displayed. The six status boxes associated with the six critical safety function trees in the emergency procedures will be displayed as a functional group. The two boxes not associated with emergency procedures (decay heat removal and radioactivity control) will be replaced with a single box that alerts the operator to an out-of-tolerance parameter on the top-level 2PS1 display. These design modifications address the staff's concerns.

The staff concludes that the applicant's proposed design satisfies the NUREG-0737, Supplement 1, requirement.

18.2.2.4 The SPDS Should Rapidly and Reliably Aid the Control Room Operators in Determining the Safety Status of the Plant

The proposed TVA program to ensure reliability of the SPDS, including the identification of causes of unavailability and recommended corrective actions, follows:

- The three primary potential causes of SPDS unreliability are hardware, software, and sensor input readings. With respect to computer hardware (data acquisition, computer, processor, memory, and peripherals and display equipment), the applicant will generate a detailed instrument maintenance instruction (IMI) for calibration, hardware, operation, and overall maintenance. The IMI will be implemented by qualified maintenance personnel on a periodic basis, or as required for system repair, or as needed by the preventive maintenance (PM) procedure. TVA will upgrade the existing preventive maintenance to encompass the upgraded system.

- Regarding software reliability, the initial SPDS software and changes will undergo formal verification and validation to ensure that requirements are accurately specified, implemented, and tested. Software changes will be documented, approved, and controlled by qualified personnel and procedures.

- In order to minimize the possibility of bad sensor inputs or inaccurate SPDS display of sensor inputs, routine instrument loop calibration of sensors that provide input to the SPDS will include verification that the SPDS-displayed values are correct. The applicant's instrument surveillance instructions will incorporate these verifications. When a problem with the SPDS is detected it will be resolved by the maintenance request (MR) process. An MR can be initiated by any individual but will be primarily issued by personnel performing surveillance instructions, control room operators discovering system malfunctions, or computer maintenance personnel documenting required maintenance. The cause and corrective action taken to resolve the problem will be documented through the implementation of the IMI, and if it is determined that the SPDS is unavailable, the duration and unavailability will be documented. In addition, the instruction will generate a periodic calculation of system unavailability, and causes will be determined to identify trends and generic corrective actions to be taken.

The applicant's implemented program should satisfy this Supplement 1, NUREG-0737, requirement.

The applicant plans to complete a live system test program before declaring the SPDS operational. This testing cannot be completed until the end of the first refueling outage (see SSER 5, Section 18.2 regarding proposed License Condition 43).

18.2.2.5 The SPDS Shall Be Suitably Isolated From Electrical and Electronic Interference With Equipment and Sensors That Are Used for Safety Systems

Having reviewed the applicant's submittals dated March 27, September 5, and October 16, 1985; September 30, 1986; February 3, 1987; and November 1, 1990; the staff concludes that the applicant has satisfied the NUREG-0737, Supplement 1, requirement.

18.2.2.6 The SPDS Shall Be Designed To Incorporate Accepted Human Factors Principles

The review team evaluated the human factor aspects of the SPDS in the Unit 1 control room. This evaluation included a review of SPDS technical content, display formats, and workstation designs. The audit team identified several concerns, discussed in Appendix P, that the applicant has satisfactorily addressed.

In addition, TVA committed by letter dated November 1, 1990, to obtain regular control room operator input to and review of the SPDS design (display formats, content, logic flow, set points, etc.) at the simulator prior to SPDS acceptance. A form for recording SPDS comments by operations personnel will be developed and used as the vehicle for ensuring that future operator feedback is considered. Changes to the emergency operating instructions (EOIs) may affect the SPDS, therefore, the EOI writer's guide will include reference to the need to verify the SPDS as part of an EOI revision. SPDS set points, logic flows, and display formats will be verified against control room instrumentation, operating procedures, and system/sensor characteristics.

When the applicant fulfills its stated commitments, it should meet this requirement of Supplement 1 to NUREG-0737 regarding incorporation of accepted human factors principles into the SPDS design.

18.2.2.7 The Minimum Information Provided Shall Be Sufficient To Provide Information to Plant Operators About the Five Safety Functions Identified in Supplement 1 to NUREG-0737

The SPDS parameters identified in the applicant's response to GL 89-06 and letter dated November 1, 1990, were used as the basis for this evaluation. The review team found that the parameters selected for the proposed SPDS design satisfy this requirement of NUREG-0737, Supplement 1.

18.2.2.8 Procedures Should Be Developed and Operators Trained With and Without the SPDS Available

The applicant committed to develop a formal program for training with and without the SPDS.

The applicant committed to provide operator training that will consist of:

- Hot license and requalification training of control room operators to include formal classroom and hands-on (simulator) training in the SPDS design basis; access to, use, and interpretation of SPDS displays and data; and integration of the SPDS into control room operations.

- Specific, one-time training of control room operators in identifying plant transients and mitigating accidents, both with and without the SPDS.

Before fuel is loaded, plant operations personnel will be given their initial training in SPDS use.

It was the review team's judgment that the applicant should satisfy this NUREG-0737, Supplement 1, requirement.

18.2.3 Conclusion

The Watts Bar Nuclear Power Plant, Unit 1, SPDS was in the design/development phase during the August 21 through 23, 1990, audit. On the basis of the audit, phone conferences to clarify docketed information, and the applicant's letter dated November 1, 1990, it is the staff's judgment that the applicant's SPDS design should meet the NUREG-0737, Supplement 1, SPDS requirements. The staff requests that the applicant notify the NRC by letter when the "functional" SPDS is installed. The staff may confirm by audit that the corrective actions described above have been completely and properly implemented for the "functional" and/or "operational" SPDS. The staff will continue to track all SPDS activities by Confirmatory Issue 43.

APPENDIX A

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

NRC Letters, Memoranda, and Summaries

June 28, 1985	Letter from E. G. Adensam to H. G. Parris (TVA), regarding "Flexibility Requirement in Pipe Support Base Plate Design Using Concrete Expansion Anchors at Watts Bar Nuclear Plant, Units 1 and 2."
October 31, 1989	Letter from S. Black to O. D. Kingsley (TVA), regarding "Watts Bar Nuclear Plant, Unit 1--Validation of SASSI Computer Code for Soil-Structure Interaction Analysis."
September 4, 1990	Summary of August 16, 1990, meeting with utility regarding status of corrective action program and related issues.
October 2, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting the February 2 and May 31, 1989, responses to Generic Letter 88-17, "Loss of DHR."
October 9, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information supplementing utility April 30, 1990, submittal regarding preservice inspection program.
October 10, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) accepting response to Bulletin 88-001, "Defects in Westinghouse Circuit Breakers."
October 19, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting additional information regarding Amendment 63 to FSAR Chapter 12.
October 19, 1990	Memorandum from P. S. Tam to NRC Document Control Desk, requesting that enclosed documents be made available in the PDR.
October 24, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) advising that March 22, 1990, response to Generic Letter 89-19 regarding USI A-47 concerning safety implications of control systems is complete.
October 26, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) forwarding safety evaluation accepting proposed changes to FSAR regarding the corrective action program plan on instrument lines.

October 30, 1990	Letter from S. D. Ebneter to O. D. Kingsley (TVA) regarding the requirements for establishment and maintenance of QA records.
November 6, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) forwarding status of NRC understanding of status of implemented generic safety issues noted in Generic Letter 90-04.
November 1, 1990	Letter from F. J. Hebdon to TVA forwarding Supplement 5 to SER (NUREG-0847) for utility review.
November 19, 1990	Summary of October 3, 1990, meeting with utility regarding overview and general plant status.
November 20, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) approving transition from ASME Code Section III to Section XI welding requirements.
November 29, 1990	Letter from P. S. Tam to O. D. Kingsley (TVA) requesting format documentation regarding seismic qualification of cable trays and conduits.

<u>TVA Letters</u>

August 22, 1985	Letter from J. A. Domer to E. Adensam (NRC), responding to NRC Bulletin 79-02.
February 2, 1989	Letter from M. J. Ray to NRC responding to Generic Letter 88-17 "Loss of Decay Heat Removal."
May 31, 1989	Letter from M. J. Ray to NRC providing additional information to respond to Generic Letter 88-17, "Loss of Decay Heat Removal."
April 30, 1990	Letter from E. G. Wallace to NRC providing additional information on the preservice inspection program.
May 9, 1990	Letter from E. G. Wallace to NRC, regarding "Watts Bar Nuclear Plant (WBN)--Revision to Corrective Action Program (CAP) Plan for Seismic Analysis."
June 8, 1990	Letter from E. G. Wallace to NRC forwarding "Second Annual Report of Employee Concerns Special Program Corrective Actions Implementation."
June 8, 1990	Letter from E. G. Wallace to NRC advising that utility committed to conforming to Regulatory Guide 1.97, Revision 2.
June 14, 1990	Letter from E. G. Wallace to NRC forwarding revised Central Emergency Control Center (CECC) Emergency Plan Implementing Procedures (EPIPs).

June 15, 1990	Letter from E. G. Wallace to NRC forwarding supplemental information on cable issues regarding ampacity and large low-power cables.
June 15, 1990	Letter from E. G. Wallace to NRC forwarding supplemental information on facility cable issues, per May 22, 1990, commitment.
June 15, 1990	Letter from E. G. Wallace to NRC forwarding information that addresses issue of cable damage as result of cable pullbys.
June 19, 1990	Letter from E. G. Wallace to NRC forwarding utility response to Bulletin 89-003, "Potential Loss of Required Shutdown Margin During Refueling Operations."
June 27, 1990	Letter from E. G. Wallace to NRC forwarding response to request for additional information regarding Topical Report TVA-NPOD89, Revision 1.
June 28, 1990	Letter from E. G. Wallace to NRC responding to Generic Letter 90-04, "Request for Information on Status of Licensee Implementation of Generic Safety Issues Resolved With Imposition of Requirements or Corrective Actions."
July 5, 1990	Letter from E. G. Wallace to NRC forwarding clarification of responses to updated Regulatory Guide 9.3 information, per NRC June 6, 1990, letter.
July 12, 1990	Letter from E. G. Wallace to NRC forwarding revision to utility February 23, 1989, response to Generic Letter 88-14 regarding instrument air supply system problems affecting equipment.
July 12, 1990	Letter from E. G. Wallace to NRC forwarding revised CECC EPIPs.
July 20, 1990	Letter from E. G. Wallace to NRC forwarding Revision 3 to "Watts Bar Nuclear Plant Q-List Corrective Action Program Plan."
July 27, 1990	Letter from E. G. Wallace to NRC forwarding revisions to physical security plan.
July 30, 1990	Letter from E. G. Wallace to NRC forwarding Revision 1 to TVA welding project final report.
July 30, 1990	Letter from E. G. Wallace to NRC forwarding description of system testing to be performed under prestart test corrective action program plan.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding Revision 2 to corrective action program plan for cable issues.

July 31, 1990	Letter from E. G. Wallace to NRC forwarding proposed updates to FSAR Section 3.11 regarding equipment qualification programs.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding Revision 3 to corrective action program plan for replacement items (piece parts).
July 31, 1990	Letter from E. G. Wallace to NRC forwarding proposed revisions to FSAR regarding the welding corrective action program plan.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding Revision 3 to corrective action program plan for design baseline and verification program.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding a restatement of NRC concerns in safety evaluation report for Watts Bar Nuclear Performance Plan and associated TVA responses.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding a clarification of applicable plant mode following a postulated main steam line break, per NUREG-1232, Volume 4.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding updates to FSAR sections regarding hanger and analysis update program corrective action program plan for large bore piping.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding proposed revisions to the FSAR regarding instrument lines corrective action program plan.
July 31, 1990	Letter from E. G. Wallace to NRC notifying NRC of completion of heat code traceability corrective action program plan.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding proposed updated to FSAR Section 3.2.2.5 regarding plant heat code traceability program.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding proposed revisions to the FSAR regarding cable issues and electrical issue corrective action program plans.
July 31, 1990	Letter from E. G. Wallace to NRC forwarding Revision 2 to "Welding Corrective Action Program Plan" and Revision 1 to "TVA Welding Project Watts Bar Nuclear Plant, Phase 1 Report."
August 6, 1990	Letter from M. O Medford to NRC responding to Bulletin 88-008, "Thermal Stresses in Piping Connected to Reactor Coolant System."

August 13, 1990	Letter from E. G. Wallace to NRC forwarding information that clarifies organizational Topical Report TVA-NPOD89, per July 27, 1990, telephone conversation.
August 16, 1990	Letter from E. G. Wallace to NRC forwarding proprietary and non-proprietary WCAP-12546 and WCAP-12547, "Watts Bar Unit 1 Evaluation for Tube Vibration Induced Fatigue," in response to Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes."
August 24, 1990	Letter from E. G. Wallace to NRC responding to Generic Letter 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2, Part 2, 'Vendor Interface for Safety-Related Components'."
August 31, 1990	Letter from E. G. Wallace to NRC advising that utility considers concrete quality evaluation at facilities complete.
August 31, 1990	Letter from E. G. Wallace to NRC discussing control of microbiologically induced corrosion.
August 31, 1990	Letter from E. G. Wallace to NRC forwarding utility approach to satisfying intent of Regulatory Guide 1.97, Revision 2.
September 12, 1990	Letter from E. G. Wallace to NRC forwarding damping values for cable tray, conduit, and HVAC systems.
September 14, 1990	Letter from E. G. Wallace to NRC responding to verbal commitment made by utility during August 2, 1990, presentation regarding validation program using worst-case method.
September 19, 1990	Letter from E. G. Wallace to NRC advising that the report regarding the loose parts monitoring system will be submitted within 90 days following completion of startup test program.
October 9, 1990	Letter from E. G. Wallace to NRC notifying of significant change in peak cladding temperature for small-break LOCA analysis.
October 10, 1990	Letter from E. G. Wallace to NRC forwarding corrected pages to Revision 2 to cable issues corrective action program plan.
October 11, 1990	Letter from E. G. Wallace to NRC advising of the installation of additional accident-monitoring instrumentation, per NUREG-0737, Item II.F.1.
October 11, 1990	Letter from E. G. Wallace to NRC forwarding response to NRC comments from August 1-3, 1990, meeting regarding cable issues corrective action program plan.

October 19, 1990	Letter from E. G. Wallace to NRC responding to inquiry on how utility will ensure that snubbers required for Unit 1 operation will function.
October 19, 1990	Letter from E. G. Wallace to NRC forwarding supplemental response regarding seismic design for certain safety-related vertical steel tanks.
October 22, 1990	Letter from E. G. Wallace to NRC forwarding response to August 13, 1990, request regarding bypassed and inoperable status indication system.
October 24, 1990	Letter from E. G. Wallace to NRC advising that utility realigned welding program to comply with ASME Sections III and XI requirements, per NRC July 2, 1987, letter.
October 24, 1990	Letter from E. G. Wallace to NRC forwarding schedule for CRDR corrective action program plan for human engineering discrepancies.
November 1, 1990	Letter from E. G. Wallace to NRC forwarding response to NRC audit concerns regarding NUREG-0737, Supplement 1, Item I.D.2 on SPDS, per Generic Letter 89-06.
November 5, 1990	Letter from E. G. Wallace to NRC forwarding proposed safeguards contingency plan and proposed security personnel training and qualification plan.
November 5, 1990	Letter from E. G. Wallace to NRC forwarding supplemental information for resolution of cables issues corrective action program plan.
November 7, 1990	Letter from E. G. Wallace to NRC forwarding Revision 4 to "Watts Bar Nuclear Plant Q-List Corrective Action Program Plan."
November 13, 1990	Letter from E. G. Wallace to NRC clarifying June 16, 1990, response to Bulletin 89-001, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs."
November 14, 1990	Letter from E. G. Wallace to NRC informing that action items resulting from security program enhancements review were addressed and documented, per December 11, 1986, letter.
November 19, 1990	Letter from E. G. Wallace to NRC forwarding Revision 5 to CECC EPIP-9, "Emergency Radiological Monitoring Procedures."
December 18, 1990	Letter from E. G. Wallace to NRC, regarding documentation of resolutions to open issues--FSAR Amendment 64.

APPENDIX B

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APPENDIX E

PRINCIPAL CONTRIBUTORS

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*Paul Cortland died unexpectedly on February 4, 1991; he was instrumental in the preparation of Section 1.14 of this supplement. His colleagues at the Nuclear Regulatory Commission will miss him personally and professionally.

APPENDIX G

ERRATA TO WATTS BAR SAFETY EVALUATION REPORT

Section	Page	Change
5.4.3	5-21	In the second paragraph, line 14, delete the incomplete sentence, "If the Diablo Canyon results have been reviewed and their applicability to Watts Bar evaluated."

APPENDIX K

SAFETY EVALUATION:
INSTRUMENT LINES CORRECTIVE ACTION PROGRAM PLAN

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSTRUMENT LINE

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

The staff had previously reviewed and accepted the Corrective Action Program (CAP) plan for instrument lines and the staff evaluation is documented in NUREG-1232 Volume 4. By letter dated July 31, 1990, TVA has submitted proposed FSAR revisions on the instrument lines CAP plan.

2.0 EVALUATION

In the proposed FSAR revision of Section 7.1.2.2, TVA has added a requirement that instrument lines for safety systems should meet the independence criteria identified in criterion 22 of General Design Criteria and IEEE-279-1971, Section 4.6. Any exception to the requirement will be technically justified and documented in design basis documents. The staff finds this approach acceptable and plans to perform an inspection to audit the design basis documents.

Also in the proposed revisions, TVA has deleted Figure 7.2-3. This figure identified the design of the pressurizer water level instrumentation. TVA's justification for the deletion is that there is no need to provide this kind of detail in the FSAR. The staff finds the deletion acceptable.

In Section 7.2.2.3.3 of the FSAR, TVA has deleted the justification for using the shared taps for redundant instrumentation. This would be acceptable to the staff as long as the justification is included in the design basis documents for the affected instruments.

3.0 CONCLUSION

Based on the above, the staff has concluded that the proposed changes to the FSAR on the instrument lines CAP plan are acceptable.

Principal Contributor : Hukam C. Garg

Dated : October 1990

APPENDIX L

SAFETY EVALUATION: DETAILED CONTROL ROOM DESIGN REVIEW*

*Previously issued as enclosure to letter from S. C. Black (NRC) to O. D. Kingsley (TVA), April 29, 1989.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING THE DETAILED CONTROL ROOM DESIGN REVIEW

WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-390/391

1.0 INTRODUCTION

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

Following completion of the Watts Bar Nuclear Plant (WBN) DCRDR, TVA submitted a Summary Report to the NRC on October 2, 1987. The Summary Report was reviewed by the cognizant NRC staff and by Science Applications International Corporation (SAIC). The results of the staff and SAIC review of the TVA Summary Report indicated a need for additional information. A pre-implementation audit was scheduled in order to obtain this information and to resolve several concerns.

The audit was conducted at WBN between November 11 and 18, 1988. The audit team evaluated the WBN DCRDR in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews," and the nine DCRDR requirements contained in NUREG-0737, Supplement 1. A summary evaluation of the results of the audit are discussed in the following paragraphs, while the attached Technical Evaluation Report (TER) provides a detailed discussion of the audit activity.

2.0 EVALUATION

The pre-implementation audit was conducted by NRC staff with the assistance of SAIC and COMEX Corporation personnel, who provided expertise in Human Factors Engineering and Reactor Operations, respectively. The purpose of the audit was to evaluate whether TVA had met the nine DCRDR requirements in NUREG-0737, Supplement 1. This SE is based on the review of the WBN DCRDR Summary Report and the results of the audit. The assessment which follows is arranged in order of the nine NUREG-0737, Supplement 1, DCRDR requirements.

2.1 Establishment of a Qualified Multidisciplinary Review Team

The WBN DCRDR team consisted of an appropriate mix of specialists in the fields of human factors engineering, nuclear engineering, instrumentation and controls, and reactor operations. It is the staff's judgment that the WBN DCRDR team satisfied the requirement for a multidisciplinary review team.

2.2 System Function and Task Analyses to Identify Control Room Operators Tasks and Information and Control Requirements During Emergency Operations

The approach to System Function and Task Analysis used by the WBN DCRDR team, which is described in detail in Section 4.0 of the Summary Report, satisfactorily achieved the goal of the required task analysis effort. However, there were three areas of the Westinghouse Owners Group Emergency Response Guidelines (ERG) based emergency procedures requiring task analysis for which task analysis was not conducted. These areas are:

a. the six critical safety function trees;

b. the symptoms sections of the emergency procedures, and;

c. six Emergency Contingency Actions (ECAs)

ECA 1.1	Loss of Emergency Cooling Circulation
ECA 1.2	Loss of Coolant Accident Outside Containment
ECA 2.1	Uncontrolled Depressurization of All Steam Generators
ECA 3.1	Steam Generator Tube Rupture Loss of Coolant Accident with Subcooled Recovery
ECA 3.2	Tube Rupture Plus Loss of Coolant Accident with Saturated Recovery
ECA 3.3	Steam Generator Tube Rupture With Loss of Pressurizer Pressure Control.

Because of the above deficiencies, the staff finds that the licensee has not met the requirement to perform a System Function and Task Analysis of Control Room Operators tasks. In order to meet this requirement, TVA needs to conduct additional System Function and Task Analysis on the ERG-based items identified above, and to document the task analysis activity and results in a supplemental DCRDR summary report.

2.3 Comparison of Display and Control Requirements with a Control Room Inventory

The operator information and control requirements identified during the task analysis were compared to the actual control room to determine the availability and suitability of controls and displays. All discrepancies identified were appropriately documented and included in plans for correction.

The audit team found that the WBN DCRDR team conducted a successful comparison of display and control requirements versus the control room inventory for those areas for which task analysis had been performed. However, because there still exist some areas requiring task analysis, as discussed in paragraph 2.2, the staff finds that the licensee has not satisfied this requirement. In order to meet this requirement, TVA must conduct a supplemental comparison of display and control requirements to the control room inventory for the additional task analysis activity discussed in paragraph 2.2. This activity should be included in the supplemental DCRDR summary report which documents the additional System Function and Task Analysis.

2.4 Control Room Survey

The WBN DCRDR team conducted a control room survey using the criteria provided in NUREG-0700, modified as necessary to be plant specific. Additionally, the DCRDR team conducted extensive interviews with control room operators to identify human engineering problems. It is the staff's assessment that the DCRDR team conducted a thorough control room survey, and that the licensee has met the corresponding NUREG-0737 requirement.

2.5 Assessment of Human Engineering Discrepancies (HEDs) to Determine Which Are Significant and Should be Corrected

In general, the HED assessment process conducted at WBN adequately determined which HEDs should be corrected based on their potential impact on plant safety. The methodology employed by TVA included evaluation of the safety significance of each HED and of the aggregate effects of HEDs.

The audit team identified two areas requiring additional HED assessment activity. First, any HEDs arising out of the additional task analysis and control room inventory activity to be conducted must be assessed for significance. Second, TVA should reassess HEDs 082 and 199 in order to address the audit team's concerns, which are detailed in Section 2.5 of the attached TER.

The staff concludes that the licensee has not met the requirement to assess HEDs for significance to determine which ones require correction. In order to meet this requirement, TVA must assess the significance of any new HEDs arising from the additional task analysis to be conducted, as discussed in paragraph 2.2 of this SE, and address the issues associated with HEDs 082 and 199. The results of the additional assessment activity should be documented in the supplemental DCRDR summary report.

2.6 Selection of Design Improvements

This attribute of the DCRDR requires licensees to develop design changes and implementation schedules to remedy the HEDs identified for correction. The audit team found that WBN DCRDR team's development of conceptual designs to fix the HEDs was thorough and technically adequate, and that TVA plans to correct all the HEDs identified for correction in the DCRDR summary report prior to Unit 1 fuel load.

The staff finds that TVA has met the NUREG-0737, Supplement 1, requirement for selection of design improvements.

2.7 Verification that Selected Design Improvements Will Provide the Necessary Correction

The audit team found that no formal mechanism existed to verify that the selected design improvements would result in the implementation of effective corrective action for their respective HEDs. As a result, the staff finds that TVA did not meet the requirement to verify that selected design improvements would provide the necessary correction of HEDs. In order to satisfy this requirement, TVA should implement the necessary formal process and should report its implementation in the supplemental DCRDR summary report.

2.8 Verification that Selected Design Improvements Will Not Introduce New HEDs

The audit team found that no formal mechanism existed to verify that the selected design improvements to be implemented would not result in the creation of any new HEDs. As a result, the staff finds that TVA did not meet the requirement to verify that selected design improvements will not create any new HEDs. In order to satisfy this requirement, TVA should implement the necessary formal process, and should report its implementation in the supplemental DCRDR summary report.

2.9 Coordination of Control Room Improvements with Changes from Other Programs, Such as the Safety Parameters Display System, Operator Training, Regulatory Guide 1.97 Instrumentation, and Upgraded Emergency Operating Procedures

As detailed in the attached TER, the staff concludes that TVA has met the NUREG-0737, Supplement 1, requirement for coordination of control room improvements with changes from other programs which affect the control room and the operators' emergency response capability.

3.0 CONCLUSIONS

In summary, the staff concludes that the DCRDR activities for Watts Bar Nuclear Plant, Units 1 and 2, will meet the requirements of NUREG-0737, Supplement 1, when TVA provides NRC with a supplemental DCRDR summary report which adequately addresses the concerns described in the following sections of this SE, and detailed in the corresponding sections of the attached TER:

2.2 System Function and Task Analyses to Identify Control Room Operators Tasks and Information and Control Requirements During Emergency Operations

2.3 Comparison of Display and Control Requirements with a Control Room Inventory

2.5 Assessment of HEDs to Determine Which Are Significant and Should be Corrected

2.7 Verification that Selected Design Improvements Will Provide the Necessary Correction

2.8 Verification that Selected Design Improvements Will Not Introduce New HEDs.

All of these concerns should be resolved prior to the issuance of an operating license for WBN Unit 1.

We further note that TVA's plan to correct the HEDs prior to fuel load represents a significant, and very positive, commitment to enhancing the safety posture of the Watts Bar Nuclear Plant. We would like to reiterate that any changes to this commitment, either in terms of schedule or content, should be identified in writing, in a timely manner, to NRC.

SAIC-88/1821

TECHNICAL EVALUATION REPORT OF THE DETAILED CONTROL ROOM DESIGN REVIEW FOR TENNESSEE VALLEY AUTHORITY'S WATTS BAR NUCLEAR PLANT, UNIT 1

TAC NO. M63655

January 17, 1989

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Contract NRC-03-87-029

Task Order No. OSP-5

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TECHNICAL EVALUATION REPORT
OF THE
DETAILED CONTROL ROOM DESIGN REVIEW
FOR
TENNESSEE VALLEY AUTHORITY'S
WATTS BAR NUCLEAR PLANT, UNIT 1

1.0 INTRODUCTION

The Tennessee Valley Authority (TVA) submitted a generic Detailed Control Room Design Review (DCRDR) Program Plan to the Nuclear Regulatory Commission (NRC) on June 9, 1983 (Reference 1) in order to satisfy the Program Plan requirements of NUREG-0737, Supplement 1 (Reference 2) for the Sequoyah, Watts Bar, Bellefonte and Browns Ferry Nuclear Plants. The Program Plan was resubmitted September 13, 1983 (Reference 3) to correct duplicating errors in the original plan. The NRC staff reviewed the submittal with reference to the nine DCRDR requirements of NUREG-0737, Supplement 1, and the guidance provided in NUREG-0700 (Reference 4) and draft NUREG-0801 (Reference 5).

NUREG-0737, Supplement 1 requires that a Program Plan be submitted within two months of the start of the DCRDR. Consistent with the requirements of NUREG-0737, Supplement 1, the Program Plan should describe how the following elements of the DCRDR will be accomplished:

1. Establishment of a qualified multidisciplinary review team.

2. Function and task analyses to identify control room operator tasks and information and control requirements during emergency operations.

3. A comparison of display and control requirements with a control room inventory.

4. A control room survey to identify deviations from accepted human factors principles.

5. Assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected.

6. Selection of design improvements.

7. Verification that selected design improvements will provide the necessary correction.

8. Verification that improvements will not introduce new HEDs.

9. Coordination of control room improvements with changes from other programs such as Safety Parameter Display System, operator training, Regulatory Guide 1.97 instrumentation, and upgraded Emergency Operating Procedures (EOPs).

The staff comments resulting from the NRC review of the TVA DCRDR Program Plan were forwarded to TVA by letter dated November 17, 1983 (Reference 6). Based on the Program Plan review, the staff concluded that TVA addressed most of the nine requirements of a DCRDR specified in NUREG-0737, Supplement 1. However, the staff determined that certain elements, notably the task analysis, needed strengthening to provide reasonable assurance that the DCRDRs based on the plan would produce results that satisfy NRC requirements.

A meeting between NRC and TVA was held on June 14, 1984, in order to provide further detailed information and address the staff's Program Plan review concerns. As a result of this meeting, NRC indicated to TVA that an opportunity to more completely assess TVA's methodology for performing the system function and task analysis activity may involve an in-progress audit at Watts Bar. However, no in-progress audit was conducted at Watts Bar during the DCRDR.

At the end of the DCRDR, licensees/applicants are required by NUREG-0737, Supplement 1 to submit a Summary Report to NRC, which must, as a minimum:

1. Outline proposed control room changes.

2. Outline proposed schedules for implementation.

3. Provide summary justification for HEDs with safety significance to be left uncorrected or partially corrected.

Tennessee Valley Authority (TVA) submitted a Summary Report for the Watts Bar Nuclear Plant Units 1 and 2 to the NRC on October 2, 1987 (Reference 7). The Summary Report was reviewed by Science Applications International Corporation (SAIC) personnel and a pre-implementation audit was conducted from November 14 through November 18, 1988. The audit team consisted of an NRC staff member, an SAIC representative, and a representative from Comex Corporation. Together, the team represented the disciplines of nuclear systems engineering, reactor operations, and human factors engineering.

This Technical Evaluation Report reflects the consolidated observations, findings, and conclusions of the audit team members. A list of audit meeting attendees is provided in Attachment 1 and the audit agenda is provided in Attachment 2.

2.0 EVALUATION

The purpose of the evaluation was to determine whether the nine DCRDR requirements of NUREG-0737, Supplement 1 had been satisfied. The evaluation was performed by comparing the information provided by TVA with the criteria in NUREG-0800, Section 18.1, Rev. 0, Appendix A of the Standard Review Plan (Reference 8). The reviewers' evaluation of the DCRDR for the Watts Bar Nuclear Plant, and a summary of the criteria from the Standard Review Plan are provided below.

2.1 Establishment of a Qualified Multidisciplinary Review Team

The organization for conduct of a successful DCRDR can vary widely but is expected to conform to some general criteria. Overall administrative leadership should be provided by a utility employee, who should be given sufficient authority to ensure that the DCRDR team is able to carry out its mission. A core group of specialists in the fields of human factors engineering and nuclear engineering are expected to participate with assistance as required from personnel in other disciplines. Human factors expertise should be included in the staffing for most, if not all, technical tasks. Finally, the DCRDR team should receive an orientation briefing on DCRDR purpose and objectives which contributes to the success of the DCRDR. NUREG-0800, Section 18-1, Appendix A describes criteria for the multidisciplinary review team in more detail.

The overall administrative leadership of the DCRDR team was provided by a TVA employee. His successor as the DCRDR administrator will continue to manage the project through the modification implementation phase. The Watts Bar DCRDR study team consisted of a core group of specialists in the fields of nuclear engineering, instrumentation and control engineering, reactor operations, and human factors engineering. Essex Corporation was contracted to provide human factors support. Each TVA DCRDR team member was given a two-day course in human factors engineering and control room design, including the purpose and objectives of a DCRDR.

The audit team evaluated the staffing for each technical task and determined that the appropriate expertise was included in the DCRDR team. It is the audit team's judgment that TVA has met the NUREG-0737, Supplement 1 requirement for a qualified multidisciplinary review team.

2.2 System Function and Task Analysis

The purpose of the system function and task analysis is to identify the control room operators' tasks during emergency operations and to determine the information and control capabilities the operators need in the control room to perform those tasks. An acceptable process for conducting the function and task analysis is as follows:

1. Analyze the functions performed by systems in responding to transients and accidents in order to identify and describe those tasks operators are expected to perform.

2. For each task identified in Item 1 above, determine the information (e.g., parameter, value, status) which signals the

need to perform the task, the control capabilities needed to perform the task, and the feedback information needed to monitor task performance.

3. Analyze the information and control capability needs identified in Item 2 above to determine appropriate characteristics for displays and controls to satisfy those needs.

The Watts Bar DCRDR task analysis methodology was presented in Section 4.0 of the Summary Report.

The function and task analysis efforts covered all of the site-specific emergency response guidelines, developed from the generic Westinghouse Owners' Group (WOG) Emergency Response Guidelines (ERGs), High Pressure version, Rev. 1, September 1983. Differences between the generic and plant-specific ERGs were considered. A list of emergency operating procedures that were analyzed during task analysis is provided in Attachment 3.

The audit team selected action steps from both the generic and supplemental ERGs and traced the methodology under which each of the task analysis methods were performed to determine the adequacy of the methods used and availability of documentation. It was noted that the sample set of tasks reviewed by the audit team were thoroughly analyzed, including the alternate Response Not Obtained Column Tasks, Cautions, Warnings and Notes. In addition, documentation was adequate and was readily available and auditable.

The system function and task analysis was based on the December, 1985 version of plant specific emergency operating procedures. Based on an evaluation of the licensee's results, the audit team identified the following concerns:

a. The DCRDR team did not perform a task analysis of the six ERG based critical safety function trees.

b. The DCRDR team did not perform a task analysis of six ERG based Emergency Contingency Actions (ECAs) including:

ECA 1.1 Loss of Emergency Cooling Circulation
ECA 1.2 Loss of Coolant Accident Outside Containment
ECA 2.1 Uncontrolled Depressurization of all Steam Generators
ECA 3.1 Steam Generator Tube Rupture Loss of Coolant Accident with Subcooled Recovery
ECA 3.2 Tube Rupture Plus Loss of Coolant Accident with Saturated Recovery
ECA 3.3 Steam Generator Tube Rupture With Loss of Pressurizer Pressure Control

c. The DCRDR team did not perform an analysis of the Symptoms sections of the emergency procedures.

Because the DCRDR team did not perform the necessary system function and task analysis for the areas described above, it is the audit team's judgment that the licensee did not meet the NUREG-0737, Supplement 1 requirement for a function and task analysis. In order to meet the requirement, TVA should conduct an additional task analysis effort that addresses the concerns listed above.

2.3 Comparison of Display and Control Requirements with a Control Room Inventory

The purpose of comparing display and control requirements to a control room inventory is to determine the availability and suitability of displays and controls required to perform the ERGs. The success of this element depends on the quality of the function and task analysis and the control room inventory. The control room inventory should be a complete representation of displays and controls currently in the control room. The inventory should include appropriate characteristics of current displays and controls to allow meaningful comparison to the results of the function and task analysis. Unavailable or unsuitable displays and controls should be documented as human engineering discrepancies (HEDs).

The verification of instrument and control availability and suitability was accomplished by comparing the operator's requirements during emergency operations derived from the task analysis activities to the equipment in the Watts Bar control room. A "walk- and talk-through" by DCRDR team members and qualified operators was performed for each of the steps analyzed on the task analysis worksheets. "Human Factors Guidelines" checksheets were used to evaluate the adequacy of the instrument/control demonstrated by the operator, and the information/control equipment for fulfilling the task analysis requirement. Real-time simulations were also performed using time-dependent emergency procedures to evaluate perceptual-cognitive loading, communications, and spatial relationships. Potential HEDs were documented as human engineering concerns (HECs) during the review phase, and then converted to HEDs during the assessment activity.

The audit team found that the Watts Bar DCRDR team conducted a successful comparison of display and control requirements versus the control room inventory for those areas for which system function and task analyses had been performed. However, because there still exist some areas to be subjected to system function and task analysis, as discussed in paragraph 2.2, it is the audit team's assessment that the licensee does not meet the NUREG-0737, Supplement 1 requirement for a comparison of display and control requirements with the control room inventory. In order to meet this requirement the licensee should conduct a supplemental comparison of display and control requirements to the control room inventory for the additional task analysis of critical safety function trees, ECAs, and Symptoms.

2.4 Control Room Survey

The key to a successful control room survey is a systematic comparison of the control room to accepted human engineering guidelines and human factors principles. One accepted set of human engineering guidelines is provided in Section 6 of NUREG-0700 (Reference 4); however, other accepted human factors standards may be chosen. Discrepancies should be documented as HEDs.

NUREG-0737, Supplement 1 does not require the performance of operator interviews as a formal part of the DCRDR. However, NUREG-0700 states that such surveys are needed to make sure that problems encountered in plant operation or in preparations for operation are addressed.

The licensee performed a comprehensive survey of operator concerns through the use of a detailed control room operations questionnaire followed up by interviews of the individual operators. Twenty operators were involved in the survey.

The audit team selected eight of the operator concerns from the raw data collected by the licensee and traced each of these through all phases of the DCRDR process. The concerns selected were those which were mentioned by a significant percentage of the interviewees as human engineering problems. The audit team was able to trace every concern through each step of the assessment process, and in all cases the concern was satisfied in an appropriate manner.

The human engineering guidelines used for the control room surveys were a modified version of Section 6 of NUREG-0700. Modifications to the checklists were primarily alterations of general guidelines to make them plant specific. Clarifications of the guidelines were made as appropriate. In addition, operator interview questions were referenced in the guidelines so that the person performing the survey was able to coordinate the operator interview questions and survey guidelines. It is the audit team's judgment that the survey guidelines and process for conducting the survey are comprehensive and thorough.

It was the audit team's judgment that TVA met the NUREG-0737, Supplement 1 requirement for a control room survey.

2.5 Assessment of Human Engineering Discrepancies (HEDs) to Determine Which Are Significant and Should Be Corrected

Based on the guidance of NUREG-0700 and the requirements of NUREG-0737, Supplement 1, all HEDs should be assessed for significance. The potential for operator error and the consequence of that error in terms of plant safety should be systematically considered in the assessment. Both the individual and aggregate effects of HEDs should be considered. The result of the assessment process is a determination of which HEDs should be corrected because of their potential impact on plant safety. Decisions on whether HEDs are safety-significant should not be compromised by consideration of such issues as the means and potential costs of correcting HEDs.

The assessment process at Watts Bar was conducted according to the Program Plan but cannot be judged complete until the licensee performs additional task analysis work and completes the comparison of the operator and display and control requirements to the control room inventory to any additional human engineering discrepancies.

The review team also identified concerns regarding the assessment and disposition of two safety significant HEDs.

082 Accidental changing of controller setpoints. - The concern is that the subject controllers will be relocated under a relocation HED. Therefore the concerns that caused the origination of 082 should be reassessed at the controller's relocation on the new panel (M-27-B).

199 Certain valves could be opened with Phase A isolation not reset. This was the result of a North Anna 2 licensee event report 82-010. It was found that the valves could be reopened from the control room by holding the control switch open when the Phase A isolation signal was present. TVA's justification for not correcting this HED, if it is in fact a HED, was that it would require a deliberate action on the part of the operator. No investigation was made by TVA to determine if the control circuit was functioning correctly. Additional engineering justification is required.

It was the review team's judgment that the licensee did not meet the requirement of NUREG-0737, Supplement 1 for an assessment of human engineering discrepancies. In order to meet this requirement, TVA should assess the significance of any new HEDs arising from the additional system function and task analysis to be conducted, and address the issues associated with HEDs 082 and 199.

2.6 Selection of Design Improvements

The purpose of selecting design improvements is to determine corrections to HEDs identified from the review phase of the DCRDR. Selection of design improvements should include a systematic process for the development and comparison of alternative means of resolving HEDs. Furthermore, according to NUREG-0737, Supplement 1, the licensee should document all of the proposed control room changes.

The DCRDR study team developed design modifications on a panel-by-panel basis. Full scale prints of the modified panels were generated by computer graphics. The prints included the revised panel layouts, labels, demarcations and mimics. In order to verify that they were making the appropriate changes, the DCRDR team used eleven Watts Bar reactor operators and other Watts Bar personnel to evaluate the adequacy of the proposed modifications.

In order to determine the adequacy of the proposed modifications and schedules for implementation, the audit team evaluated all Category 1 and 2 HEDs (Attachment 4) against the NRC guidance provided in Appendix A of NUREG-0800 Standard Review Plan Section 18.1.

Based on the audit team evaluation of all Category 1 and 2 proposed modifications, along with review of a sample of Category 3 modifications and schedules for implementation, it is the audit team's judgment that TVA has met the NUREG-0737, Supplement 1 requirement for selection of design improvements.

2.7 Verification that Selected Design Improvements Will Provide the Necessary Correction

A key criterion of DCRDR success is a consistent, coherent, and effective interface between the operator and the control room. This criterion may be met by effectively executing the processes of selection of design improvements, verification that selected improvements will provide the necessary correction, and verification that the improvements will not introduce new HEDs. According to NUREG-0800, techniques for the verification process might include resurveys of panels, applied experiments, engineering analyses, environmental surveys, and operator interviews. The consistency, coherence, and effectiveness of the entire operator-control room interface are important to operator performance. Thus, evaluation of both the changed and unchanged portions of the control room is necessary during the verification process.

Based upon expertise of the individuals, DCRDR Team members were assigned responsibility for proposing corrective actions for each of the HEDs. The proposals for corrective action were presented to the whole of the DCRDR Team for evaluation against two primary criteria:

- o The corrective action should resolve the original concern

- o The correction should not result in new concerns

A formal review and approval process equivalent to the assessment and categorization methodology was employed. Corrective actions resulting in panel arrangements were mocked up in an iterative process. Full-size computer generated modified panel layouts were then evaluated by operators and human factors specialists.

Upon completion of the iterative proposal process described above, HEDs enter the formal plant engineering change procedures of preparation, review, and implementation, which includes an additional human engineering review (Human Factors Engineering - Design Review). However, there is no formal procedure for verifying that each modification, as implemented, corrects its associated HED without creating any new HEDs.

It was the audit team's judgment that the licensee did not meet the NUREG-0737, Supplement 1 requirements for verification that selected improvements will produce the necessary correction.

2.8 Verification that Selected Design Improvements Will Not Introduce New HEDs.

As discussed in Section 2.7 above, the implementation of HED corrective actions at Watts Bar go through a formal plant engineering change procedure for preparation, review, and implementation, which includes a human engineering review (Human Factors Engineering - Design Review). However, there is no formal process for verifying that the implemented modifications do not introduce new discrepancies. It was the audit team's judgment that TVA did not have a process which meets the requirement of NUREG-0737, Supplement 1, for verifying that selected design improvements do not introduce new HEDs.

2.9 Coordination of Control Room Improvements With Changes From Other Programs, such as the Safety Parameter Display System, Operator Training, Regulatory Guide 1.97 Instrumentation, and Upgraded Emergency Operating Procedures

Improvement of emergency response capability requires coordination of the DCRDR with other activities. Satisfaction of Regulatory Guide 1.97 requirements and the addition of the Safety Parameter Display System (SPDS) necessitate modifications and additions to the control room. The modifications and additions should be specifically addressed by the DCRDR. Exactly how the modifications are addressed depends on a number of factors including the relative timing of the various emergency response capability upgrades. Regardless of the means of coordination, the result should be integration of Regulatory Guide 1.97 instrumentation and SPDS equipment into a consistent, coherent, and effective control room interface with the operators.

a. The licensee made the decision to construct a new post accident monitoring system that includes SPDS. The new SPDS will receive a DCRDR type survey and additional man in the loop testing. It was the review team's judgment that the licensee coordinated SPDS with DCRDR.

b. Regulatory Guide 1.97 instrumentation requirements were coordinated with DCRDR as evidenced by the modified panel layouts being implemented as a result of the DCRDR. It was the review team's judgment that the licensee coordinated Regulatory Guide 1.97 instrumentation with DCRDR.

c. The DCRDR team identified approximately 100 procedures-related concerns that were combined into HED-006 and sent to the Emergency Operating Procedures writer staff for assessment and correction. In addition, the DCRDR task analysis was based on the draft December 1985 version of the plant specific emergency operating procedures that were derived from the Revision 1 Westinghouse Emergency Response Guidelines. It was the review team's judgment that TVA coordinated DCRDR with upgraded EOPs.

The audit team has concluded that the Watts Bar Nuclear Plant met the NUREG-0737, Supplement 1 requirement for coordination of the DCRDR with other NUREG-0737, Supplement 1 improvement programs.

3.0 CONCLUSIONS

TVA submitted the Detailed Control Room Design Review (DCRDR) Summary Report for Watts Bar Nuclear Plant, Units 1 and 2, to NRC on October 2, 1987. A preliminary evaluation of the Summary Report was conducted by SAIC which resulted in the identification of a number of concerns. In order to resolve the concerns and evaluate the Watts Bar DCRDR, a pre-implementation audit was conducted from November 14 to November 18, 1988. During the audit, the NRC staff, accompanied by SAIC and Comex representatives, performed a detailed evaluation of TVA's DCRDR. The evaluation included examination of TVA's DCRDR documentation, discussions with the DCRDR study team, inspection of the existing control room, and inspection of mockups and proposed corrective action modifications. This report reflects the consolidated findings and conclusions of the NRC audit team. The conclusions are provided below, organized by the nine NUREG-0737, Supplement 1 DCRDR requirements.

1. The establishment of the multidisciplinary review team used for the DCRDR has met the requirement of NUREG-0737, Supplement 1.

2. The system function and task analysis, which was based on Revision 1 of the Westinghouse Emergency Response Guidelines and supplements, does not meet the requirements of NUREG-0737, Supplement 1. While the audit team found that the task analysis was appropriately conducted at Watts Bar, three concerns were identified:

    1. The critical safety function trees were not analyzed.
    2. Six ECA procedures were not analyzed.
    3. The Symptoms sections of the emergency instructions were not analyzed.

    The operator information and control requirements embedded in these procedures should be analyzed using the DCRDR task analysis methodology.

3. The control room inventory does not meet the requirements of NUREG-0737, Supplement 1. While the audit team found that an adequate comparison of operator information and control requirements to the control room inventory was made for the tasks identified by the DCRDR team, it will be necessary for TVA to conduct an additional control room inventory for any new display and control requirements identified by the additional system function and task analyses performed pursuant to criterion 2.

4. The control room survey methodology and results meet the requirement of NUREG-0737, Supplement 1.

5. The licensee did not meet the requirement of NUREG-0737, Supplement 1 for an assessment of human engineering discrepancies. In order to meet this requirement, TVA should assess the significance of any new HEDs arising from the additional system function and task analysis to be conducted, and address the issues associated with HEDs 082 and 199.

6. The licensee met the NUREG-0737, Supplement 1 requirement for selection of design improvements.

7. The methodology for verifying that control room improvements correct HEDs did not meet the requirements of NUREG-0737, Supplement 1. The audit team found that a formal process had not been implemented at Watts Bar to verify that the DCRDR modifications correct the human engineering discrepancies and do not introduce new discrepancies.

8. The methodology for verifying that the control room modifications do not introduce new HEDs did not meet the requirements of NUREG-0737, Supplement 1. The audit team found that a formal process had not been implemented at Watts Bar to verify that the DCRDR modifications correct the human engineering discrepancies and do not introduce new discrepancies.

9. The coordination of the DCRDR with other programs, including upgraded EOPs, SPDS, Regulatory Guide 1.97, and training, met the requirements of NUREG-0737, Supplement 1.

4.0 REFERENCES

1. Letter from D.S. Kammer to E. Adensam, forwarding "Program Plan for Control Room Design Reviews for All TVA Nuclear Plants," June 9, 1983.

2. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability" (Generic Letter No. 82-33), December 17, 1982.

3. Letter from L.M. Mills to E. Adensam, forwarding TVA Program Plan, September 13, 1983.

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

5. NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Reviews," Draft for Comment, October 1981.

6. Memorandum for: T. Novak, NRC, From: W. Russell, NRC, Subject: Review of Tennessee Valley Authority Program Plan for Control Room Design Reviews, NRC, November 17, 1983.

7. Detailed Control Room Design Review Summary Report for the Watts Bar Nuclear Plant Units 1 and 2, Tennessee Valley Authority, October 2, 1987.

8. NUREG-0800, "Standard Review Plan," Section 18.1, "Control Room," and Appendix A, "Evaluation Criteria for Detailed Control Room Design Reviews (DCRDR)," September 1984.

ATTACHMENT 1

LIST OF MEETING ATTENDEES

MEETING ATTENDEES

NAME	ORGANIZATION
D.G. Bennett	TVA/OPS
M.C. Brickey	WBN/NE/EEB
P.I. Castleman	NRC/OSP
C.R. Cook	TVA/OPS
G.R. Davis	TVA
J. DeBor	SAIC
G.A. Elliff	SCI Services Inc.
J.J. Erpenbach	WBN
J.E. Gibbs	WB Engineering Project
R.J. Griffin	WBN-Project Management
W. Hansen	Comex Corporation
M.K. Jones	WBN Technical Support
A.E. Little	NE/Engineering Assurance
J.A. Martin	WBN-NE
G.W. Mauldin	WBN/EA
D.E. McCloud	WBN Site Licensing
J.A. McDonald	TVA-WBN
R.G. Orendi	Westinghouse
B. Paramore	Essex Corporation
B. Pedde	TVA-WBN
H.E. Price	Essex Corporation
M.E. Reeves	WBN-Project Management
M.J. [illegible]allitto	WBN/NE/EEB
M. Von Schimmelmann	NE/EA
B.S. Willis	WBN/OPS
J. Young	WBN Site Licensing
M.L. Young	WBN/NE/EEB

ATTACHMENT 2

MEETING AGENDA

AGENDA

WATTS BAR
DETAILED CONTROL ROOM DESIGN REVIEW
PREIMPLEMENTATION AUDIT

November 14-18, 1988

Monday, November 14

8:30 NRC Entrance Briefing

9:00 Licensee Overview Discussion of Watts Bar DCRDR

10:00 Tour of Control Room

11:00 Evaluation of DCRDR Review Team

1) Management and Structure
2) Composition and Qualifications
3) Team Support and Interactions
4) Orientation

12:00 Lunch

1:00 Evaluation of DCRDR System Function and Task Analysis

The team will review the system function and task analysis documentation for:

1) E-0 Reactor Trip or Safety Injection
2) E-1 Loss of Reactor or Secondary Coolant
3) E-2 Faulted Steam Generator Isolation

3:00 Comparison of Display and Control Requirements with Control Room Inventory

The team will review the control room inventory documentation for the three procedures (E-0, E-1, E-2) evaluated for task analysis. The documentation needed will include:

1) Action-Information Requirements Detail (AIRD) forms
2) Action-Information Requirements Summary (AIRS) forms
3) DCRDR Validation Forms
4) Human Engineering Discrepancy forms for resulting from validation activities.

NOTE: Part of the task analysis and inventory evaluation will be conducted in the control room.

Watts Bar Agenda Page 1

5:00 NRC Caucus

1) Summarize Findings
2) Request personnel, documentation and access needs for Day 2

5:30 End - Day 1

Tuesday, November 15

8:30 Evaluation of DCRDR Control Room Survey

1) Review team will conduct a sample survey in the control room. The purpose of the survey is to identify ten typical Human Engineering Discrepancies that should have been identified during the DCRDR survey.

2) Licensee will locate the NRC sample survey Human Engineering Discrepancies in their documentation. The purpose of this exercise is to evaluate the comprehensiveness and categorization of discrepancies.

11:00 Evaluation of DCRDR Human Engineering Discrepancy Assessment

Review team will evaluate the adequacy of the assessment activity including:

1) Identification of relative degree of degradation on operator performance.

2) Assessment of effect on plant safety.

3) Consideration of human engineering discrepancy interactions (aggregate effect).

4) Prioritization of corrective actions.

5) Justifications for leaving safety significant discrepancies uncorrected or partially corrected.

12:00 Lunch

1:00 Evaluation of DCRDR Design Improvements

The review team will evaluate the proposed and implemented design improvements. This will include a review of:

1) Hardware modifications

2) Procedure modifications

3) Training modifications

4) Schedules for modification implementations

NOTE: The review team will conduct this activity in the control room to the extent possible.

5:00 NRC Caucus

5:30 End - Day 2

Wednesday, November 16

8:30 Continue Evaluation of DCRDR Modifications

12:00 Lunch

1:00 Evaluation of Procedures:

1) Procedure to ensure that the proposed modifications correct the human engineering discrepancy.

2) Procedure to ensure that the proposed DCRDR modifications do not introduce new human engineering discrepancies.

2:00 Evaluation of the coordination of the DCRDR activity with other control room upgrade programs.

The review team will evaluate the coordination of the DCRDR program with 4 specific control room upgrade programs:

1) Safety Parameter Display System

- Coordination with DCRDR instrument range and setpoint modifications.

- DCRDR type human engineering review of displays.

2) Operator Training

- Operator training as a method to correct HEDs.

- Operator training on DCRDR modifications.

3) Regulatory Guide 1.97 Instrumentation

- DCRDR evaluation of availability of Regulatory Guide 1.97 instrumentation during EOP validation activities.

- DCRDR evaluation of suitability of Regulatory Guide 1.97 instrumentation during EOP validation.

Watts Bar Agenda Page 3

4) Upgraded Emergency Operating Procedures

   o Westinghouse Owners Group, Emergency Response Guidelines Revision 1, use as the basis for identification of operator information and control needs during DCRDR task analysis.

   o DCRDR modifications made to upgraded EOPs to correct human engineering discrepancies.

5:00 NRC Caucus

5:30 End - Day 3

Thursday, November 17

8:30 Three Sample Simulator Exercises if possible:

The purpose of the simulator exercises is to demonstrate the human engineering adequacy of the control room, with the control room team staffing at Watts Bar. The exercises should include:

1) Reactor trip or safety injection.

2) Loss of reactor or secondary coolant.

3) Faulted steam generator isolation.

10:30 Review of DCRDR-related Concerns or Allegations

12:00 Lunch

1:00 NRC Caucus Continues

2:30 Detailed Technical Exit Briefing

The purpose of the detailed exit briefing is to ensure that the NRC team findings are technically accurate. This meeting should be attended by all appropriate licensee technical staff. This meeting will include a detailed NRC evaluation of where the licensee stands with regard to:

1) Nine Supplement 1 to NUREG-0737 DCRDR requirements.

2) DCRDR-related concerns or allegations.

5:00 End - Day 4

Watts Bar Agenda Page 4

Friday, November 18

8:30 NRC Exit Briefing

The NRC exit will include a management level summary of where TVA stands with regard to the Watts Bar DCRDR. This will include:

1) Nine Supplement 1 to NUREG-0737 DCRDR requirements.

2) DCRDR-related concerns and allegations.

3) Tentative schedule for NRC Safety Evaluation Report on Watts Bar DCRDR.

Watts Bar Agenda Page 5

ATTACHMENT 3

EMERGENCY OPERATING PROCEDURES SUBJECTED TO TASK ANALYSIS

TABLE 3
EMERGENCY INSTRUCTIONS USED
FOR VALIDATION

NUMBER	TITLE
E-0	Reactor Trip or Safety Injection
ES-0.1	Reactor Trip Response
ES-0.2	SI Termination
ES-0.3	Natural Circulation Cooldown
E-1	Loss of Reactor or Secondary Coolant
ES-1.1	Post LOCA Cooldown
ES-1.2	Transfer to Containment Sump
ES-1.3	Transfer to Hot Leg Recirculation
E-2	Evaulted Steam Generator Isolation
E-3	Steam Generator Tube Rupture (SGTR)
ES-3.1	SI Termination Following SGTR
ES-3.2	Post-SGTR Cooldown Using Backfill
ES-3.3	Post-SGTR Cooldown by Ruptured S/G Depressurization
E-FOP	Foldout Page
FR-S.1	Response to Nuclear Power Generation/ATWS
FR-S.2	Response to Loss of Core Shutdown
FR-C.1	Response to Inadequate Core Cooling
FR-C.2	Response to Saturated Core Cooling
FR-H.1	Response to Loss of Secondary Heat Sink
FR-H.2	Response to Steam Generator Overpressure
FR-H.3	Response to Steam Generator High Level
FR-H.4	Response to Loss of Normal Steam Release Capabilities
FR-H.5	Response to Steam Generator Low Level
FR-P.1	Response to Pressurized Thermal Shock
FR-P.2	Response to Cold Overpressure Condition
FR-Z.1	Response to Phase B Containment Pressure
FR-Z.2	Response to Containment Flooding
FR-Z.3	Response to High Containment Radiation
FR-I.1	Response to High Pressurizer Level
FR-I.2	Response to Low Pressurizer Level
FR-I.3	Response to Voids in Reactor Vessel
ECA-0.0	Loss of All AC Power
ECA-0.1	Loss of All AC Power Recovery Without SI Required
ECA-0.2	Loss of All AC Power Recovery With SI Required

ATTACHMENT 4

SAFETY SIGNIFICANT HEDs

WATTS BAR NUCLEAR PLANT
CONTROL ROOM DESIGN REVIEW

HED	SHORT TITLE	SAFETY
CATEGORY 1		
099	Lack of Narrow Range Containment Pressure Indication In The Horseshoe.	Yes
151	Eberline System Usability.	Yes
153	Lack of Adequate Pyrotronics Alarm Power Supply For The Control Room Panels.	Yes
159	Lack Of Feedwater Isolation Reset And Status.	Yes
200	Lack of Phase B Isolation Status Lights.	Yes
202	Functional Description Not Included In Change Package.	Yes
CATEGORY 2		
008	Industrial Safety/Personnel Electrical Shock Hazard.	Yes
015	Noise Problems.	Yes
157	Panel Layout Problems On M-3/M-4.	Yes
163	M-6 Panel Layout, Emergency Core Cooling System Layout.	Yes
167	M-9 Panel Layout.	Yes
176	Pressure Indicator For Annulus Vacuum Not Located On M-27 With EGTS. Alarm Setpoint Is Such That LCO Exists Before Alarm Comes In.	Yes
CATEGORY 3		
019	Spare Parts And Supplies For The Main Control Room and Auxiliary Control Room.	Yes
043	Multiple Input Annunciators.	Yes
056	Need For Seal Water Flow Alarm/Unalarmed Seal Flow Could Exceed 40 GPM Tech Spec.	Yes
062	Shared Alarms Not Duplicated In The Unit 2 Control Room.	Yes
091	Scales/Math Conversions Required Between Controls and Indicators.	Yes

WATTS BAR NUCLEAR PLANT
CONTROL ROOM DESIGN REVIEW

HED	SHORT TITLE	SAFETY
CATEGORY 3 (continued)		
092	Lack of Main Control Room Controls And Indicators For Control and Service Air Compressors.	Yes
093	Multipoint Records RR-90-1.	Yes
107	Square Root Scale Used On Bit Flow Indicator.	Yes
119	Multipoint Records Are Hard To Read.	Yes
132	Failure Mode For Delta Flux Differential Indication Not Apparent.	Yes
160	M-4/M-5 Panel Layout Problems.	Yes
162	Lack of Status Light For Cold Overpressurization Mitigation System Arm/Block.	Yes
181	L-10 Layout Problems.	Yes
192	Auxiliary Feedwater Level Controllers Can Be Changed In Auxiliary Control Room.	Yes
193	Rod Bottom Lights Not Adequate.	Yes
CATEGORY 4		
076	CCP Burnout After Blackout After Switchover To Containment Sump After Location.	Yes
082	Accidental Changing Of Controller Setpoints.	Yes
087	Inadvertent Operation of Rad Monitor Test Switches.	Yes
103	Controllers Include Moving Scale Fixed Pinter Meters.	Yes
110	Improper Scale On Incore Thermocouple Indicator Readout.	Yes
199	Certain Valves Could Be Opened With Phase A Isolation Not Reset.	Yes

APPENDIX M

PAGES 28-38 OF INSPECTION REPORT 50-390, 391/90-05: AN UPDATE OF THE RESOLUTION OF SEVERAL GENERIC AND SPECIFIC CONCERNS

~~TVA's second interim deficiency report was submitted in a letter from R. Gridley (TVA) to USNRC dated April 21, 1988. In the second interim report, TVA committed to determine the existing strength of the various weld groups through destructive testing of weld samples taken from as-constructed weld joints. In the third interim report submitted on November 6, 1989, TVA reevaluated this interim corrective action and replaced it with a worst-case evaluation which is documented in TVA's Safety Significance Report, WCG-1-324. The inspection team stated that TVA's assumption of weld strength equivalency based on thickness replacement of a full penetration weld by a partial penetration/crown weld might not be valid. The inspection team concludes that tensile testing of the welded joint would provide conclusive evidence of the structural adequacy of the partial penetration welds. As a result, the inspection team considers the HVAC duct weld issue an unresolved item in the HVAC corrective action program (URI 90-05-01).~~

~~References~~

1. ~~Watts Bar Nuclear Plant Phase II Report, TVA Welding Project, Volume VII, Item 7.6, HVAC Ductwork Welding, page 51, April 10, 1989.~~

2. ~~Letter from E. G. Wallace (TVA) to USNRC, Subject: Watts Bar Nuclear Plant Units 1 and 2 - Safety-Related HVAC Duct Welding - WBRD-50-390/87-09 and WBRD-50-391/87-09 - Final Report, dated March 1, 1990.~~

3. ~~TVA Report WCG-1-324, "Safety Significance Evaluation for Seismic Category I HVAC Duct Welding Concerns," Rev. 1, RIMS No. B26-90-0207-101, February 7, 1990.~~

4. ~~Significant Condition Report (SCR) 7077-S, Rev. 1, Requirements for Full Penetration Welds on HVAC Duct and Duct Transitions, RIMS No. B26-87-0724-014, July 21, 1987.~~

5. ~~Significant Condition Report (SCR) WBN MEB 8714, Rev. 3, Changes to Design Input, RIMS No. B26-89-0203-201, February 3, 1989.~~

6. ~~Significant Condition Report (SCR) WBN MEB 8721, Rev. 2, Qualification Testing of HVAC Duct Welds, RIMS No. B26-89-0203-204, February 3, 1989.~~

7. ~~Significant Condition Report (SCR) WBN MEB 8722, Rev. 2, Qualification Testing of HVAC Duct Welds, RIMS No. B26-89-0207-201, February 7, 1989.~~

8. ~~"HVAC Schedule Pipe - Safety Significance Evaluation Report," EPM-JC-051688, RIMS No. B41-88-0627-800, May 16, 1988.~~

5.2 Equipment Seismic Qualification

Supplement No. 3 of the Watts Bar Safety Evaluation Report (NUREG-0847), dated January 1985, presents the status and disposition of the concerns identified by the NRC staff regarding seismic and dynamic qualification of safety-related electrical and mechanical equipment. Section 3.10 of Supplement No. 3 described the various issues. Based on information provided by the applicant (TVA) some issues were closed in the SER. However, there was a total of three generic and five specific issues that still remained open.

After the SER was issued, TVA transmitted to the NRC additional information to address the remaining open issues. The additional information was reviewed during this inspection as were other documents (reports, letters, project documents, test data, etc.). As a result of its review of these documents and discussions with TVA technical personnel, the team was able to close seven of the eight open issues. A description of each issue and the basis for its closure if applicable, is presented in Sections 5.2.1 through 5.2.8 below.

5.2.1 SER Section 3.10.1(1) - Generic Concern

Description of Issue(s)

TVA often relied upon single-frequency and single-axis testing to qualify electrical equipment. To justify this form of testing for all equipment qualified in this manner, TVA referred to Westinghouse Topical Reports (References 2 and 3) previously submitted to the NRC, NRC letter (Reference 4) and NRC memorandum (Reference 5). The staff had previously received the applicable documents and had further discussions in a meeting with Westinghouse representatives. During the meeting, TVA was requested to provide verification of analysis, including the following:

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

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

(c) The test response spectrum (TRS) envelopes the required response spectrum (RRS) for all directions.

TVA was requested to verify this for equipment at Watts Bar procured from Westinghouse. Based on the Reference 6 letter, the staff had previously concluded that the nature of the response was acceptable but the scope was not. To be acceptable, the SER Supplement No. 3 states that the responses must address the concerns for all safety-related equipment supplied by Westinghouse.

Review

To address the concerns described above, TVA letter to the NRC (Reference 6) was reviewed. It stated that Westinghouse performed a supplemental seismic test program using multiple-axis, multiple-frequency excitation. TVA believes that this testing demonstrated the seismic adequacy of the previously qualified cabinets. This letter also provided justification for not considering directional coupling for the solid-state protection system (SSPS), safeguards test cabinet, and main control board.

Additionally, TVA letter (Reference 7) was reviewed. It summarized TVA's past efforts to resolve this issue and reiterated their position. In particular, this letter referred to the NRC memorandum (Reference 5) for NRC's acceptance of Westinghouse supplemental seismic test program for resolving the multi-frequency/multi-axis concern.

TVA also referred to Westinghouse Topical Reports (References 2 and 3) and NRC letter (Reference 4) for NRC's acceptance of these reports. These documents were also reviewed by the inspection team.

Evaluation

TVA letter (Reference 6) addresses the three issues a, b, and c described above. However, as stated in SER Supplement No. 3, the responses did not address these concerns for all safety-related equipment supplied by Westinghouse, but only to certain audited items.

The additional information referred to by TVA during this inspection was the two Westinghouse topical reports (References 2 and 3), the NRC letter (Reference 4), and NRC memorandum Reference 5. Topical report WCAP 8587 (Reference 2) provides general qualification methods to be utilized for qualification of equipment to IEEE-323-1974. This includes qualification methods for environmental conditions such as aging, radiation, seismic, etc. Page 7-G of this report states that for equipment which has been previously qualified by the single-axis sine beat method..., no additional qualification testing will be required to demonstrate acceptability to IEEE 344-1975 provided that: 1) aging effects are addressed, 2) any design modifications made to the equipment does not significantly affect the seismic characteristics of the equipment, and 3) the previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra. As yet, such information/justification was not made available to the inspection team. It should be noted that the expression "test inputs can be shown to be conservative" is interpreted to include conservatism to account for potential directional coupling effects and multi-mode excitation unless otherwise justified.

The other Westinghouse topical report, WCAP 9714/9750 describes the methodology used to seismically qualify seismic Category I equipment. Page 3-6 of this report permits qualification based on previous single-frequency single-axis testing provided the same three conditions presented above for WCAP 8587 are satisfied. Section 2.0 of this report, which discusses the history of Westinghouse seismic test methods, describes the Westinghouse supplemental seismic qualification program. To review the results of this program and to verify that adequate margin exists for equipment tested by Westinghouse prior to May 1974, the NRC performed a seismic audit in 1975-76. WCAP 9714/9750 stated that "Although no generic NRC acceptance on the supplemental testing was received, it has been used to demonstrate the seismic qualification of electrical equipment for specific plants that have been licensed by the NRC." Thus, this topical report cannot be generically applied to Watts Bar.

The NRC letter (Reference 4) provides the NRC acceptance of the two topical reports. However, such acceptance is limited to the extent specified and under the limitations described in the corresponding SER. The basis for accepting the topical reports regarding seismic qualification is described in Section 3.11 of the NRC SER attached to Reference 4. That section of the SER relies upon the WCAP-9714 and the actual testing described in Section 6 of the SER. As stated in Section 3.11 (page 17) of the SER, the equipment mounting results in excitation of all three equipment orthogonal axes (simultaneously). No justification or acceptance could be found in the SER which discusses any deviation from this multi-axis testing.

In regard to NRC's memorandum (Reference 5), it did not generically accept single-frequency single-axis testing performed by Westinghouse. Therefore, the memorandum (on page 3) reached the conclusion that "additional assurance was needed by an audit... for the specified equipment employed in each specific plant application." Thus, the conclusions reached by this memorandum are applicable only to the specific Westinghouse equipment reviewed. Thus, this data can be used for qualification of some equipment at Watts Bar. However, a comparison must be made to demonstrate that the equipment are the same and that the same margins between TRS and RRS exists using the Watts Bar seismic spectra.

In summary, the two topical reports WCAP-8587 and 9714 specify that previous testing is utilized when additional conditions are met. Information satisfying these conditions has not been provided. Also, item (b) of the generic issue described above (acceleration at device locations) is independent of the three conditions required by Westinghouse and would still have to be addressed. Items (a) and (c) of the generic issue described above could be resolved by satisfying the three conditions. However, the "conservatism of the test inputs" must be sufficient to account for the directional coupling concern (item (a)). Thus, this issue remains open (IFI 90-05-02).

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. WCAP-8587, Rev. 6 (NP), "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," March 1983.

3. WCAP-9714(P)/9750(NP), "Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment," May 1980.

4. NRC letter, Cecil O. Thomas to E.P. Rake, Jr. of Westinghouse Electric Corp., "Acceptance for Referencing of Licensing Topical Reports WCAP-8587, Revision 6 (NP) ... and WCAP-9714(P)/9750 (NP) ..., dated 11/10/83.

5. NRC memo, J.P. Knight to R.C. DeYoung, "Report on Seismic Audit of Westinghouse Electrical Equipment (TAR's: 3678-1, 3683-1, 0706, 0921-1, 0788-2, 1111-2, 3000-2)," dated 8/26/76.

6. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 5/17/84.

7. TVA letter, J.W. Hufhan to Ms. Adensam of the NRC, "In the Matter of Application of TVA Docket Nos. 50-390 and 50-391," dated 2/22/85.

5.2.2 SER Section 3.10.1(3) - Generic Concern

Description of Issue(s)

In numerous cases, equipment was mounted by bolting during seismic tests whereas the equipment was installed in the field by welding at the base. This difference led to concerns regarding strength, frequency shifts, and damping. In Reference 2, TVA stated that by inspection and simple consideration of the respective metal areas involved, it is obvious that TVA's welded attachments are at least as adequate (strength and rigidity) as the bolted configurations. Instead of relying on the results of purely analytical predictions made by TVA, the staff previously had requested that TVA select a cabinet and verify its natural frequencies by in situ testing. This test which was to be performed on a Watts Bar cabinet was intended to show that the response of the cabinet is essentially unaffected by the difference in mounting. Discussion under specific issues - SER Section 3.10.2(1)(a) and 3.10.2(2)(a) relates to this generic issue.

Review

TVA letter (Reference 3) was provided to the inspection team for resolution of this issue. This letter provides the summary of the in-situ vibration test performed on the Watts Bar main control panel assembly. Because, this letter was only a summary, the staff requested and reviewed Reference 4 as well. This letter provided a complete copy of the in-situ test report. TVA also provided WCAP-8540 (Reference 5) which reported the seismic qualification by analysis and testing for the main control boards. Discussions were also held with TVA technical personnel who explained various aspects of the test and responded to questions raised by the inspection team.

Evaluation

Based upon the original testing of one Westinghouse control board section, the natural frequency was found to be 14.2 Hz (Reference 6). Based on a finite element analysis performed by Westinghouse on the M4 control board section (most critical for frequency) the natural frequency was calculated to be 14.0 Hz (Reference 5). This analysis realized that in the field, the base of the control boards will be welded to the control room floor and thus restrained the base of the finite element model in all coordinate directions. This same analysis (Reference 5) also analyzed a complete panel (horseshoe) assembly and calculated the first two natural frequencies at 14.3 Hz and 17.7 Hz. The in-situ test also performed on an entire control panel assembly at Watts Bar found the first two natural frequencies to be 14.6 Hz and 19.2 Hz (Reference 3).

On the basis of information previously provided to the NRC and the consistent results from a) the original single cabinet test, b) the analysis of the control panels, and c) the new in-situ tests, the inspection team concludes that the issue regarding mounting configurations is resolved.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. TVA letter, L.M. Mills to E. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 12/1/82.

3. TVA letter, J.A. Domer to E. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 4/30/85.

4. TVA letter, R. Gridley to B. J. Youngblood of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 1/30/86.

5. Westinghouse WCAP-8540, "Seismic Qualification of the Full Size Main Control Boards Sequoyah and Watts Bar Nuclear Power Plants," dated May 1975.

6. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant," dated 9/23/82.

5.2.3 SER Section 3.10.1(4) - Generic Concern

Description of Issue(s)

Some safety-related equipment is sensitive to aging which in turn may affect their seismic performance. To ensure that aging does not degrade the seismic capability of the equipment, TVA was requested to provide their detailed program of surveillance and maintenance for review and approval. TVA's response was provided in Reference 2. It simply referred to the response to a similar question (response to item 20 in Reference 3) from NRC on TVA's environmental qualification program. However, Supplement No. 3 of the SER (Reference 1) stated that the response did not address the equipment located in a mild environment.

Review

The inspection team discussed TVA's response to item 20 in Reference 3 with TVA technical personnel familiar with the maintenance and surveillance programs. Specific maintenance and surveillance instruction manuals were randomly selected and a cursory review performed.

Evaluation

As a result of the review of Reference 3, discussions with TVA personnel, and cursory review of the maintenance and surveillance instruction manuals, no deficiency has been identified. However, based on major revisions made to the environmental qualification program since the response in Reference 3 (November 1983) was made, the NRC will be reviewing the Watts Bar environmental qualification program separately. Thus, the inspection team finds this issue is closed from a seismic standpoint since the environmental qualification program at Watts Bar will be reviewed as part of the Watts Bar Special Program effort.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 5/17/84.

3. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 11/7/83.

5.2.4 SER Section 3.10.2(1)(a) - Specific Concern

Description of Issues(s)

For the reactor trip switchgear, TVA was requested to demonstrate that the welded field mounting is structurally as sound as the bolted mounting used in testing the unit. Supplement No. 3 of the SER stated that resolution of this issue relies on the response to item 3 in SER Section 3.10.1 and will be evaluated under that generic issue.

Review

The background of this issue was determined by reviewing Reference 2. Since resolution of this issue is evaluated under generic issue item 3 of the SER Section 3.10.1, the same references listed in Section 5.2.2 of this inspection report were also reviewed as they relate to resolution of this issue.

Evaluation

As discussed in Section 5.2.2 of this inspection report, TVA provided the basis for demonstrating strength and rigidity requirements when field mounting did not match the bolted test mounting. In addition, Section 5.2.2 discussed the in-situ test results for one NSSS Westinghouse electrical cabinet assembly which showed consistency with the original single cabinet test and subsequent finite element analysis. Based on the above discussion and the resolution of generic issue item 3 of SER Section 3.10.1, this issue is considered closed.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant, dated 9/23/82.

5.2.5 SER Section 3.10.2(2)(a) and (c) - Specific Concerns

Description of Issue(s) - 3.10.2(2)(a)

For the reactor protection system cabinet, TVA was requested to demonstrate that field mounting is as adequate as test mounting. Supplement No. 3 of the SER stated that resolution of this issue relies on the response to item 3 in SER Section 3.10.1 and will be evaluated under that generic issue.

Review

The background of this issue was determined by reviewing Reference 2. Since resolution of this issue is evaluated under generic issue item 3 of the SER Section 3.10.1, the same references listed in Section 5.2.2 of this inspection report were also reviewed as they relate to resolution of this issue.

Evaluation

As discussed in Section 5.2.2 of this inspection report, TVA provided the basis for demonstrating strength and rigidity requirements when field mounting did not match the bolted test mounting. In addition, Section 5.2.2 discussed the in-situ test results for one NSSS Westinghouse electrical cabinet assembly which showed consistency with the original single cabinet test and subsequent finite element analysis. Based on the above discussion and the resolution of generic issue item 3 of SER Section 3.10.1, this issue is considered closed.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant, dated 9/23/82.

Description of Issue(s) - 3.10.2(2)(c)

TVA was requested to evaluate the degree of amplification that occurred in the reactor protection system cabinet response motion during tests to clearly justify single-frequency testing. TVA replied by referring to their response to Item 1 in SER Section 3.10.1. TVA also indicated that the TRS envelopes the RRS. Supplement No. 3 of the SER stated that resolution of this item relies on a satisfactory response to Item 1 in Section 3.10.1 and will be evaluated under that generic issue.

Review

References 2 and 3 were reviewed to understand the background for this issue. Discussions were also held with TVA technical personnel who provided the justification for single-frequency tests to seismically qualify the cabinets.

Evaluation

TVA requested that this issue be reviewed separately from the resolution of generic issue item 1 of the SER Section 3.10.1. The inspection team agreed to evaluate the available information for this cabinet and to determine if for this specific case there is sufficient data to resolve the concern related to single-frequency testing.

TVA referred to Attachment 3.3-1 of Reference 3 where the TRS has been shown to be substantially higher than the broadened RRS throughout the frequency range. At 5 Hz for example the sine beat test had a response acceleration level of three times the required spectrum peak response. Although a specific sine beat test was not performed at 8 Hz (one of the natural frequencies), a review of Supplement No. 3 of Reference 4 points out that sine beat tests were also performed in the frequency range of 5 to 25 Hz. One test was performed at 8.5 Hz with a 0.7 g input. This test at 8.5 Hz is close to 8 Hz and considering the response of a sine beat test at 8.5 Hz with 0.7 g input, there should be sufficient energy content from this test also at 8 Hz. In addition, the shape of the RRS demonstrates that the seismic ground motion has been filtered due to one predominant structural mode. In this case a sine beat test can be an acceptable input excitation.

Considering that the building input motion is predominantly of one mode, that sine beat tests were performed throughout the frequency range, and in view of the conservatism in the test input motion as exemplified by the large margins in TRS versus RRS curves, this issue is considered closed.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant, dated 9/23/82.

3. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 12/1/82.

4. Westinghouse WCAP-7817 including Supplements 2 and 3, "Seismic Testing of Electrical and Control Equipment," dated December 1971.

5.2.6 SER Section 3.10.2(5)(a) - Specific Concern

Description of Issue(s)

The seismic analysis performed for the main control boards assumed the panel was fixed at its base. However, the panel is attached to the floor with spot welds along only the inside edge of an angle-shaped member at the base of the panel. TVA was requested to consider the potential effects due to the flexibility of this type of connection. Additional calculations were performed by TVA and discussed with the NRC. However, as stated in Section 5.2.2 of this inspection report, the NRC staff had requested that TVA perform an in-situ test of the boards to gain a measure of confidence for the minimum horizontal natural frequency of the boards. It was also believed that such a test could also close the generic concern item 3 of SER Section 3.10.1.

Review

Refer to Section 5.2.2 of this inspection report for the scope and method of review performed to resolve this issue.

Evaluation

Refer to Section 5.2.2 of this inspection report for the evaluation of this issue. In view of the consistency in the natural frequencies determined by the in-situ tests and the Westinghouse seismic analysis, this issue is closed.

References

Refer to Section 5.2.2 of this inspection report.

5.2.7 SER Section 3.10.2(7)(b) - Specific Concern

Description of Issuance

TVA was requested to provide verification that the 125-V DC vital batteries will have spacers installed, as was done during qualification tests. In Reference 3 TVA committed to install the spacers by December 1, 1984. This item will be closed on confirmation of the field modification.

Review

Reference 2 was reviewed to understand the concern regarding the battery spacers. TVA letter Reference 3 was reviewed to determine the details of TVA's commitment to install the spacers. The inspection team requested the vendor drawings to permit a comparison against the actual field installation. The inspection team inspected the installation of the spacers in the 125-V DC vital batteries and compared it to the vendor requirements.

Evaluation

The field inspection of the 125-V DC vital batteries by the inspection team demonstrated that the spacers were installed in accordance with the manufacturer's requirements and thus are consistent with the seismic test configuration. Therefore, this issue is closed.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant, dated 9/23/82.

3. TVA letter, D.S. Kammer to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 6/19/84.

5.2.8 SER Section 3.10.2(13)(b)

Description of Issue(s)

The Barksdale pressure switch was seismically qualified using single-axis, single-frequency tests. TVA was requested to provide justification for single-frequency tests. The Reference 3 letter provided some explanation for the acceptance of this type of testing. It referred to IEEE-344 which states that, "When the seismic ground motion has been filtered due to one predominant structural mode ... short duration steady-state vibration can be a conservative input excitation." Broadened floor response spectra were provided to demonstrate this. However, for the Barksdale pressure switch, at least two of the equipment natural frequencies lie at or near the RRS peak. Thus, the potential exists for multimodal excitation for which single-frequency testing may not be acceptable. The Reference 4 letter provided a comparison of the individual TRS for each sine beat test superimposed on the broadened RRS. However, as stated in SER Supplement No. 3, the fact that the individual TRS curves had enveloped the broadened

RRS does not, by itself, ensure that the unbroadened spectra will not generate multimodal response.

Review

Reference 2 was reviewed for a description of the original concern. TVA responses and data provided in References 3 and 4 were reviewed for establishing the rationale transmitted previously to the NRC. During technical discussions with TVA, unbroadened floor spectra were also provided to justify that the unbroadened RRS will not excite multimodal response.

Evaluation

As stated in Reference 2, horizontal natural frequencies were identified at 7, 12, and 33 Hz. For qualification, a check was made that the TRS of each sine beat test individually envelopes the broadened RRS at each of the test frequencies.

As shown by the unbroadened RRS, the actual motion at the floor is filtered. There is only one sharp peak centered at about 9.1 Hz. The responses at the adjacent 7 Hz and 12 Hz are substantially lower. Due to variation in properties of the building/soil, the RRS (single peak) could shift to the equipment frequencies at 7 Hz or 12 Hz. However, it could not excite both frequencies to the spectral peak. A close examination of the unbroadened RRS for Watts Bar demonstrates that since the peak is so narrow, both equipment frequencies could not be significantly excited simultaneously. In addition, a comparison of the TRS to the Watts Bar RRS (rather than to the test report RRS) shows that the equipment was tested to very significant levels far in excess of the required accelerations at Watts Bar. Thus, the single-frequency test is considered acceptable and this issue is considered closed.

References

1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, dated January 1985.

2. NRC letter, Thomas M. Novak to H.G. Parris of TVA, "Seismic and Dynamic Qualification Review of Safety Related Equipment for Unit 1 of the Watts Bar Nuclear Plant, dated 9/23/82.

3. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 12/1/82.

4. TVA letter, L.M. Mills to Ms. Adensam of the NRC, "In the Matter of the Application of TVA Docket Nos. 50-390 and 50-391," dated 5/17/84.

6.0 PERSONS CONTACTED

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*Attended exit meeting

APPENDIX N

SAFETY EVALUATION: REPLACEMENT ITEMS PROGRAM*

*Previously issued as enclosure to letter from P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991.

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

Enclosure 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE WATTS BAR REPLACEMENT ITEMS PROGRAM

TENNESSEE VALLEY AUTHORITY

WATTS BAR UNIT 1

DOCKET NO. 50-390

INTRODUCTION

In a letter dated July 31, 1990, the Tennessee Valley Authority (TVA) submitted the third revision to the Corrective Action Program (CAP) Plan for their Replacement Items Program (Piece Parts). The revised CAP reinstated the Quality Release Program in place of the Inventory Release Tracking Log, implemented additional program controls, and identified organizational changes. The tracking log was an interim measure employed to ensure that installed parts and materials were retrievable during the warehouse inventory evaluation. Additional information regarding the inventory evaluation methodology was requested by the NRR staff and subsequently provided by TVA.

EVALUATION

The Watts Bar (WBN) Replacement Items Program (RIP) as described in the CAP plan is implemented to ensure that commercial grade items intended for use in safety-related applications will not degrade the ability of the basic component to perform its safety function. The RIP addresses all 10 CFR 50.49 component replacements and commercial grade replacements for basic component future procurements, inventory items and installed items. The RIP established criteria to evaluate the adequacy of safety-related replacement parts and to dedicate commercial grade items, as needed.

The Materials Application Group (MAG),formerly Contract Engineering Group (CEG), was established to ensure that current and future replacement part procurements are appropriately qualified. The MAG is responsible for the following procurement engineering activities:

- providing technical and quality requirements in procurement documents
- identifying required testing and inspections
- performing dedication of commercial grade items
- performing technical reviews for bids received
- performing technical contract administration
- providing technical disposition for identified nonconforming items.

The warehouse inventory evaluation was performed to provide reasonable assurance of the acceptability of stored spare and replacement parts by qualifying inventory for the most critical end use. The inventory evaluation included the following:

- o performing engineering evaluations for all inventory parts (100% verification) listed in environmental qualification (EQ) binders including dedication as required.
- o performing engineering evaluations for seismically sensitive, electrically active (SSEA) QA Level II devices, and QA Level II metallic ASME pump and valve parts, including dedication, as required.
- o categorizing the remaining QA Level II inventory parts based on commodity grouping and evaluating acceptability case-by-case or by sampling.

The engineering evaluations (1) determined the required safety function of the part and, (2) verified the documentation necessary to justify acceptability of the item.

The Quality Release Program was implemented to ensure proper qualification of 10 CFR 50.49 and commercial grade parts prior to issuance from inventory. During the inventory evaluation, the Quality Release Program was temporarily suspended. The Inventory Release Tracking Log was the interim measure implemented for control inventory items. The log identified unreviewed parts and materials released for installation and allowed for their replacement if the inventory evaluation could not verify acceptability. The Quality Release Program was reinstituted upon completion of the inventory evaluation. Additionally, the program was revised to include SSEA devices and 10 CFR 50.49 replacement parts.

The RIP also reviewed previous plant maintenance activities to verify the adequacy of replacement items installed in Unit 1 safety-related components. The following maintenance records were reviewed:

- o corrective maintenance
- o preventive maintenance
- o maintenance initiated during surveillance testing
- o maintenance performed during plant modifications.

Any 10 CFR 50.49 component and SSEA device replacement parts identified to have been installed received an engineering evaluation and were dedicated as required.

Construction procurements that provided replacement parts for Unit 1 safety-related equipment were also reviewed. The review included: types and uses of procurements, procurement controlling procedures, testing subsequent to installation, component replacements related to 10 CFR 50.49 compliance, and Conditions Adverse to Quality (CAQ) reports written against construction procurements. The initial review identified specific items requiring detailed engineering evaluations.

The licensee contracted with Stone and Webster Engineering Corporation to perform the warehouse inventory evaluation, which provided the basis for identifying and qualifying the current inventory. The other major work activities that comprised the CAP were accomplished by existing licensee groups:

- o Current and future procurements - Materials Application Group (MAG)
- o Maintenance installed items - Maintenance History Group (MHG)
- o Installed construction procurements - Construction Procurement Review Group (CPRG)

CONCLUSION

Based on our review of TVA's revision of the CAP plan for the Replacement Item Program, we conclude the revision is acceptable.

Principal Contributor:

R. Wharton

Dated:

January 1991

APPENDIX O

SPDS HUMAN FACTORS CONCERNS IDENTIFIED DURING SITE AUDIT

(1) Parameter set points were inconsistent (i.e., steam generator pressure was 1224 psig on the SPDS rather than 1220 psig as in printed procedures).

The applicant committed to (a) provide consistency between SPDS parameter set points and procedures and (b) ensure that control room instruments provide indication specified in procedures. On the basis of the applicant's commitments, the staff concludes that this concern has been satisfactorily addressed.

(2) Regarding the subcriticality display, procedure references were incorrect (i.e., sending the operator to the E-0 reactor trip, or safety injection procedure instead of to the anticipated transient without scram (ATWS) procedure).

The applicant documented by letter dated November 1, 1990 (i.e., in Appendix B regarding the subcriticality display), that the error has been corrected. The staff concludes that this concern is resolved.

(3) The core cooling display appeared to have logic errors (i.e., Response to the RVLIS indication being either greater than or less than 40 percent is for the operator to go to Procedure FR-C.1).

During the conference call on February 13, 1991, the applicant clarified that (a) although the operator action is the same for both conditions, the priority of the action is different and is indicated by a difference in the color of the paths; and (b) the logic is consistent with plant procedures. The staff concludes that this concern has been satisfactorily addressed by the applicant.

(4) Sensor characteristics were not completely analyzed (containment hydrogen analyzer is off during normal operations resulting in invalid information on the containment display).

The applicant documented by letter dated November 1, 1990, that the containment display has been corrected. The staff concludes that this concern is resolved.

(5) Graphic presentation was not accurate (i.e., combining primary and secondary systems into one system on the 2CS1 plant overview display).

The applicant documented and confirmed by letter dated November 1, 1990, that this display has been deleted. The staff concludes that this concern is resolved.

(6) Eight critical safety function trees were on the SPDS, not the six described in the emergency operating instructions.

The applicant documented by letter dated November 1, 1990, that there are now six critical safety function trees on the SPDS. The staff concludes that this concern is resolved.

(7) Keyboard had layout problems (page-up/page-down keys were inappropriately placed).

The applicant documented by letter dated November 1, 1990, that the layout problems have been corrected. The staff concludes that this concern is resolved.

(8) Color contrast was poor (dark blue details were used on a black background).

The applicant committed by letter dated November 1, 1990, to use lighter colors, cyan and white, to provide greater contrast between the dark blue detail (text, piping, and poor/bad data) on the black display background. The staff concludes that this concern has been satisfactorily addressed.

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Supplement No. 6 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. 5 was issued, and (2) matters that the staff had under review when Supplement No. 5 was issued.

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