NUREG–0847
Supplement No. 8

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

January 1992

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ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), Supplement No. 6 (April 1991), and Supplement No. 7 (September 1991) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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ABBREVIATIONS

BTP	branch technical position
CAP	corrective action program
CFR	Code of Federal Regulations
CNPP	Corporate Nuclear Performance Plan (NUREG-1232, Vol. 1)
CQC	complete quadratic combination
CSB	Control Systems Branch
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EPA	Environmental Protection Agency
EQ	equipment qualification
ERCW	emergency raw cooling water
FMU	flow measurement uncertainty
FSAR	final safety analysis report
GDC	general design criterion
HVAC	heating, ventilation, and air conditioning
IE	Office of Inspection and Enforcement
ISEG	independent safety engineering group
LCSR	loop current step response
LLD	lower limit of detection
LOCA	loss-of-coolant accident
NRC	Nuclear Regulatory Commission
OBE	operating basis earthquake
PHMS	permanent hydrogen mitigation system
PORV	power-operated relief valve
PWR	pressurized-water reactor
QA	quality assurance
RCCA	rod cluster control assembly
RCS	reactor coolant system
REMP	radiological environmental monitoring program
RSA	redundant sensor algorithm
RTD	resistance temperature detector
SER	safety evaluation report
SP	special program
SRP	Standard Review Plan

ABBREVIATIONS (Continued)

SSE	safe-shutdown earthquake
SSER	supplement to SER
SSI	soil-structure interaction
TAC	technical assignment control
TVA	Tennessee Valley Authority
WBNPP	Watts Bar Nuclear Performance Plan (NUREG-1232, Vol. 4)
WRC	Welding Research Council

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), and Supplement No. 7 (September 1991).

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, respectively, which follow).

In addition to the guidance of the SRP, the staff would from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The implementation status was addressed in Section 1.14 of SSER 6. The staff is reevaluating the status of implementation of all bulletins and generic letters. Results of this reevaluation will be published in a future SSER.

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 8) 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 corrective actions that the applicant is engaged in (see Section 1.13).

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.* Appendix C updates the NRC staff's evaluation of Unresolved Safety Issue A-48 as it applies to Watts Bar. Appendix E is a list of principal contributors to this supplement. In Appendix Q, the staff's safety evaluation of September 13, 1991, is reproduced. In Appendix R, the staff's safety evaluation of June 13, 1989, is reproduced. This supplement made no changes in other appendices.

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

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1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SSERs that followed. This SSER updates the status of some of those items. The current status of each of the issues is tabulated below with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

Issue*	Status	Section
(1) 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 8)	3.10
(b) Environmental (TAC 63591)	Under review (SER)	3.11
(5) Preservice inspection program (TAC 63627)	Under review (SER)	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) Branch Technical Position CSB 6-4	Resolved (SSER 3); see License Condition 8	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5

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

Issue	Status	Section
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC 63649)	Under review (SSER 2, SSER 3)	8.2
(12) Fire protection program (TAC 63648)	Awaiting submittal (SER)	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 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC 63657)	Under review (SER)	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, R00516)	Resolved (SSER 8)	3.2.1, 3.10
(19) Seismic design concerns (TAC 79717, 80346):		
(a) Number of OBE events	Resolved (SSER 8)	3.7.3
(b) 1.2 multi-mode factor	Updated (SSER 8)	3.7.3
(c) Code usage	Resolved (SSER 8)	3.7.3
(d) Conduit damping values	Resolved (SSER 8)	3.7.3
(e) Worst case, critical case, bounding calculations	Under review (SSER 6)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Opened (SSER 6)	3.7.2.12
(h) Category 1(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Under review (SSER 6)	3.8
(k) Update FSAR per 12/18/90 letter	Resolved (SSER 8)	3.7
(20) Mechanical systems and components (TAC 79718, 80345)		
(a) Feedwater check valve slam	Under Review (SSER 6)	3.9.1
(b) New support stiffness and deflection limits	Resolved (SSER 8)	3.9.3.4
(21) Removal of RTD bypass system (TAC 63599)	Resolved (SSER 8)	5.1

Issue	Status	Section
(22) Removal of upper head injection system (TAC 77195)	Resolved (SSER 7)	6.3.1
(23) Containment isolation using closed systems (TAC 63597)	Opened (SSER 7)	6.2.4
(24) Main steam line break outside containment (TAC 63632)	Opened (SSER 7)	15.4.2
(25) Health Physics Program (TAC 63647)	Opened (SSER 7)	12.3, 12.5, 12.6, 12.7
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TAC 77550)	Under review (SSER 7)	7.5.2
(27) Containment sump screen design anomalies (TAC 77845)	Under review (SSER 7)	6.2
(28) Operating, maintenance, and emergency procedures (TAC 77861)	Opened (SSER 7)	13.5.2

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 document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

(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)	Resolved (SSER 8)	3.9.3.4
(10) Thermal performance analysis	Resolved (SSER 2)	4.2.2
(11) Cladding collapse	Resolved (SSER 2)	4.2.2
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13) Loose-parts monitoring system	Resolved (SSER 3)	4.4.5
(14) Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15) Natural circulation tests (TAC 63603, 79317, 79318)	Under review (SER)	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)	Resolved (SSER 7)	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC 63649)	Resolved (SSER 7)	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2 (TAC 63649)	Under review (SSER 3)	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC 63649)	Resolved (SSER 7)	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 (SER)	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 (SER)	8.3.3.1.1
(43) Safety parameter display system	Updated (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 current status of the proposed license conditions is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

Proposed Condition	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2
(2) Inservice testing of pumps and valves (TAC 74801)	Updated (SSER 5)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TAC 77132 and 77133)	Awaiting submittal (SER)	4.4.8
(4) Inservice Inspection Program (TAC 76881)	Awaiting submittal (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)	Resolved (SSER 8)	6.2.5, App. C
(10) Status monitoring system/BISI (TAC 77136, 77137)	Resolved (SSER 7)	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)	Resolved (SSER 7)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (II.B.3) (TAC 77543)	Updated (SSER 3, SSER 5)	9.3.2
(20) Fire protection program (TAC 63648)	Awaiting submittal (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)	Resolved (SSER 8)	13.4
(26) Use of experienced personnel during startup (TAC 63592)	Resolved (SSER 8)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC 63656)	Awaiting submittal (SER)	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)	Awaiting submittal (SER)	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC 77861)	Under review (SER)	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)	Resolved (SSER 7)	14.2
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC 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 (SSER 1)	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 waste 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) (TAC 63615)

(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 on its plans for correcting plant-specific problems. In response to this letter, TVA prepared a Corporate Nuclear Performance Plan (CNPP) that identified and proposed corrections to problems 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) <u>Cable Issues (TAC 71917)</u>

Program review status:	NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation was reproduced in SSER 7 as Appendix P); review in progress.
Implementation status:	Full implementation expected by December 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); to come.

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

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by August 1993.
NRC inspections:	Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 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 April 1993.
NRC inspections:	Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20; (September 25, 1990); 50-390/91-201 (March 22, 1991); 50-390, 391/91-20 (October 8, 1991); 50-390, 391/91-25 (December 13, 1991); to come.

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

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.
Implementation status:	Full implementation expected by October 1993.

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

(5) Electrical Issues (TAC 74502)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 8 of a future SSER.
Implementation status:	Full implementation expected by September 1993.
NRC inspections:	Inspection Report 50-390, 391/90-30 (February 25, 1991); to come.

(6) Equipment Seismic Qualification (TAC 71919)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.10.
Implementation status:	Full implementation expected by September 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); to come.

(7) Fire Protection (TAC 63648)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER.
Implementation status:	Full implementation expected by February 1993.
NRC inspections:	To come.

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

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

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); to come.

(9) Heat Code Traceability (TAC 71920)

Program review status: Complete: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991.

Implementation status: 100% (certified by letter, E. Wallace (TVA) to NRC, July 31, 1990); staff concurrence in SSER 7, Section 3.2.2.

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

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

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

Implementation status: Full implementation expected by April 1993.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); 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 was reproduced as Appendix K in SSER 6).

Implementation status: Full implementation expected by June 1993.

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

(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 September 1993.

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); 50-390, 391/90-33 (March 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 1993.

NRC inspections: Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 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 October 1992.

NRC inspections: Inspection Reports 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 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 was reproduced as Appendix N in SSER 6).

Implementation status: Full implementation expected by September 1992.

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

(16) Seismic Analysis (TAC R00514)

Program review status: Complete: Letters, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7 and October 31, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7.

Implementation status: Full implementation certified by letter, J. H. Garrity to NRC, December 2, 1991; staff concurrence to come.

NRC inspections: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); audit report by L. B. Marsh, October 10, 1990; 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 July 1992.

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

(18) Welding (TAC 72106)

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

Implementation status: Full implementation expected by August 1992.

NRC inspections: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-18 (October 8, 1991); to come.

1.13.2 Special Programs

(1) Concrete Quality (TAC 63596)

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

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

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

(2) Containment Cooling (TAC 77284)

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

Implementation status: Full implementation expected by April 1993.

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 January 1993.

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 December 1992.

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 June 1992.

NRC inspections: Inspection Report 50-390, 391/86-24 (February 12, 1987); to come.

(6) Mechanical Equipment Qualification (TAC 76974)

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

Implementation status: Full implementation expected by February 1992.

NRC inspections: To come.

(7) Microbiologically Induced Corrosion (TAC 63650)

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

Implementation status: Full implementation expected by September 1992.

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 September 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 March 1993.
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 March 1992.
NRC inspections:	Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990); audit report by L. B. Marsh (October 10, 1990); to come.

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

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by March 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); 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 in SSER 6. The interim result of such effort was shown in Sections 1.14.1 and 1.14.2 of SSER 6. Because 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, and will be reported in a future SSER.

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

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

In SSER 6, the staff identified a concern with the classification of safety-related conduits and cable trays as "seismic Category I(L)" (limited structural integrity) (Outstanding Issue 18). The staff found that this classification did not comply with Regulatory Guide (RG) 1.29, Position C.1.q which clearly states that Class 1E electrical systems are to be designated seismic Category I. In addition to the limited structural integrity classification, the applicant also proposed design criteria to qualify the Category I(L) cable trays. The adequacy of the proposed design criteria and methodology is discussed in Section 3.10 of this SSER.

During a January 29, 1991, meeting with the staff (NRC meeting summary dated February 6, 1991), the applicant stated that it would delete the seismic Category I(L) classification and would consider cable trays and conduit containing Class 1E cable as safety related or as seismic Category I. This commitment was also stated in the applicant's May 8, 1991, letter. The applicant's revised seismic classification of cable trays and conduit complies with RG 1.29, and Outstanding Issue 18 is considered closed (see also Section 3.10 of this SSER).

3.7 Seismic Design

In SSER 6, the staff stated that a number of issues were found acceptable based on the applicant's commitment (made in its December 18, 1990, letter) to revise the FSAR. Outstanding Issue 19(k) was opened to track the applicant's efforts. The staff has, since then, reviewed FSAR Amendment 68 and found that changes the applicant committed to make have been incorporated in the FSAR. This resolves Outstanding Issue 19(k).

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

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

Reactor Building

The staff issued the following evaluation by letter dated September 30, 1991:

> In response to SSER 6 which introduced Outstanding Issue 19(f), the applicant clarified the issue of mass eccentricities in evaluating the steel containment vessel for an earthquake load (TVA submittal, dated May 8, 1991). TVA stated that actual mass eccentricities from such items as equipment hatch and lock are now replaced by a 5-percent accidental eccentricity in accordance with Standard Review Plan Section 3.7.2 (Rev. 2, August 1989). Conservatism of the new eccentricity

was demonstrated by comparing torques due to two different types of eccentricities, and the applicant showed that the 5-percent accidental eccentricity is more conservative. The applicant also proposed a revision to the FSAR to document this change.

The staff finds the applicant's response to the issue of mass eccentricity, including proposed changes to the Final Safety Analysis Report (FSAR), acceptable. On this basis, Outstanding Issue 19(f) is considered resolved.

3.7.3 Seismic Subsystem Analysis

In SSER 6, the staff identified an issue regarding the number of earthquake stress cycles considered in the design of seismic subsystems (Outstanding Issue 19(a)). The Watts Bar FSAR was revised in Amendment 64 to specify that the number of equivalent peak stress cycles considered for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE) are 20 cycles and 10 cycles, respectively. The numbers of equivalent stress cycles proposed by the applicant are based on the postulated occurrence of two OBEs and one SSE during the design life of the plant (40 years). The applicant proposed to consider 10 cycles of maximum stress for each event based on the Standard Review Plan (SRP) (NUREG-0800) and IEEE Standard 344-1975.

The use of 10 peak stress cycles for the SSE is consistent with SRP Sections 3.7.3 and 3.9.2 and with IEEE 344-1975 requirements and is, thus, acceptable to the staff. However, the applicant's proposal did not consider a total of 50 peak stress cycles for the OBE which should be considered based on the guidelines of the SRP and IEEE 344-1975 (5 OBEs × 10 peak stress cycles per event).

In a letter dated May 8, 1991, the applicant proposed to revise the FSAR for ASME Section III Class I piping analyses. The proposed revision stated that since the piping in this scope has been reanalyzed in accordance with SRP requirements, the reanalysis will assume the occurrence of five OBEs and one SSE. The number of peak stress cycles per event will be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the analysis, or a minimum of 10 peak stress cycles per event will be assumed. Since this criterion is in accordance with SRP Section 3.9.2, this proposed revision for piping analysis is acceptable.

In the May 8, 1991, letter, the applicant also stated that for Class A Category I components, the most significant response of the components is conservatively considered using an average frequency of 20 Hz. Therefore, the number of cycles considered for these components for the OBE and SSE are 600 and 300, respectively. Since this paragraph is a reinstatement of the criteria in FSAR Amendment 51, which was previously reviewed and approved, this proposed revision is acceptable.

In the May 8, 1991, letter, the applicant further proposed to add a paragraph to the FSAR stating that seismic qualification testing of Category I equipment considers the number of events and durations described above, in accordance with IEEE Standard 344-1975. Amendment 64 of the Watts Bar FSAR as well as Amendment 51 stated that "during the design life of the plant, two earthquakes of OBE magnitude and one of SSE magnitude are postulated to occur." Since this was previously reviewed and accepted as a suitable basis for testing, the proposed additional paragraph regarding seismic testing of Category I equipment is acceptable.

On the basis of this discussion, the applicant's proposed FSAR revisions regarding the number of earthquake cycles as stated in its May 8, 1991, letter are acceptable, and Outstanding Issue 19(a) is considered resolved.

As discussed in SSER 6, the staff identified a concern regarding the proposed use of a multimode factor of 1.2 in the seismic analysis of cable tray, conduit, and heating, ventilation, and air conditioning (HVAC) systems at the Watts Bar facility (Outstanding Issue 19(b)). The proposed factor is less conservative than the multimode factor of 1.5 recommended in the SRP and, consequently, the staff reviewed the basis for its use. In the review of the calculations initially provided by the applicant to support the proposed factor, the staff raised questions regarding: (1) the use of the complete quadratic combination (CQC) method for combining modal responses and (2) an uncertainty regarding how representative the models employed in the applicant's study are when compared to actual plant configurations.

In response to the staff's concerns, the applicant performed additional studies which were submitted for the staff's review in a letter dated June 13, 1991. The results of these studies were incorporated into the Sargent and Lundy calculation WCG-1-397 entitled, "Two Degree of Freedom Comparisons to a Coupled System Response." The latter calculation was revised to TVA Rev. 1/S&L Rev. 3 to reflect the additional studies. These studies include additional configurations and parametric variations. Specifically, the applicant considered an additional 14 systems on flexible supports as well as 4 systems on rigid supports, with the new systems including elbow and tee fittings.

The additional models and the variation of parameter studies included in the revised calculation substantially improved the basis for justifying the proposed factor. However, two aspects of the study were identified as requiring resolution before the staff could complete its evaluation. First, in comparing results between the dynamic analyses developed with GTSTRUDL and those developed in the corresponding approximate 2DOF calculations, it appears that the estimates of the fundamental frequencies of the systems considered by the applicant agree reasonably well. However, comparable frequencies do not necessarily imply that the modes of interest are the same from both models. In a followup discussion, the applicant agreed to perform mode shape comparisons to ensure that the fundamental modes correspond equally as well. The second aspect of the applicant's study requiring resolution is related to the use of the CQC method. As discussed in SSER 6, the latter method yields varying results as compared to the methods recommended in RG 1.92. During the followup discussion, the applicant also agreed to carry out comparative studies with the objective of resolving this staff concern. Such studies would compare the results developed using the CQC method to results developed using methods deemed acceptable in the SRP or appropriate RGs.

The applicant responded to the staff's concerns in a letter dated October 10, 1991. The staff is reviewing this response. Pending completion of this review, Outstanding Issue 19(b) remains open.

In Section 3.7.3.8.1B of FSAR Amendment 64, the applicant listed specific ASME code cases it proposes to use in the design of piping systems. In SSER 6, 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 to revise its FSAR to include the specific revisions utilized. The staff designated the evaluation of the acceptability of the proposed code cases as Outstanding Issue 19(c).

The applicant's letter dated July 31, 1991, listed the following specific revisions to ASME code cases it proposes to use in the design of piping systems:

- Code Case N-122, January 21, 1988, Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1.

- Code Case N-313, November 28, 1986, Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1, Class 2.

- Code Case N-318-3, September 5, 1985, Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.

- Code Case N-319, July 13, 1984, Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping.

- Code Case N-391, November 28, 1983, Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.

- Code Case N-392, November 28, 1983, Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and Class 3 Piping, Section III, Division 1.

- Code Case N-411-1, February 20, 1986, Alternate Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Systems, Section III, Division 1, may be used.

- Code Case 1606-1, December 16, 1974, Stress Criteria, Section III, Classes 2 and 3 Piping Subject to Upset, Emergency, and Faulted Operating Conditions.

The staff evaluated Code Case N-411 in SSER 6. Code Case 1606-1 is contained in Position C.2 of RG 1.84 which lists code cases that were endorsed by the NRC in an earlier version of RG 1.84 and were later annulled. However, Code Case 1606-1, which contains stress criteria for Code Class 2 and 3 piping subject to upset, emergency, and faulted operating conditions, was subsequently incorporated in the ASME Code. Since these allowable stress values were subsequently incorporated in the code, the staff considers that the use of these allowable values provides an acceptable basis for the design of ASME Code Class 2 and 3 piping. The remaining code cases have been endorsed in Position C.1 of Regulatory Guide 1.84 as acceptable to the staff within the limitations given with the individual code case in the listing. On the basis of this discussion, Outstanding Issue 19(c) is considered resolved.

In SSER 6, the staff identified a concern with the specified damping values for conduit systems of 4 percent and 7 percent for OBE and SSE, respectively (Outstanding Issue 19(d)). The staff concluded that there was insufficient basis for the applicant's use of averaged damping test results to establish these values, particularly since the scatter of the test data ranged from 3 percent to 22 percent. An additional concern was whether the applicant's test data sufficiently covered all the variations in configurations and design parameters for conduit systems such as cable fill, span lengths, diameters, and support conditions.

The applicant sent the staff clarifying information in a letter dated December 18, 1990. In this letter, the applicant presented a rationale for the validity of its damping values for OBE and SSE. The applicant argued that the proposed OBE value was lower than the mean minus one standard deviation of all test data and that the proposed SSE value was lower than the mean of all test data. The letter also described the parametric nature of the test in encompassing the variations in conduit configuration and design. In addition, the letter described the precedence for 4-percent and 7-percent damping values for OBE and SSE at other nuclear power facilities. The staff reviewed this submittal and determined that the applicant had not presented a sufficient justification for using average values for the SSE.

In a letter dated May 8, 1991, the applicant provided supplemental justification for the proposed damping values. In this submittal, the applicant explained why the conduit system at Watts Bar should be considered as bolted steel structures and thus allow the use of 4-percent and 7-percent damping in accordance with RG 1.61. While the analogies to bolted steel structures apply in certain conduit configurations (e.g., use of threaded fitting conduit connections), they do not apply in all cases. The May 8, 1991, letter also made comparisons (e.g., conduit configuration, support types, ground acceleration) with other licensed nuclear plants at which 4-percent and 7-percent damping values were used. A review by the staff of some of the FSARs for these plants confirmed the applicant's contention concerning the conduit damping values used at other nuclear facilities. However, from the review of the licensing documentation, the staff could not confirm the basis for accepting these damping values which could have been on the assumption that the supports were actually constructed of bolted steel structures. By contrast, the supports at Watts Bar consist primarily of welded tube sections.

In a letter dated August 22, 1991, the applicant summarized its rationale for justifying the use of 4-percent and 7-percent damping. This submittal reiterated many of the reasons presented earlier. However, to provide additional technical support for its position, the applicant used an approach endorsed by the Welding Research Council (WRC) in Bulletin 300 to determine damping in piping systems. The WRC technical position allows system damping values to be determined experimentally. One of the two experimental options is to perform a test on a similar system and define the damping value as two-thirds of the mean value of damping from the test.

In the August 22, 1991, letter, the applicant applied this experimental option to the laboratory test data. In doing this, the applicant first performed a least squares fit to all the steel conduit damping data as a function of strain. From this curve, mean damping values corresponding to OBE and SSE strain levels were determined. This statistical evaluation showed a trend of increasing damping at higher strains. The allowable damping values for analysis purposes was calculated as two-thirds of the mean values determined at the OBE/SSE strain levels. The calculation resulted in estimates of the damping values of 6.3 percent for OBE and 7.3 percent for SSE.

On the basis of the applicant's statistical evaluation of the experimental test data which demonstrated that the proposed damping values are more conservative than mean values and with consideration of the other rationale presented in the correspondence described above, the use of 4-percent and 7-percent conduit damping for OBE and SSE, respectively, for the set B and set B + C analysis, is acceptable. Outstanding Issue 19(d) is considered resolved.

3.9 Mechanical Systems and Components

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

In SSER 6, 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 (Outstanding Issue 19(h)). 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 C.2 of Regulatory Guide 1.29. Position C.2 of Regulatory Guide 1.29 states that these systems should be designed and constructed so that the SSE would not cause a failure that would affect the functioning of a safety-related system.

The applicant, in its December 18, 1990, letter, stated that it was performing a verification program to validate the original seismic design basis for Category I(L) piping. The applicant proposed to use screening criteria based on earthquake experience data to identify items requiring further evaluation. In addition, the applicant proposed to perform bounding case stress analyses to demonstrate the conservatism of the screening criteria. In its September 20, 1991, submittal, the applicant provided revised criteria for the bounding case evaluation of piping and supports.

These revised criteria require pressure boundary retention piping to meet the same stress limit for the SSE load case that is applicable to ASME Class 2 and 3 piping. On the basis that the applicant's analysis of the bounding cases, using the same seismic input for the SSE applicable to Category I systems, confirms the adequacy of the screening criteria, the staff considers that the proposed program is adequate for the verification of the existing Category I(L) piping. The applicant also proposed criteria for the evaluation of Category I(L) piping supports. The proposed criteria include the use of a factor of safety of three for concrete expansion anchor bolts. This factor of safety is less than the factor of safety of four or five, depending on anchor type, specified in IE Bulletin 79-02 for Category I pipe supports. However, IE Bulletin 79-02, which contains the staff's position on factors of safety for concrete expansion anchors, applies only to Category I systems. The factor of safety of three has been proposed for the evaluation of some equipment anchorages in the resolution of Unresolved Safety Issue (USI) A-46. The use of the lower factor of safety in the resolution of USI A-46 is compensated, in part, by an additional proposed requirement to inspect the concrete in the vicinity of the anchor for surface cracking. The criteria require that the factor of safety be increased if surface cracks greater than 10 mils are identified. In addition, the proposed criteria for the resolution of USI A-46 contain recommended values for anchor spacing and concrete edge distance. These criteria require that the anchor bolt capacity be reduced if these recommendations are not met. The staff considers the applicant's proposed use of the factor of safety of three for the validation of the existing design of concrete expansion anchors used in Category I(L) piping systems, when used in conjunction with the recommendations discussed above for concrete inspection and spacing and edge distance, to provide an adequate margin of safety for the existing designs and is acceptable. The staff further recommends that the applicant use the required factor of safety from IE Bulletin 79-02 for any future Category I(L) pipe support design until such time as the staff revises its guidance.

On the basis of the considerations discussed above, Outstanding Issue 19(h) is considered resolved.

3.9.3.4 Component Supports

In SSER 6, the staff identified a concern that new stiffness and deflection limits for seismic Category I piping supports had been proposed in Amendment 64 (Outstanding Issue 20(b)). The new criteria contained a check of deflection limits whereas the original criteria checked either deflection or support natural frequency. In addition, the new criteria specified a slightly larger allowable maximum deflection to be used with the total design load, whereas the original criteria specified a smaller deflection to be used with the normalized load.

The applicant had changed the pipe support design criteria to address an issue identified in the employee concerns program. This change in the criteria, in response to the issue raised by the employee concern, contained an allowable deflection limit with the addition of a minimum design load based on the weight of a standard span of piping. The applicant subsequently revised these criteria and eliminated the minimum design load. The revised criteria which eliminated the minimum design load were eventually incorporated into Amendment 64. The staff requested that the applicant address the discrepancy between the employee concerns program resolution of the issue and the current criteria incorporated in Amendment 64.

In response to the staff's concern, the applicant initiated a three-phased program to demonstrate that the later changes neither impact the closure of the employee concern nor compromise the adequacy of the pipe supports at the Watts Bar Nuclear Plant. The purpose of the applicant's program was to demonstrate compliance with the minimum design load requirements.

The applicant's letter report of April 18, 1991, covered the first two phases of the action plan. The third phase results were reported in the applicant's letter report of July 31, 1991, which was subsequently revised and updated in the September 30, 1991, letter report.

On the basis of its review and evaluation, the staff finds that the applicant's three-phased program adequately addressed the employee concerns program resolution. In addition, the staff considers the new criteria, with the addition of a minimum design load requirement, an improvement in the original deflection criteria used to design the supports. The staff further concludes that the new criteria provide an adequate basis for demonstrating the structural capacity of the supports for their intended service. The staff concludes that Outstanding Issue 20(b) is resolved.

In SSER 6, the staff updated the status of Confirmatory Issue 9, which addressed the adequacy of the applicant's evaluation of the effects of baseplate flexibility to meet the requirements of IE Bulletin 79-02. The applicant's previous submittals on the subject had been superseded as a result of the corrective action programs being implemented at Watts Bar. The applicant's revised submittal of January 31, 1991, stated that the design methodologies have changed for the analysis of flexible and rigid baseplate, anchor stiffness, and prying.

The applicant, in a letter to the staff on July 26, 1991, described the updates to its previous response to IE Bulletin 79-02 and its civil design standard for concrete anchorage. The civil design standard update incorporated an increase in anchor stiffness and the consideration of prying forces for thin baseplates which are analyzed by hand methods.

In the updated civil design standard, a conservative expansion anchor stiffness value of 400 kips per inch based on upper bound values from static tests is specified for the analysis of flexible baseplates.

On the basis of its review of the applicant's responses (discussed above), the staff finds that the applicant has adequately addressed the issue of pipe support baseplate flexibility and its effects on anchor bolt loads. Confirmatory Issue 9 is considered resolved.

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

In SSER 6, the staff identified an issue concerning the seismic qualification program of Category I electrical and mechanical equipment (Outstanding issue 4(a)). Watts Bar FSAR Amendment 64 specifies that the seismic qualification program is based on the guidelines of IEEE Standard 344-1971 and IEEE Standard 344-1975, depending on the date of procurement of the particular equipment. Before the amendment, the applicant committed to IEEE Standard 344-1971 and SRP Section 3.10. Although the SRP does not require adherence to IEEE 344-1975 for plants with construction permit applications docketed before October 27, 1972, it does specify certain guidelines that should be satisfied. Since reference to SRP Section 3.10 was deleted, it was not clear that the SRP guidelines would be followed.

In its May 8, 1991, letter, the applicant proposed to revise the Watts Bar FSAR Section 3.10 to reinstate its original criteria and to refer to Section 3.7.3.16 of the FSAR for a further description. Since the proposed revision would reinstate the commitment to meet IEEE Standard 344-1971 and SRP Section 3.10, this change would have been acceptable. However, a review of SRP Section 3.7.3.16, which was referred to by Section 3.10 of the FSAR, revealed commitment only to certain guidelines of the SRP. This was not considered acceptable because particularly important guidelines, such as justification of single-frequency and single-axis testing, were not included.

In an August 22, 1991, letter, the applicant revised its proposed revision to FSAR Section 3.7.3.16 to indicate that the alternate qualification method is to follow the requirements of IEEE Standard 344-1971 and, in addition, to address the guidelines of SRP Standard 3.10. Since the guidance of SRP Section 3.10 is referred to with no exclusions, this proposed FSAR revision is acceptable.

On the basis of this discussion, the proposed revisions to Watts Bar FSAR Sections 3.7.3.16 and 3.10 are acceptable. Outstanding Issue 4(a) will be considered fully resolved when item 3.10.1(1), as described in SSER 6, is resolved.

It should be understood though that this acceptance presumes that for equipment qualified to IEEE Standard 344-1971 and SRP Section 3.10, documentation/ justification is developed, either on a case-by-case basis or generically, to demonstrate that the guidelines of the SRP have been addressed.

In SSER 6, the staff identified an open item regarding the applicant's design criteria and methods for the seismic qualification of conduit and cable trays (Outstanding Issue 18). For cable trays, the applicant had proposed criteria based on a consideration of nonlinear response behavior to show seismic compliance for systems categorized as having limited structural integrity (Category I(L)). In its May 8, 1991, response, the applicant proposed to use the same analysis methods that are applicable to seismic Category I subsystems for the analysis of conduit. This proposal is acceptable to the staff.

In a January 29, 1991, meeting the staff requested additional information to address concerns related to the use of inelastic spectrum techniques in the qualification of cable tray systems (NRC Meeting Summary dated February 6, 1991). The applicant responded in a letter dated May 8, 1991. After reviewing the response, the staff still had concerns with the use of the inelastic spectrum. In a subsequent discussion with the applicant, the staff requested additional information regarding the test data used to derive the allowance moments, and the application of this data in the cable tray evaluation.

The applicant provided supplemental information in a letter dated August 22, 1991. Following review of this letter, the staff concluded that the issue could not be resolved on the basis of this information. As a result, an additional discussion was held with the applicant in which the staff reiterated its concern with regard to the applicant's proposed criteria and methods, and with the applicant's justification for the use of a ductility factor of three in the inelastic response design procedure.

In a letter dated September 18, 1991, the applicant advised the staff it was revising the cable tray qualification criteria. Specifically, general cable tray qualification will be carried out using conventional linear elastic analysis methods. Qualification considering nonlinear response behavior would be employed, when appropriate, only on a case-by-case basis. In those instances, the case-by-case application of inelastic response spectra techniques would be submitted to the staff for approval. The staff finds that the latter approach in the design criteria of cable trays regarding the use of inelastic response spectra is consistent with the guidelines of SRP Section 3.7.2 (1989 revision) and is, therefore, acceptable. This finding coupled with the finding regarding classification (see Section 3.2.1 in this SSER) resolve Outstanding Issue 18.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.4 Reactor Coolant System Temperature Measurement*

By letter dated December 1, 1986, the applicant indicated that it would modify temperature measurement for the hot legs of the reactor coolant system. (The applicant had previously discussed its proposed modification with NRC in a meeting on October 14, 1986. See TVA letter dated November 3, 1986.) This modification is being made to eliminate the resistance temperature detector (RTD) bypass manifold, reducing radiation exposure, improving availability, and reducing maintenance. However, the new method for measuring hot-leg temperature has the disadvantage of a slightly longer response time.

The staff has reviewed pertinent pages in Chapters 4 and 15 of the FSAR, as revised up to Amendment 63. Earlier, the staff reviewed the instrumentation aspects of this modification and published its findings in a letter to TVA, dated June 13, 1989. That evaluation is reproduced in this supplement as Appendix R.

The new method of measuring hot-leg temperatures uses narrow-range RTDs in thermowells. These are located in each hot leg at three locations, 120 degrees apart, where there were formerly sampling scoops. The new method, with a thermowell RTD, measures the temperature at one point. This is in contrast to the five sample holes used for scoop measurement at the same location to obtain a representative sample of hot-leg fluid for temperature measurement. The RTD for the new method is placed at the center hole location of the five holes in the former scoop. The applicant has analyzed that this location measures the equivalent temperature of the average scoop sample.

A microprocessor-based system is used to determine the average of the reactor coolant hot-leg signals from the three RTDs in each hot leg and then transmit the signal for the average hot-leg temperature to protection and control systems. This system is the Eagle-21 system and is discussed in Appendix R of this supplement.

The RCS delta temperature (delta-T) and RCS average temperature (T_{avg}) is computed from the narrow-range hot- and cold-leg thermowell-mounted RTD inputs by the Eagle-21 electronic protection system. The main control board has alarms for deviation of T_{avg} and delta-T. If any single-loop T_{avg} deviates from the auctioneered high T_{avg}, an alarm indicates the deviation. Also, the deviation of any single-loop delta-T from the auctioneered high delta-T is indicated on

*Section 4.4.3.4 is a new section, introduced in this supplement.

this control board. These alarms actuate at a deviation of 2° (on the Fahrenheit scale).

The Eagle-21 system employs an algorithm that automatically detects a defective hot-leg RTD input signal and eliminates that input from the calculation of "T_{hot} average." This is accomplished by incorporating a redundant sensor algorithm (RSA) into the hot-leg temperature signal processing. The RSA determines the validity of each input signal and automatically rejects a defective input. The typical tolerance bandwidth for automatic rejection is from 2° to 6°(F). The exact value will be determined during startup based on actual measurements of hot-leg temperature. Because of hot-leg streaming, the temperature varies in the cross-section of the hot legs. The RTDs at the three locations in each hot leg are processed to get a "T_{hot} average" temperature. The Eagle-21 system can add a bias to the averaging calculation, if needed, in order to compensate for the loss of one of the three RTD sensor inputs. The bias considers the past history of the previous hot-leg readings. The input bias that is used to compensate "T_{hot} average" upon loss of one narrow-range T_{hot} signal is based on "T_{hot} average" with three valid RTD inputs. There is one bias value associated with each narrow-range T_{hot} RTD input signal. The bias value for each RTD is calculated while all three RTDs are considered to be valid by subtracting the average of the remaining two RTDs from the "T_{hot} average" value for the loop. Then, if an RTD should fail, "T_{hot} average" for that loop is calculated by adding the bias value for the failed RTD to the average of the remaining two RTDs. If a single RTD does fail, the value of "T_{hot} average" would be calculated as described above and an "RTD failure" alarm and status light indicating "trouble" would be activated in the control room. The failed RTD would be replaced during a subsequent plant outage. If two or three hot-leg RTDs in the same loop fail, a dedicated alarm and annunciator would be activated indicating a failed channel. The Watts Bar Technical Specifications or similar documents will detail the action that must be taken for a failed overtemperature delta-t/overpower delta-T channel.

The applicant stated that the measurement of the cold-leg temperature (T_{cold}) has been modified with a single thermowell RTD and spare in each cold leg downstream of the reactor coolant pump in place of an external reading in the cold-leg manifold. In the Eagle-21 system there is an RTD failure alarm and annunciator window to indicate when there is an invalid T_{cold} for a loop. Cold temperature streaming is not a problem due to the mixing action of the reactor coolant pump. Therefore, a single sensor for each cold-leg loop is sufficient to obtain the average cold-leg temperature.

By letter dated January 27, 1987, the applicant provided FSAR changes regarding the new modifications required in reactor coolant system (RCS) temperature measurement because of the elimination of the RTD bypass loop. Included were the results of the reanalysis of several FSAR Chapter 15 accidents that were not loss-of-coolant accidents. The staff questioned the applicant regarding the accuracy and response time effects on the new temperature measurement system. The accuracy of the hot-leg temperature affects the accident analysis. Also, it is a principal contributor in the analysis for calculating the RCS flow measurement uncertainty. The longer response time of the new RTD affects the results of the accident analysis. The applicant responded to staff questions in a letter dated March 17, 1987. Regarding accuracy, the applicant stated that the total temperature streaming uncertainty for the new design with thermowell RTD measurements is smaller than that of the RTD bypass system. This is mainly due to the increase in the number of RTD measurements for each loop from the former one to the current three. This results in a statistically lower error. This improved result is obtained even though the individual temperature measurement uncertainty for the new RdF RTDs is slightly increased from that of the former Rosemount RTDs. Furthermore, the applicant has increased the conservatism by adding several degrees of temperature uncertainty to the FSAR Chapter 15 analyses conducted for Watts Bar. This is to ensure that the temperature uncertainty for the RdF RTDs is bounded and does not impact the overall system accuracy or the safety analyses. In a letter dated July 9, 1991, the applicant stated that the RTDs will be recalibrated after each refueling, using the Westinghouse cross-calibration method.

Although the thermal lag time from the bypass piping is eliminated in the new RTD design, there is an increase in response time because of insertion of each RTD in a thermowell rather than by direct immersion. The increased response time results in longer delays for the time when fluid conditions in the RCS require overtemperature delta-T or overpower delta-T reactor trips until a trip is actually generated. In a letter dated July 9, 1991, the applicant stated that the accident analysis for overtemperature delta-T and overpower delta-T assumes a 7.0-second delay. The overall response time (RTD response time plus electronics delay) for the new RdF RTDs is 0.5 second longer (6.5 vs. 6.0 seconds) than the former Rosemount RTDs. This leaves a margin of 0.5 second (7.0-6.5) between the analysis and overall RTD response time. The breakdown of components used to arrive at the overall response time is 5.5 seconds for the RTD/thermowell and a conservative electronics delay of 1.0 second. The applicant stated that it will use the loop current step response (LCSR) test to measurement RTD response time. A 10-percent allowance for LCSR test uncertainty will be used to ensure an overall channel response time of 7.0 seconds or less.

The applicant presented information concerning the FSAR Chapter 15 non-LOCA accidents that relay on the above-mentioned trips and the affect of the longer response time. The non-LOCA accidents affected by the longer response time include (1) the uncontrolled rod cluster control assembly (RCCA) withdrawal, (2) the loss of load/turbine trip, and (3) the accidental depressurization of the RCS. The accidents are described in FSAR Section 15.2. The applicant stated that the LOFTRAN code was used and the results showed that the departure from nucleate boiling ratio (DNBR) was met in all three accidents.

Regarding LOCA analysis, the applicant stated in a letter dated July 9, 1991, that th RCS primary-side conditions that are used in LOCA analysis models are unaffected by elimination of the RTD bypass system. The change in RCS volume due to the elimination of the RTD manifold piping is insignificant and does not affect LOCA analysis input. The RCS primary-side and steam generator secondary-side temperatures used in LOCA analyses are determined on the basis of the anticipated, best-estimate loop average full-power operating temperature (T_{avg}) without uncertainty. Since the Watts Bar best-estimate T_{avg} value (together with thermal design flow) is unaffected by RTD bypass removal, the RCS operating condition values used for LOCA analysis input are unaffected. Considering these, the applicant concluded, and the staff agrees, that the elimination of the RTD bypass piping will not affect the LOCA analysis input. Therefore, the results of these analyses for Watts Bar are unchanged and no LOCA reanalysis is required.

A flow measurement analysis was previously submitted for Watts Bar showing a flow measurement uncertainty of ± 1.8 percent, including a venturi fouling penalty of ± 0.1 percent. This was reviewed by the applicant and found to bound the changes due to the RTD bypass system removal and is, therefore, acceptable. In the letter of July 9, 1991, the applicant stated that it now plans to install the second phase of the Eagle-21 process protection system before fuel load. This may affect the flow measurement analysis. Therefore, that analysis will be revalidated, and will be tracked by TAC 81063. The revalidation will be documented in a revision to the setpoint methodology document.

In conclusion, the impact of the RTD bypass elimination for Watts Bar on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated and is acceptable. For the events affected by the increase in the channel response time, the applicant has demonstrated that the conclusions presented in the FSAR remain valid and the DNBR limit value is met. The applicant presented an analysis to support an RCS flow measurement uncertainty value, which includes the new hot-leg RTD temperature accuracy. The staff's acceptance is based on commitments made by the applicant in its letter of July 9, 1991, as follows:

(1) During initial startup testing, actions will be taken to correct any resistance temperature detector (RTD) channel with an overall response time of greater than 7.0 seconds including electronics delay and a 10-percent allowance for loop current step response test uncertainty. After any such corrective action, the channel will be retested to verify an overall response time of 7.0 seconds or less (the value assumed in pertinent safety analyses).

(2) Subsequent to plant startup, a reanalysis of non-LOCA transients will be performed to model Watts Bar's overall as-built performance more accurately and to establish a more realistic safety analysis value for overall RTD channel response time.

(3) The flow measurement uncertainty (FMU) will be revalidated as part of the reactor protection system evaluation which is included in the remaining scope of the Eagle-21 upgrade (tracked by TAC 81063). The results of FMU revalidation will be documented in a revision to the Westinghouse setpoint methodology document (WCAP-12096).

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.5 Combustible Gas Control Systems

In Appendix C to the Safety Evaluation Report (SER), the staff concluded that the Watts Bar facility could be operated before complete resolution of Unresolved Safety Issue (USI) A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." This conclusion was based on interim measures related to hydrogen control for degraded-core conditions. These interim measures were issued as a proposed rule on December 23, 1981, in the Federal Register (46 FR 62281). This proposed rule required that all reactors with Mark III or ice-condenser containment types (such as Watts Bar) install hydrogen-control systems capable of accommodating an amount of hydrogen equivalent to that generated from a 75-percent metal-water reaction of the active fuel cladding without loss of containment structural integrity. The final rule was published in the Federal Register on January 25, 1985 (50 FR 3498), and became effective on February 25, 1985. The requirements in the final rule are basically the same as in the proposed rule. In resolving USI A-48, the staff concluded that current regulatory requirements (10 CFR 50.44 as amended by the final rule) are adequate for hydrogen control for recoverable degraded-core accidents (like the one that occurred at Three Mile Island (TMI) Unit 2) and that no new regulatory guidance was necessary. However, it should be noted that the resolution of USI A-48 does not mean that all research or regulatory actions with regard to hydrogen are complete. Ongoing concerns related to hydrogen are in the areas of very-low-probability severe accidents, and are being pursued in accordance with the Commission's Severe Accident Policy issued on August 8, 1985 (50 FR 32138).

To meet the requirements of the final rule, the appliant has installed a hydrogen-ignition system very similar to the igniter system at Sequoyah (lead plant identified in the final rule), and since the Watts Bar plant itself is quite similar to Sequoyah, the previous demonstration of adequacy of the Sequoyah igniter system is sufficient to justify the use of igniters at Watts Bar. Therefore, the applicant proposed that the Watts Bar plant-specific analysis of degraded-core accidents was unnecessary. The design of the ignition system at Watts Bar was described in Amendment 55 to the applicant's FSAR, Section 6.2.5. The applicant sent the staff supplemental information by letters dated May 10, 1983, May 25 and November 27, 1984, and February 14, March 18, April 5, April 17, and June 25, 1985. The staff reviewed this information and noted two differences between the Watts Bar and Sequoyah plants that might affect the conclusion of adequacy of the Watts Bar igniter design. These two differences are:

(1) The design flow rate of the spray system is lower (4000 gpm vs. 4750 pgm) at Watts Bar.

(2) Watts Bar is not equipped with a vacuum breaker arrangement similar to that at Sequoyah.

To resolve the staff's concern about the difference in the spray flow rates, the applicant stated that the Watts Bar spray system is capable of providing a flow rate equivalent to that assumed in the Sequoyah analyses (4750 gpm per train). The matter of vacuum breakers arose because containments such as those at Sequoyah and Watts Bar have relatively low design pressures in the reverse direction, and combustion of large amounts of hydrogen with eventual depressurization can result in containment atmosphere pressures lower than ambient pressure. The applicant stated that based on the application of ASME Code Case N284, Service Level D, a pressure reduction of 4 psi below atmospheric pressure would not affect containment integrity. The staff reviewed this issue and concluded that the Watts Bar containment is capable of safely withstanding a reverse pressure differential of 3.4 psi based on ASME Code Case N284, Service Level C. Furthermore, the staff concluded that the containment capability is adequate to deal with depressurization resulting from hydrogen combustion. Thus, the staff's concern regarding these two issues has been resolved.

The igniter system at Watts Bar is referred to as the hydrogen-mitigation system (HMS) and consists of 68 igniters located throughout the containment. The igniters are intended to burn hydrogen in a controlled fashion if hydrogen reaches the lower flammability limit; thereby preventing the accumulation of explosive concentrations. The igniters are powered by Class 1E power panels that have normal and alternate power supplies from offsite sources. In the event of a loss of offsite power, the igniters would receive power from the emergency diesel generators. When activated, they have a surface temperature of approximately 1700°F. The igniters are strategically located throughout the containment.

The resolution of USI A-48 determined that implementation of the final hydrogen-control rule published on January 25, 1985, provided adequate hydrogen-control measures for degraded-core accidents and no new regulatory requirements or guidance for such accidents were necessary. To ensure compliance with the final rule, the applicant will further investigate the adequacy of the HMS for accident scenarios developed during the performance of the Individual Plant Examination for severe accident vulnerabilities (Generic Letter 88-20). The staff will report its findings under TAC 74488.

On the basis of its review of the HMS at Watts Bar, the staff concludes that the requirements of the final rule (10 CFR 50.44(c)(3)) have been adequately implemented by the applicant and that the HMS (igniter system) is acceptable. If further modifications are shown to be required by the Individual Plant Examination, the staff will provide its evaluation of such modifications. Proposed License Condition 9 is, therefore, considered resolved. Hydrogen-control measures (hydrogen recombiners) for design-basis accidents were evaluated and found acceptable in Supplement 4 to the Watts Bar SER.

11 RADIOACTIVE WASTE MANAGEMENT

11.3 Gaseous Waste Management

SRP Section 11.3 (NUREG-0800, July 1981) provides that, for systems not designed to withstand a hydrogen explosion, dual gas analyzers with automatic control functions are required to preclude explosive hydrogen/oxygen mixtures. The staff's review of the FSAR up to Amendment 67 reveals that the Watts Bar gaseous waste processing system is not designed to withstand a hydrogen explosion. The system provides for automatic monitoring of hydrogen and oxygen in the gas space of the volume control tank, pressurizer relief tank, holdup tanks, evaporators, gas decay tanks, reactor coolant drain tank, and spent resin storage tank. If the oxygen content exceeds a predetermined level, an alarm will sound in the control room, alerting the operator to the condition. When this happens, the decay tank being filled will be isolated and operator action will be required to direct flow to a standby tank. Gas can be transferred from one decay tank to another. Nitrogen diluent can be introduced into the system to reduce the potential for a hydrogen explosion. The hydrogen and oxygen monitoring system does not meet the provisions of the SRP because dual monitors are not provided and the system is not designed to automatically initiate action to reduce the potential for explosion in the event of a high oxygen content.

The staff has undertaken a study (Generic Issue 106, "Highly Combustible Gases in Vital Areas") to evaluate problems associated with explosive gas mixtures in pressurized-water-reactor waste gas systems. The study addresses the waste gas systems in use, monitoring instrumentation available and in use, operating experience, and the flamability and detonability of hydrogen/air mixtures. The study has not been completed, but on the basis of the work accomplished to date, the staff tentatively concludes that the design of the Watts Bar hydrogen and oxygen monitoring system provides a level of safety that is commensurate with the risks and consequences of a hydrogen/oxygen explosion and is, therefore, acceptable on an interim basis until the staff completes this study.

By letter dated January 25, 1983, the applicant stated that there exists a significant safety concern with having a technical specification that requires that the reactor be shut down within a specified time when the hydrogen or oxygen monitors are out of service. This would lead to degassing at a time when both monitors are inoperable. The staff has evaluated this requirement and concludes that an equivalent level of protection (rather than reactor shutdown) against the uncontrolled release of radioactive materials would be provided for Watts Bar by a technical specification. This would require that grab samples be collected at least once during a batch transfer to a waste gas decay tank and at least once every 4 hours and analyzed within the following 4 hours when a hydrogen or oxygen monitor is inoperable. A special report will have to be submitted to the staff when either monitor is inoperable for more than 7 days or both monitors are inoperable. These requirements will be considered for incorporation into the Watts Bar Technical Specifications, currently being developed.

The above review was tracked by TAC 63644.

11.6 Evaluation Findings

11.6.1 Offsite Radiological Monitoring Program

By letter dated January 6, 1989, the applicant proposed a reduced meteorological and environmental monitoring program because of the extensive preoperational data previously collected and the target schedule at that time of approximately 2 years for the completion of the Watts Bar nuclear plants. Thus, in January 1989, the Watts Bar monitoring program was reduced. The reduced program basically discontinued the collection of atmospheric samples, reduced the frequency of milk sampling from semimonthly to monthly, and reduced the frequency of all monthly sampling to quarterly. In its June 6, 1989, response, the staff indicated that the duration proposed by the applicant for the reduction or discontinuation of the monitoring program was unacceptable. In its July 20, 1989, letter, the applicant reevaluated the program and is in agreement with the staff's position as stated in NRC Regulatory Guide 4.1 regarding preoperational radiological environmental monitoring programs (REMPs).

By letter dated June 14, 1991, the applicant stated that it has resumed the full preoperational REMP. The collection of all atmospheric samples was resumed, the frequency for collecting milk samples was increased from monthly to semimonthly, and the frequency of all sampling that had been reduced from monthly to quarterly was reinstated to monthly. All other quarterly, semiannual, and annual sampling continued as before.

The staff reviewed the applicant's submittals mentioned above, and Section 11.6 of the FSAR as amended up to Amendment 67 (Section 11.6 was in reality substantively updated by Amendment 56). The following review findings (five paragraphs) have been issued to the applicant by letter dated September 10, 1991:

TVA is conducting the preoperational REMP to provide for measurement of background radiation levels and radioactivity in the plant environs. The REMP, which provides a necessary comparative basis for the operational radiological monitoring program, will also permit TVA to train personnel, evaluate procedures, equipment, and techniques, as indicated in Regulatory Guide 4.1. The preoperational program for [Watts Bar] was implemented in December 1976, and subsequently revised in 1986, to comply with [Watts Bar's] draft Technical Specifications. The program follows the staff's guidelines described in the Branch Technical Position (BTP) on environmental monitoring. Highlights of TVA's REMP include monitoring of air at the offsite locations where the highest concentrations of radionuclides are expected, placement of dosimeters in two concentric rings around the plant, [taking] water samples (i.e., surface, ground, and drinking), [taking] upstream and downstream milk samples at locations where the highest doses are expected, and [taking] various food samples. Lower limits of detection (LLDs) for the various types of samples and nuclides are specified. The LLDs are consistent with those specified in NUREG-0472, "Standard Radiological Effluent Technical Specifications for PWRs."

During the sampling period, a small number of samples were not collected and several analyses were not completed on some collected samples. The missed samples and analyses were the result of equipment malfunction, sample unavailability, the scarcity of sample media, and the lack of sufficient quantities of sample for complete analyses. In addition, one dairy farm went out of business and was deleted from the monitoring program. The clam population was diminishing, which led to reduced sampling at several locations. A list of missed samples, missed analyses, causes, and remedies to prevent recurrence was given in a related table in the applicant's June 14, 1991, letter. The missed samples and analyses are not unexpected for preoperational REMP. The applicant has taken corrective action which the staff finds acceptable.

A quality assurance (QA) program has been implemented by the applicant to ensure that the environmental monitoring data are reliable. This program includes the use of written, approved procedures in performing the work, a nonconformance and corrective action tracking system, systematic internal audits, a complete training and retraining system, audits by various external organizations, and a laboratory quality control program. The staff finds this QA program acceptable.

The applicant's Western Area Radiological Laboratory participates in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA). The program provides periodic cross-check samples of the type and radionuclide composition normally analyzed in an environmental monitoring program. The results obtained in the monitoring program and the EPA's program are reported annually to NRC.

The staff concludes that the Watts Bar preoperational REMP proposed by the applicant is adequate to provide baseline data which will assist in verifying radioactivity concentrations and related public exposures during plant operation, and is therefore acceptable. This review was tracked by TAC 77661.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.3 Plant Staff Organization

In the SER, the staff stated that Proposed License Condition 26 will be imposed in the operating license to ensure that the applicant will augment the Watts Bar shift staff with individuals who have prior experience with large pressurized-water-reactor (PWR) operations. Specifically, the staff would require on each shift a licensed senior reactor operator who has had at least 6 months of hot operating experience at the same type of plant, including at least 6 weeks at power levels greater than 20 percent of full power, and who has had startup and shutdown experience.

By Amendment 63 to the FSAR, and the applicant's Nuclear Quality Assurance Plan (submitted to NRC by letter dated February 15, 1990, staff review documented in SSER 5), the applicant committed to comply with the guidelines of Regulatory Guide 1.8, "Personnel Selection and Training," Revision 2, April 1987. The staff considers this commitment to provide adequate assurance that the intended requirements of Proposed License Condition 26 will be met. This eliminates the need for the proposed license condition.

13.4 Review and Audit

Onsite Review

In the SER, the staff stated that it will require the applicant, through Proposed License Condition 26, to establish an independent safety engineering group (ISEG). It is the staff's current practice to address the ISEG issue as part of the plant Technical Specifications, which are under development. Since specific requirements of the ISEG will be imposed via the Watts Bar Technical Specifications or related documents, the staff considers Proposed License Condition 26 resolved.

APPENDIX A

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

<u>NRC Letters, and Summaries</u>

June 6, 1989	Letter, S. C. Black to O. D. Kingsley (TVA), on Meteorological and Environmental Monitoring Program.
July 20, 1989	Letter, S. Black to O. D. Kingsley (TVA) on Environmental and Meteorological Monitoring Program for Watts Bar Nuclear Plant.
February 6, 1991	Summary of January 29, 1991, meeting with utility regarding the seismic classification of cable trays and conduit.
June 6, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding interim safety evaluation of FSAR Section 14.2.
June 18, 1991	Letter, F. J. Hebdon to D. A. Nauman (TVA), forwarding environmental assessment and finding of no significant impact regarding May 16, 1991, request to extend construction permit.
June 20, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on FSAR Section 8.
June 26, 1991	Letter, F. J. Hebdon to D. A. Nauman (TVA), forwarding safety evaluation on Topical Report TVA-NPOD89, "Nuclear Power Organization Description."
June 27, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding order to extend construction permit to December 31, 1993, for Unit 1 and to June 30, 1997, for Unit 2.
July 1, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information to address Outstanding Issue 19(j).
July 9, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on FSAR Section 2.5 regarding stability of subsurface materials.
July 11, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on FSAR Section 4.2
July 19, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting support for site review of FSAR Chapter 8.

July 29, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting support for site review of compliance with ASME Code Section III, Subsection ND-2000.
July 29, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation on Westinghouse Owners Group methodology regarding thermal stratification.
July 31, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation on dynamic and static load on main steam safety valves.
August 6, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation on FSAR Chapter 15 analysis regarding zero-power rod withdrawal in Mode 3.
August 10, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation accepting Offsite Radiological Monitoring Program.
August 13, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation on the Special Program on Microbiologically Induced Corrosion.
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August 21, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), forwarding staff position regarding resolution of Bulletin 88-05.
August 26, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information concerning compliance with Regulatory Guide 1.97.
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September 18, 1991	Letter, F. J. Hebdon to D. A. Nauman (TVA), forwarding copies of SSER 7.
September 19, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), advising that response to Bulletin 88-08 is acceptable.
September 23, 1991	Letter, F. J. Hebdon to D. A. Nauman (TVA), granting relief from specific ASME Code requirement as requested in TVA's April 11, 1991, letter.

September 26, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), informing that response to Bulletin 89-01, Supplement 2, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," is acceptable.
September 30, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), informing that response to Outstanding Issue 19(f) concerning mass eccentricity of steel containment is acceptable.

TVA Letters

November 3, 1986	Letter, R. Gridley to B. Youngblood, regarding elimination of resistance temperature detector bypass and Eagle-21.
December 1, 1986	Letter, J. A. Domer to B. Youngblood, regarding elimination of resistance temperature detector bypass and Eagle-21.
January 27, 1987	Letter, R. Gridley to B. J. Youngblood (NRC), regarding elimination of resistance temperature detector bypass and Eagle-21.
March 17, 1987	Letter, R. Gridley to S. Ebneter (NRC), regarding elimination of resistance temperature detector bypass and Eagle-21.
January 6, 1989	Letter, R. Gridley to NRC Document Control Desk, regarding radiological and environmental monitoring program.
July 20, 1989	Letter, M. J. Ray to NRC Document Control Desk, regarding radiological and environmental monitoring program.
December 18, 1990	Letter, E. G. Wallace to NRC, regarding documentation of resolution to open issues--FSAR Amendment 64.
January 31, 1991	Letter, E. G. Wallace to NRC, forwarding a revised response to NRC request for information on IE Bulletin 79-02.
April 18, 1991	Letter, E. G. Wallace to NRC, regarding minimum load factors for pipe supports.
May 8, 1991	Letter, E. G. Wallace to NRC, forwarding responses to NRC request for additional information on Watts Bar Amendments 54 through 64.
June 3, 1991	Letter, E. G. Wallace to NRC, transmitting FSAR Amendment 66.
June 6, 1991	Letter, E. G. Wallace to NRC, transmitting response to NRC questions on FSAR Section 3.8
June 7, 1991	Letter, E. G. Wallace to NRC, transmitting response to questions asked during seismic analysis audit of April 15-19, 1991.
June 13, 1991	Letter, E. G. Wallace to NRC, transmitting response to questions asked during seismic analysis audit of April 15-19, 1991.

June 13, 1991	Letter, E. G. Wallace to NRC, providing additional information on Outstanding Issue 19(b) regarding multimode factor.
June 14, 1991	Letter, E. G. Wallace to NRC Document Control Desk, transmitting the 1990 Watts Bar Radiological Environmental Monitoring Report.
June 21, 1991	Letter, E. G. Wallace to NRC, providing additional information on dynamic and static loads on main steam safety valves.
June 28, 1991	Letter, E. G. Wallace to NRC, providing implementation details for Conduit, Cable Tray, and HVAC Corrective Action Programs (CAPs).
July 1, 1991	Letter, E. G. Wallace to NRC, providing commitment to supersede Proposed License Condition 31 on startup tests.
July 2, 1991	Letter, E. G. Wallace to NRC, responding to request for information on the QA Records CAP.
July 9, 1991	Letter, E. G. Wallace to NRC, providing additional information regarding removal of the resistance temperature detector bypass system.
July 10, 1991	Letter, E. G. Wallace to NRC, notifying of plan to use Westinghouse Eagle-21 process protection system.
July 10, 1991	Letter, E. G. Wallace to NRC, providing information to address staff concerns on FSAR Section 14.2 regarding startup tests.
July 11, 1991	Letter, E. G. Wallace to NRC, providing analysis on natural circulation cooldown.
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July 22, 1991	Letter, E. G. Wallace to NRC, providing notification of changes to ECCS analysis model.
July 26, 1991	Letter, E. G. Wallace to NRC, providing additional information on implementation of Bulletin 79-02 requirements.
July 31, 1991	Letter, E. G. Wallace to NRC, forwarding "Watts Bar Nuclear Plant Pipe Support Minimum Design Load Evaluation" final report.
July 31, 1991	Letter, J. H. Garrity to NRC, providing a list of ASME code cases used for piping analysis.

August 1, 1991	Letter, J. H. Garrity to NRC, providing additional information in response to Bulletin 88-05.
August 1, 1991	Letter, J. A. Garrity to NRC, responding to Inspection Report 50-390/91-201 regarding integrated design inspection.
August 5, 1991	Letter, D. A. Nauman to NRC, informing of discontinuance of Program Team.
August 6, 1991	Letter, E. G. Wallace to NRC, providing additional information on Bulletin 89-01.
August 8, 1991	Letter, E. G. Wallace to NRC, forwarding FSAR Amendment 67.
August 12, 1991	Letter, J. H. Garrity to NRC, providing additional information on FSAR Section 4.2 regarding seismic qualification of reactor vessel internals.
August 22, 1991	Letter, J. H. Garrity to NRC, providing additional information on Outstanding Issue 6 regarding equipment seismic qualification.
August 22, 1991	Letter, J. H. Garrity to NRC, providing additional information on implementation of the Cables CAP.
August 22, 1991	Letter, J. H. Garrity to NRC, providing additional information on cable tray classification, conduit damping, and feedwater check valve slam analysis.
August 22, 1991	Letter, J. H. Garrity to NRC, responding to questions on FSAR Section 2.5.
August 22, 1991	Letter, J. H. Garrity to NRC, responding to Outstanding Issues 18 and 19.
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August 30, 1991	Letter, J. H. Garrity to NRC, providing list of civil/structural calculations completed.
September 6, 1991	Letter, D. A. Nauman to NRC, forwarding Revision 1 to Volume 4 of "Watts Bar Nuclear Performance Plan."
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September 20, 1991	Letter, J. H. Garrity to NRC, addressing NRC concern on Category 1(L) piping.

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September 23, 1991	Letter, E. G. Wallace to NRC, addressing commitments to Regulatory Guides 1.33 (Rev. 2) and 1.38 (Rev. 2).
September 24, 1991	Letter, J. H. Garrity to NRC, responding to July 28, 1986, questions on the fifth diesel generator.
September 30, 1991	Letter, J. H. Garrity to NRC, providing Revision 1 to "Watts Bar Plant Pipe Support Minimum Design Load Evaluation" final report.
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APPENDIX B

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APPENDIX C

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

This appendix updates the NRC staff's evaluation of one unresolved safety issue (USI) that is applicable to the Watts Bar facility.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

This issue has been resolved by NUREG-1370, Resolution of Unresolved Safety Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," dated September, 1989. Refer to Section 6.2.5 of this supplement for an evaluation of the igniter system used at Watts Bar to meet the requirements of 10 CFR 50.44(c)(3)(iv) and be consistent with the resolution of USI A-48.

APPENDIX E

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APPENDIX Q

SAFETY EVALUATION: MICROBIOLOGICALLY INDUCED CORROSION PROGRAM

ISSUED BY LETTER, P. S. TAM TO D. A. NAUMAN (TVA),
SEPTEMBER 13, 1991

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MICROBIOLOGICALLY INDUCED CORROSION PROGRAM

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-390 AND 50-391

1.0 INTRODUCTION

In the early 1980s, corrosion product buildup in the essential raw cooling water (ERCW) system resulted in replacement of the carbon steel piping with 316 austenitic stainless steel. In addition, the main yard ERCW headers were lined with concrete. Technical Instruction TI-27 was implemented to assure system cleanliness during ERCW breaches and TI-31.13 was instituted to monitor wall thinning resulting from cavitation, two-phase erosion/corrosion, microbiologically induced corrosion (MIC) and generalized corrosion.

In August 1986, MIC caused leaks in the butt welds of 12-inch diameter, 316 austenitic stainless steel ERCW piping. The response to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment" was transmitted to NRC on January 26, 1990. TI-36, issued in August 1990, describes the program for monitoring, detection, assessment of extent, and control of MIC.

In April 1991, TI-106 was issued to evaluate welds with leaks by a representative sample of ERCW welds which are radiographed to trend MIC growth in the remaining ERCW welds. The applicant submitted the MIC program report on February 26, 1991. On January 11, 1991, the NRC staff visited Watts Bar for a discussion and on-site review of the MIC program. The staff has reviewed TI-27, 31.13, 36 and 106 and the applicant's letters dated January 26, 1990 (response to Generic Letter 89-13), and February 26, 1991 (MIC program report), and presents the finding in this safety evaluation.

2.0 EVALUATION

A. Inspection

TI-31.13, "Wall Thinning Monitoring Program for Cavitation, Microbiologically Induced Corrosion, and Dual Phase Erosion/Corrosion," is utilized to obtain information to determine repair or replacement intervals to prevent through-wall leaks and to trend the rate of degradation. TI-31.13 provides recommended ERCW inspection locations by ultrasonic testing (UT) for cavitation, MIC, and two-phase erosion/corrosion.

TI-106 addresses radiographic examination (RT) of butt-welded ERCW stainless steel piping to evaluate MIC damage. It also provides data to trend MIC damage in the ERCW welds. Based on the RT results, the structural integrity of the welds will be evaluated. All leaking welds identified will be repaired or replaced during the next scheduled outage exceeding 30 days but no later than the next refueling outage.

B. Surveillance

TI-36 describes the program for monitoring, detection, assessment of the degree, and the control of MIC at the Watts Bar Nuclear Plant. This program encompasses:

- the monitoring of plant systems for biological activity and the degree of corrosion after biocide treatment,
- the assessment of the degree of MIC, the degree of stainless steel butt weld damage, and the distribution of carbon steel piping damage,
- the control of MIC by appropriate water treatment, and
- the control of MIC during layup.

A comprehensive corrosion monitoring program has been established to monitor the effectiveness of the biocide and dispersant/corrosion inhibitor treatment. The monitoring program includes:

- Betz Cosmos portable corrosion monitor
- Weight loss coupon racks
- Total residual oxidant analyzer
- Visual observation test spool pieces
- Sessile bacteria bead monitors

Monitoring of ERCW with surveillance coupon, biocide levels, electrochemical probes, and representative chemical samples provides valuable data of the effectiveness of the treatment program and can alert the operator to changes in either the environment or the corrosion response of materials.

C. Leak Position

The staff's position on continued operation after detection of a leaking pipe is that a repair/replacement in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI is required. If the applicant desires relief from ASME Code Section XI repair/replacement requirements, the provisions of Generic Letter 90-05 should be followed.

D. Cleaning

TI-27 provides the criteria for internal and external surface cleaning and cleanliness of fluid systems and components during initial installation modification, and maintenance activities. This TI was implemented to ensure system cleanliness during system breaches.

E. Treatment

Biocide treatment will be implemented when the microbiological level is greater than 10^4 CFU (colony-forming units)/ml. Visual examinations are periodically performed on carbon and stainless steel outside surfaces of components for through-wall leaks and nodule deposits. Non-destructive examinations (NDE) are performed on plant systems and/or components based on the results of biological monitoring. If a system that contains stainless steel pipings is found to have a through-wall leak at welds and heat-affected zones, the defect size and distribution is determined for a structural integrity analysis to determine the remaining margin. A system that contains carbon steel piping is evaluated to determine that the minimum wall thickness meets the requirements specified in the design calculations.

The applicant has installed a bromine/chorine biocide injection system for treatment of the new water system, including the ERCW. Hypobromous (HOB) and hypochlorous acid (HOCL) are injected into the raw water cooling system by passing a side stream through a bed of granular bromo-chloro-dimethyldantoin. An on-line dispersant/corrosion inhibitor treatment system has been selected to chemically treat iron and MIC corrosion deposits. The chemical composition consists of zinc sulfate and tetra-sodium polyphosphate corrosion inhibitors with either a polyphosphate or dimethyl amide as a penetrant/dispersant or other equivalent. The applicant indicates that it may take one to two years to clean up the MIC deposits. The applicant has indicated that the on-line dispersant/corrosion inhibitor treatment was selected to prevent blockage or damage to instrumentation or equipment.

There is a concern about initiating biocide treatment without prior cleansing of a system. It is important that fouled systems be cleaned as a first step for mitigation of corrosion. It is then possible that treatments can be effective in preventing the recurrence of the problem (Reference 1). Addition of most inhibitors to treat MIC is unlikely to have any effect at all unless the biological growth has been removed from the surfaces either mechanically or chemically, and the microbial infestation has been controlled (Reference 2). It is to be stressed, however, that the cleaning method employed must completely remove the slime, scale and other material, since if some material is left over, corrosion in the pit may proceed (Reference 3). One of the components in the applicant's proposed dispersant/corrosion inhibitor treatment program (polyphosphates) can be broken down by some microorganisms rendering it ineffective as a corrosion inhibitor (Reference 3). Some biocides and corrosion inhibitors are ineffective in penetrating the MIC tubercles and therefore, are ineffective in controlling MIC corrosion. Ozone and hydrogen peroxides, strong oxidizing biocides with greater penetrating power than chlorine, have been used for treatment of existing MIC. To assure effective (i.e., immediate) MIC control, mechanical or chemical cleaning of the ERCW appears necessary. The use of online dispersant/corrosion inhibitor treatment in conjunction with biocide treatment may take up to 24 months to attain effective MIC control.

3.0 CONCLUSION

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

Principal Contributor: F. Witt

Date: September 13, 1991

References

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2) EPRI ER.6345, "Microbial Corrosion: 1980 Workshop Proceedings," Section 7, "Cleaning Methods and Philosophy of Cleaning to Prevent or Mitigate Microbiological Influenced Corrosion," B.D. Fillers, April 1989.

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APPENDIX R

SAFETY EVALUATION: EAGLE-21 SYSTEM

ISSUED BY LETTER, S. BLACK TO O. D. KINGSLEY (TVA)
JUNE 13, 1989

ENCLOSURE 1

SAFETY EVALUATION REPORT
EAGLE-21 SYSTEM
WATTS BAR UNIT 1

1.0 BACKGROUND

Improved electronics technology and accumulated operating plant experience have led to the development of a new design to replace the RTD bypass system for reactor coolant system (RCS) temperatures. The benefits attributable to the RTD bypass elimination modification fall into three primary areas: reduced radiation exposure, improved availability, and reduced maintenance. As a result of removing the bypass piping the Tennessee Valley Authority (TVA) states that, radiation exposure to personnel can be reduced on the average of 80 manrems per outage. Availability can be improved by avoiding forced outages attributed to the present RTD bypass system. Maintenance requirements can be reduced by eliminating hardware which require periodic maintenance and inspection.

For the Watts Bar design, the Eagle-21 qualified microprocessor based equipment is being utilized for this RTD bypass elimination. In all, the Eagle-21 process protection system replacement hardware performs the following major functions:

1. Reactor trip protection (channel trip to voting logic)
2. Engineered safeguard features (ESF) actuations.
3. Isolated outputs to control systems, control panels, and plant computers.
4. Isolated outputs to information displays for post-accident monitoring (PAM) indication.
5. Automatic surveillance testing to verify channel performance.

The staff performed a comprehensive review of the hardware and software design aspects of the Eagle-21 System. This included a review of the Verification & Validation (V&V) program on the Eagle-21 System to ensure the functionality of the system commensurate with that described in the system requirements. Two staff audits were conducted on the Eagle-21 System design and V&V process. The first audit was held in February 1987, and the primary areas of discussion were the V&V plan and the system design. The second audit was held in April 1989, and the primary areas of discussion were the resolution of issues from the first audit, the verification process, the validation process, and the system design. The results of these audits are presented below.

2.0 EAGLE-21 SYSTEM DESCRIPTION AND DESIGN REVIEW

The mechanical modification removes the valves, piping, snubbers, and supports associated with the RTD bypass system and replaces them with thermowell mounted fast response RTDs which are installed directly into the Reactor Coolant Pipe. Mechanical modifications begin with the removal of the existing bypass piping at each connection point to the Reactor Coolant System. The existing hot and cold leg penetrations are machined to accept RTD thermowells. On the hot leg, the scoop tip will be removed to allow the thermowell to protrude directly into the flow stream. The thermowell is installed inside the modified scoop and the RTD is installed within the thermowell. The crossover leg connection is capped and an additional cold leg boss, thermowell and RTD are added as an installed spare.

The Eagle-21 family of qualified microprocessor based equipment is utilized to electronically average three hot leg RTD's to obtain a single hot leg average temperature (TH_{AVG}). The system used to calculate this average temperature is referred to as the temperature averaging system (TAS). The temperature averaging system (TAS) becomes part of the thermal overpower and overtemperature protection system (Delta T / T_{AVG} Protection) because TAS output (TH_{AVG}) replaces the hot leg temperature signal previously measured in the bypass manifold RTD. The TH_{AVG} signal is used in the calculation of the delta temperature (Delta T) and average temperature (T_{AVG}). The modular design of the Eagle-21 electronics allows for installation of the digital hardware into existing process racks. One rack per protection channel set is configured primarily for Delta T / T_{AVG} protection. However, other analog signals such as neutron flux from upper and lower ion chambers, and pressurizer pressure are routed to the Eagle-21 loop processor. All analog hardware with the exception of the field termination blocks will be removed from these racks and be replaced with Eagle-21 digital electronics.

2.1 Redundancy and Isolation

The Eagle-21 Process Protection System is designed to provide redundant instrumentation channels and outputs to two trip logic trains for each protective function. These redundant channels and trains are electrically isolated and physically separated. Thus any single failure within a channel or train will not prevent a required protection system action. The Eagle-21 Process Protection System is independent from the control system. The transmission of signals from the Eagle-21 to the control system is through qualified isolation devices. The results of the fault testing of the isolation devices was provided in WCAP-11733 "Noise, Fault, Surge, and Radio Frequancy Interference Test Report" (dated June 1988) with clarifying information provided by a letter dated May 22, 1989. The Eagle-21 System uses the output signal conditioning boards as an isolation barrier between field level signals and the microprocessor subsystem. The Eagle-21 uses the following types of isolation devices for interfacing Class 1E signals with Non-Class 1E equipment:

Isolator Board Type	Isolation Device
Analog Output Board (current loop)	Transformer
Digital contact Output Board	Relay
Partial Trip Output Board	Optical Isolator

In addition, high voltage transient protection is provided for each cabinet input/output, including the ac power feed, by transient suppression circuitry.

The purpose of the fault tests was to demonstrate that credible faults injected into the Non-Class 1E system do not propagate across the Non-Class 1E to Class 1E isolation barrier or from channel to channel within the Eagle-21 process rack. These tests were designed to verify that the Eagle-21 system isolation devices are in compliance with IEEE-279-1971, IEEE-384-1981, and Regulatory Guide 1.75, Rev. 2 concerning the physical independence of Class 1E circuits and Class 1E/Non-Class 1E interaction.

Maximum credible fault voltages were determined to be 580 Vac and 250 Vdc per previous protection system tests (7300 system, Qualified Display Processing System). In addition, 125 Vac and 125 Vdc tests were performed where applicable. A fault was considered applicable only if the fault challenged the nominal voltage or current ratings of the channel under test. For cases where 125 Vac and 125 Vdc tests were considered not applicable, 580 Vac and 250 Vdc tests were performed thus enveloping the lower voltage tests.

The Surge Withstand Capability (SWC) tests were conducted under normal operating conditions of the system in accordance with IEEE-472-1974. The purpose of this test was to show: (1) the protective actions of the Eagle-21 System are not affected by application of the surge withstand test wave to the designated Non-Class 1E to Class 1E isolators, (2) that no component failures occurred, and (3) that no change in channel calibrations occurred due to the application of the surge withstand test wave. All system inputs/outputs were surge tested in the common and transverse modes including the system power supply input circuitry.

All of the isolators passed the pass/fail criteria for all of the tests noted above. Therefore, the requirement that the isolators protect the Class 1E side of the isolator is satisfied and the requirements of General Design Criterion

(GDC) 25 and IEEE-STD-279-1971 regarding isolation are met. The staff concludes that the isolation devices are acceptable.

2.2 Bypass and Testing

The Eagle-21 Process Protection System performs automatic surveillance testing at the digital process protection racks via a portable Man Machine Interface (MMI) test cart. The MMI test cart is connected to the process rack by inserting a connector into the process rack test panel. Using the MMI, the "Surveillance Test" option is then selected. Following instructions entered through the MMI, the rack test processor automatically performs calibrations, Analog to Digital convertor tests, response time tests and dynamic algorithms and bistable setpoint accuracy tests.

Interruption of the bistable output to the logic circuitry for any reason (test, maintenance purposes, or removal from service) causes that portion of the logic to be actuated and accompanied by a channel trip alarm and channel status light in the control room. Each channel is fully testable via the portable MMI test cart.

Status lights on the process rack test panel indicate when the associated bistables have tripped. The value (in engineering units) that caused the bistable to trip is displayed on the MMI screen.

The Eagle-21 Process Protection System provides for continuous on-line self-calibration of analog input signals. The Digital Filter Processor (DFP) provides high and low reference signals to a multiplexer circuit on each analog input channel. The DFP then compares the output of its Analog to Digital (A/D) Converters to the high and low reference signals to determine if any errors have been introduced by analog signal processing and A/D conversion. If necessary, the DFP automatically adjusts the D/A gain and offset to eliminate any errors that have been introduced.

The Eagle-21 Process protection equipment is designed to permit any one channel to be maintained, and when required, tested during power operation without initiating a protective action at the system level. During such operation, the process protection system continues to satisfy single failure criterion.

If an Eagle-21 protection channel has been bypassed for any purpose, a signal is provided to allow this condition to be continuously indicated in the control room.

The Eagle-21 design has provided for administrative controls and multiple levels of security for bypassing a protection channel. To place a protection channel in bypass, an individual must have access to the following:

A. Man-Machine Interface test cart.

B. Keyboard for the MMI test cart.

C. Key for the process rack door. A status light on the control board alerts the operator that the protection set has been entered. If a technician opens the doors of two protection sets, the operator is alerted by an annunciator.

D. Key for the rack mounted test panel selector switch.

E. Password that is entered through the MMI keyboard.

The Eagle-21 design has provided for administrative controls and multiple levels of security for access to setpoint and tuning constant adjustments. In order to adjust a setpoint or tuning constant in the Eagle-21 system, an individual must have access to A through E as stated above and, in addition, must have knowledge of the allowable range for the specific parameter to be updated.

2.3 Diagnostics

The Eagle-21 Process Protection equipment provides specific diagnostic information to the user via numerous printed circuit cards and test panel status LEDs, as well as information available through the portable Man-Machine-Interface (MMI). This design feature allows for easy recognition, location, replacement, and repair or adjustment of malfunctioning components or modules.

2.4 Equipment Qualification

The Equipment Qualification Program demonstrated that the Eagle-21 equipment is capable of performing its designated safety-related function under the requir environmental and seismic conditions. This was accomplished by testing as follows:

(1) Environmental testing (IEEE-STD-323-1974)

(2) Seismic testing (IEEE-STD-344-1975)

The tests and results were documented in WCAP-8687, "Eagle-21 Process Protection System (Environmental and Seismic Testing)," dated May 1988.

Noise, Fault, Surge Withstand Capability, and Radio Frequency Interference (RFI) tests demonstrated that the Eagle-21 equipment is capable of performing its designated safety related function when subjected to specified test conditions. The tests and results were documented in WCAP-11733, "Noise, Fault, Surge, and Radio Frequency Test Report for Eagle-21 Process Protection Upgrade System," dated June 1988.

The Eagle-21 equipment was subjected to the following noise sources:

- o Random Noise Test (Antenna Coupled)
- o Crosstalk Noise - Chattering Relay Test (Antenna and Direct Coupled)
- o Military Specification MIL-N-19900B Noise Test (Antenna Coupled)
- o High Voltage Transient Noise Test (Antenna Coupled)
- o Static Noise Test (Antenna and Direct Coupled)

For the random, high-voltage transient, and military specification noise tests, the noise source was antenna coupled to the Non-Class 1E field wiring under test. The noise source was applied to a 40-foot antenna wire adjacent to a 40-foot length of unshielded Non-Class 1E field wiring. The antenna was brought directly into the cabinet and bundled with Class 1E input/output cables upon entering the process rack.

The Non-Class 1E test cable was terminated with a nominal load value. The crosstalk and static noise tests were conducted similarly, except that an additional test was performed where the noise source was applied directly to the Non-Class 1E wiring. To prevent damage, the isolator was disconnected at the Eagle-21 termination frame and the disconnected Non-Class 1E wires shorted to complete the cross talk noise circuit loop. The disconnected Non-Class 1E wires were open-circuited for the static noise test.

The purpose of the Radio Frequency Interference (RFI) susceptibility test was to evaluate the performance of the system when subjected to electromagnetic fields such as those generated from portable radio transceivers or any other devices that generate continuous-wave radiated electromagnetic energy. The Eagle-21 System remained operational while exposed to RFI. Analog input/output processing and protective action functions were affected but demonstrated full recovery upon removal of the RFI. To avoid system perturbations, the vendor has recommended that the Eagle-21 System equipment rooms be zoned to prohibit the use of transceivers in the 20-700 MHz band. TVA has stated that to alleviate this concern, the use of transceivers would be prohibited during plant operation in these equipment rooms. The staff concludes that the RFI test results concurrent with the ban of transceivers in the 20-700 MHz band is acceptable.

2.5 Reliability

An availablity assessment of the Eagle-21 equipment versus the equivalent analog process protection systems was performed by the vendor. The results of this assessment were provided to the staff during the second audit and illustrated that the Eagle-21 digital system availability was equal to or higher than the equivalent analog system availability. Therefore, it was concluded that the reliability of the Eagle-21 system is at least as reliable as and perhaps even more reliable than the analog system. Furthermore, it was believed that by incorporating the fail-safe design principal, redundancy, functional diversification and test features of the Eagle-21 system, its availability results would show further improvement. The staff concluded that the issue regarding the Eagle-21 reliability was resolved. This conclusion was based on our analysis of the vendor's Eagle-21 reliability study.

3.0 Eagle-21 Software Description and Review

The Eagle-21 hardware has been designed in a modular fashion. The basic subsystems are:

1. Loop processor controller

   The Loop process controller receives a subset of the process signals, performs one or more of the protection algorithms, and drives the appropriate channel trip (or partial engineered safeguards actuation) signals. It also drives the required isolated outputs.

2. Tester subsystem

   The tester subsystem serves as the focal point of the human interaction with the channel set. It provides a user-friendly interface that permits test personnel to configure (adjust setpoints and tuning constants), test, and maintain the system.

3. Input/Output (I/O)

The microprocessor based system interfaces with the field signals through various input/output (I/O) modules. The modules accommodate the plant signals and test inputs from the Tester Subsystem, which periodically monitors the integrity of the Loop Processor Subsystem.

The separation of these three elements into separate buses and microprocessors reduces the probability of interaction between them.

The purpose of the first audit (February 3-4, 1987) was to review the Eagle-21 design process and perform an evaluation of the V&V plan. By letter dated March 24, 1989 from R. Gridley to U.S. Nuclear Regulatory Commission pertinent information was provided to the staff. Included was a revised Design, Verification and Validation Plan (Rev. 2) dated February 25, 1989. A second audit regarding the utilization of the Eagle-21 hardware, the resolution of the concerns that remained from the first audit, and the results of the verification and validation process was performed by the staff on April 24, 25, and 26, 1989.

Building upon the experience gained in performing software verification and validation (V&V) on the IPS prototype and implementing the process, a much improved program was defined for the South Texas Qualified Display Processing System (QDPS). The V&V process to be implemented for Watts Bar RTD bypass elimination modification is the same as the one conducted on the South Texas QDPS, modified only to the extent of refining the process to resolve South Texas staff comments. It should also be noted that a portion of the software modules required for the Watts Bar project have already been verified as part of the South Texas V&V program.

3.1 Verification and Validation Plan

During our first audit, we evaluated the V&V plan. We compared the V&V plan to ANSI/IEEE-ANS-7.4.3.2.-1982, "Application Criteria for Programmable Digital Computer System of Nuclear Power Generating Stations". We noted in Reference 1 that the independent design verification of initial design activities and products was not clearly present in the V&V plan. The manufacturer (Westinghouse) had proposed using members of the design organization in the verification phase. However, the individual who participated in the design of a module of code would not participate in its subsequent verification. Furthermore, the plan proposed that the designers and verifiers will be able to report to the same supervisor.

This concern was resolved during the second audit when it became apparent that organizational independence (e.g., different first line supervisor) was provided for the verification and validation process. This clarification resolved this first audit concern for Watts Bar. However, the vendor has not formally incorporated organizational independence in the V&V plan so this remains an open item regarding its future use.

The staff reviewed the verification techniques associated with the Class 1E and Non-Class 1E software and found the techniques acceptable. However, the staff, as a result of the first audit, believed that all software associated with the Eagle-21 mainframe should be classified as Class 1E software and receive the highest verification level available unless the applicant can provide acceptable justification for classifying this software as Non-Class 1E. The basis for this conclusion is that it has not been shown that this particular software meets the three criteria outlined by the applicant for the determination of the safety category for the software. During the second audit, the applicant provided documentation that showed all Eagle-21 software being treated as Class 1E. As a result, the staff concluded that this first audit concern is resolved.

During our first audit, we identified a concern regarding the criteria for determining a "simple" or a "complex" unit (Section 5.4.4.2 of the Verification and Validation Plan). Revision 2 of the Eagle-21 Design Verification and Validation Plan contains revised criteria for determining whether a software unit is "simple" or "complex". The staff reviewed this revised criteria (V&V Plan, Rev. 2) and concluded that it was acceptable. The purpose of this classification is to determine the need for unit testing. All units classified as complex undergo a formal unit test program, whereas, simple units have their code reviewed and are not tested as a unit.

During the second audit, seven units of code were identified as simple units within the protection part of the system. The staff inspected each of these units and accepted their classification as a simple unit. As a result, these seven units were not subjected to a formal unit test. However six of these units were exercised during the validation testing and the resulting evidence indicated that these units performed their function properly. The code for the seventh unit was not exercised during the validation tests because it would have required a destructive test. Upon our inspection, it was determined th the unit was very short (less than six lines of code) and the logic was straight forward. Based on this inspection it was determined that the code would perform its intended function. TVA agreed to document the data and the basis for the acceptability of this unit of code. The staff concluded that this was acceptable and that the "simple" and "complex" concern was resolved.

3.2 Verification Process and Results

The verification process may be divided into two distinct phases: (1) Review of design documentation, and (2) Testing of software. The reviews consisted of design documentation review, source code review, and a functional test review. The design documentation review involves the comparison of a design document for a unit of software to the design requirement to verify performance requirements. The source code review interprets operation of the code and compares it with the expectation. In a functional test review, the verifier reviews the documentation associated with the functional tests performed by the designer of the code. Tests of the software within the verification process consisted of structural testing and functional testing. Structural testing attempts to comprehensively revise the code and its logic within a unit. The test input is chosen to exercise all the possible control paths within the unit. In functional testing, test cases are constructed from the functional properties of the program, which are documented in the design specification. Functional tests were required to evaluate modules and subsystems of code.

During the second audit, the staff verified that formal trouble reports were utilized to document all anomalies found during the verification process. The trouble reports were forwarded to the software design organization for resolution. The software is then recaptured within the verification process for the independent verification of its correct resolution.

In addition to trouble reports, clarification reports were issued when the verifier found something of a minor nature which was not significant enough to fail a unit. These were typically typographical or other minor documentation errors. The clarification reports also provided a mechanism for identifying to the designer something minor which occurred during testing. All clarification reports were satisfactorily resolved.

The verification trouble reports were assigned error codes as each report was generated. Working from a list of possible error codes used to classify previous software efforts, a significant portion (67%) of the total was made up of five error types. These were expected to be the dominant error types.

Our audit review of the documentation associated with this process confirmed that the verification plan was followed and the results were satisfactory.

3.3 Validation Process and Results

The validation process is performed to demonstrate the system functionality. This process consists of functional requirements testing, abnormal-mode testing, system prudency review/testing, and specific man-machine interface testing.

The functional requirements tests illustrate conformance to the top level functional requirements and sub-requirements as identified in a requirements matrix. Each sub-requirement has a test or series of tests to illustrate conformance.

The requirements for abnormal mode testing are established by a review of functional requirements to identify abnormal-mode conditions. Each abnormal-mode condition is identified, test criteria established, and then tested for performance.

A system prudency review was conducted when the software units and modules were integrated into a system. The prudency review resulted in system level requirements that were not obvious at the start of the design process. These requirements were integrated into a checklist called the System Prudency Checklist. The System Prudency Checklist addresses the following technical issues:

1. firmware program storage,
2. data-base information storage,
3. multiple-processor shared memory storage
4. data-link oriented system architectures, etc.

Most of these items do not relate directly to a functional requirement, but address the issue of integrated system integrity. Test cases were developed and run in response to the checklist to confirm system integrity.

The specific man-machine interface testing ensures that the operation interface used to modify the system's data-base performs properly under normal-mode and abnormal-mode data entry sequences. This is a critical area requiring special attention due to the impact on the software of the system-level information which can be modified through this interface.

A formal trouble report documented each anomaly found during validation is issued and is forwarded to the software design organization for resolution. The software is then recaptured in the verification and validation process. In our audit review of these trouble reports, we found that the reports dealt only with system level problems which was to be expected. As such, these problems would not normally have been detected during the verification process.

This conclusion reflects favorably on the verification process in that no errors were discovered that should have been detected in the verification process. In addition, the modifications to the as-coded and verified system resulting from validation testing were small in number and random in nature.

The validation and design teams identified five methods for resolving the problem reports; software changes; hardware changes; functional requirement changes; validation test procedure changes; and no problem identified. Seventy four (74) percent of the validation problem report and resolution were either test procedure changes or no problem identified.

As a further assurance that the verification and validation process was adequate, the staff conducted a walkthrough of a "thread" of information that was being used by the Eagle-21 System. The wide range pressure signal was selected for the walkthrough. The walkthrough began at the transmitter input (1PR-406) to the analog input board (1PR-406 025-08 Channel 3). The analog input board contained the power source, the surge and filter network, the test relay (IPS/406), a multiplexer, an operational amplifier used as a buffer and one used as a transformer coupled isolation device. The signal exited this board via plug J1 (pins 14 and 15) at which time it entered the loop processor subsystem at plug J3 (pins 12 and 14). There are four modules within the loop processor subsystem. The first module (DFP#1) houses the multiplexer, and the analog to digital convertor with a fixed filter and a shared memory. The second module (LCP) is the loop control processor which houses the process protection program. The third module (DAC#2) is a digital to analog convertor with a multiplexer and sample and hold output driver. The fourth module (DDC) has a parallel input/output interface along with an output driver. The third and fourth modules output the signal to an analog output card (EAO-02) and a partial trip output (EPT-01) respectively. These cards contain buffers, isolation devices, a deadman timer (trip output) which is usually set at 125ms and surge networks. At this point the signal returns to the normal path of the 7100 process system. During the walkthrough, a unit of software was selected at random and audited in detail. The V&V procedures for the unit of software were found acceptable and in conformance with the design specification. The unit of software selected was the DFP-ERROR unit which is used to set quality. Other units of software pertinent to the wide range pressure channel were reviewed but not to the level of detail as the first unit selected.

Based on the results of our first audit (Reference 1) and the results of this audit, the staff concludes that the Eagle-21 functional upgrade as implemented for Watts Bar is demonstrated to meet its functional and design requirements. Furthermore, the staff concludes that the Design, Verification and Validation Plan and resulting processes are acceptable.

3.5 Software Maintenance

The applicant has committed to utilize the Eagle-21 vendor and the existing V&V program as approved by the staff for all software maintenance/modifications. There appears to be strict control within the present V&V configuration management system and adequate procedures for issuing new system revisions are present. The applicant's present software maintenance program is acceptable. However, if in the future, the applicant proposes any changes in the software maintenance practice area, the staff will review these proposed changes based on current software regulatory guidance.

In addition, a procedure has been implemented by the applicant and the vendor which produces computer-generated labels, one for the top and one for the bottom of each PROM. This label generation occurs at the same time the code is generated that is burned onto the PROM. The purpose here is to provide a unique and unremovable identification so that the PROMs will not be inadvertently placed on the wrong boards or in the wrong place on the correct board. The applicant is required to maintain the dual PROM labeling practice for any PROM replacements. The staff concludes that the PROM identification method is acceptable.

4. CONCLUSIONS

Based on our review of information provided by the licensee, the results of the first audit (Reference 1) and the results of the second audit, the staff finc that there is reasonable assurance that the Eagle-21 system conforms to the applicable regulations and guidelines. The scope of the review included the FSAR descriptive information; electrical, instrument, and control drawings; and several Westinghouse Topical Reports. In addition, the staff met twice with the applicant and the NSSS vendor. These meetings provided a focus for exchanging information and answering staff questions. Based on the review noted above and the exchange of information at the two meetings, the staff has reached the following conclusions.

The Eagle-21 System adequately conforms to the guidance for periodic testing i. RG 1.22, "Periodic Testing of Protection System Actuation Functions," and IEEE 338, as supplemented by RG 1.118, "Periodic Testing of Electric Power and Protection Systems." The bypassed and inoperable status indication adequately conforms to RG 1.47, "Bypassed and inoperable Status Indication for Nuclear Power Plant Safety Systems." The Eagle-21 System adequately conforms to the guidance on the application of the single-failure criterion in IEEE 379, as supplemented by RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Systems." On the basis of its review, the staff concludes that the Eagle-21 System satisfies IEEE 279 with regard to system reliability and testability. Therefore, the staff finds that GDC 21 is satisfied.

The Eagle-21 System adequately conforms to the guidance in IEEE 384 as supplemented by RG 1.75, "Physical Independence of Electric Systems," for protection system independence. On the basis of its review, the staff concludes that this system satisfies IEEE 279 with regard to independence of systems and hence satisfies GDC 22.

On the basis of its review of the interface between the Eagle-21 System and plant-operating control systems, the staff concludes that the system satisfies IEEE-279 with regard to control and protection system interaction. Therefore, the staff finds that GDC 24 is satisfied. On the basis of its review of the software design and its verification and validation, the staff concludes that the Eagle-21 System satisfies the requirements of ANSI/IEEE-ANS-7.4.3.2-1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations" and Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants".

The staff's conclusions noted above are based on the requirements of IEEE 279 with respect to the design of the safety-related portion of the Eagle-21 System. Therefore, we find that 10 CFR 50.55 a (h) is satisfied. In summary, we conclude that the Eagle-21 System meets all of the applicable guidelines and regulations and that its utilization as discussed previously is acceptable. However, this acceptance is conditional on the staff's post installation inspection that verifies that the Eagle-21 system has been implemented as discussed in this SER and satisfactory completion of a pre-operational test prior to plant start-up.

5.0 REFERENCE

Memorandum from Charles E. Rossi to Tom Kenyon, "First Audit Report on RTD Bypass Loop Elimination Mitigation of Eagle-21 Electronics Watts Bar," dated March 23, 1983

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10. SUPPLEMENTARY NOTES

Docket Nos. 50-390 and 50-391

11. ABSTRACT *(200 words or less)*

Supplement No. 8 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 Evalaution of (1) additional information submitted by the applicant since Supplement No. 7 was issued, and (2) matters that the staff had under review when Supplement No. 7 was issued.

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